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### LIST OF EXTENDED SYNOPSIS

<table>
<thead>
<tr>
<th>Number</th>
<th>Title of Paper</th>
<th>Name of Main Author</th>
<th>Page Number</th>
</tr>
</thead>
<tbody>
<tr>
<td>IAEA-CN-155-001KS</td>
<td>IAEA activities on plant life management for long term operations</td>
<td>K.S. Kang, T. Inagaki</td>
<td>1</td>
</tr>
<tr>
<td>IAEA-CN-155-002KS</td>
<td>Proactive plant life management – one of TVO’s key success factors</td>
<td>P. Simola</td>
<td>2</td>
</tr>
<tr>
<td>IAEA-CN-155-003KS</td>
<td>Prospects of PLiM in China</td>
<td>M.G. Zheng</td>
<td>3</td>
</tr>
<tr>
<td>IAEA-CN-155-004KS</td>
<td>Plant life management of a large nuclear fleet</td>
<td>F. Hedin</td>
<td>6</td>
</tr>
<tr>
<td>IAEA-CN-155-005KS</td>
<td>Refurbishments &amp; life extensions: Lessons learned from the past to get it right in the future</td>
<td>J. Froats</td>
<td>8</td>
</tr>
<tr>
<td>IAEA-CN-155-006KS</td>
<td>Plant life management in German nuclear power plants</td>
<td>E. Fischer</td>
<td>9</td>
</tr>
<tr>
<td>IAEA-CN-155-007KS</td>
<td>Control of permanent set structures: Condition at lifetime extension of nuclear power plants</td>
<td>Y.I. Strombach</td>
<td>12</td>
</tr>
<tr>
<td>IAEA-CN-155-008KS</td>
<td>Economics of plant life management</td>
<td>L. Bond</td>
<td>13</td>
</tr>
<tr>
<td>IAEA-CN-155-009KS</td>
<td>Needs in research and development supporting nuclear power plant life management</td>
<td>M. Bieth</td>
<td>16</td>
</tr>
<tr>
<td>IAEA-CN-155-010KS</td>
<td>Securing the stability of ageing nuclear power plants – improving inspection system and ageing management and planning of technological strategy for research basis</td>
<td>N. Sekimura</td>
<td>18</td>
</tr>
<tr>
<td>IAEA-CN-155-001</td>
<td>European research network aiming at harmonized nuclear plant life prediction procedures</td>
<td>R. Rintamaa</td>
<td>21</td>
</tr>
<tr>
<td>IAEA-CN-155-002</td>
<td>Recent PLiM advances for current operation and long life</td>
<td>R.F. Dam</td>
<td>23</td>
</tr>
<tr>
<td>IAEA-CN-155-003</td>
<td>Key elements of long-term operation of WWER-440/213 units at Paks NPP</td>
<td>T. Katona</td>
<td>25</td>
</tr>
<tr>
<td>IAEA-CN-155-004</td>
<td>NPP long term operation in Spain-first application for license renewal</td>
<td>L. Francia</td>
<td>27</td>
</tr>
<tr>
<td>IAEA-CN-155-005</td>
<td>Implementation of the rules for the continued operation in Korea</td>
<td>J.J. Kwon</td>
<td>28</td>
</tr>
<tr>
<td>Reference</td>
<td>Title</td>
<td>Author</td>
<td>Page</td>
</tr>
<tr>
<td>-----------</td>
<td>----------------------------------------------------------------------</td>
<td>-------------------</td>
<td>------</td>
</tr>
<tr>
<td>IAEA-CN-155-006</td>
<td>Nuclear power plant life management: strategy for long term operation of the Beznau NPP unit 1 and 2</td>
<td>H. Rust</td>
<td>30</td>
</tr>
<tr>
<td>IAEA-CN-155-007</td>
<td>Plans for long term operation in Angra NPP</td>
<td>P.A. Costa Filho</td>
<td>32</td>
</tr>
<tr>
<td>IAEA-CN-155-008</td>
<td>Ageing Management of EDF NPP: From the design phase up to end of life</td>
<td>C. Faidy</td>
<td>35</td>
</tr>
<tr>
<td>IAEA-CN-155-009</td>
<td>Maintenance measures related to plant life management taken by Japanese PWRs</td>
<td>T. Kamada</td>
<td>36</td>
</tr>
<tr>
<td>IAEA-CN-155-010</td>
<td>The role of license renewal in PLiM for US nuclear power plants</td>
<td>G. Young</td>
<td>38</td>
</tr>
<tr>
<td>IAEA-CN-155-011</td>
<td>The experience of service life prolongation of NPP units of the first generation</td>
<td>M. Bakirov</td>
<td>40</td>
</tr>
<tr>
<td>IAEA-CN-155-012</td>
<td>Embalse NGS PLiM overview</td>
<td>G. Diaz</td>
<td>41</td>
</tr>
<tr>
<td>IAEA-CN-155-013</td>
<td>Ageing management at the NPPs of EnBW in Germany</td>
<td>B. Bletz</td>
<td>44</td>
</tr>
<tr>
<td>IAEA-CN-155-014</td>
<td>Ageing management and long term operation of NPP Borssele</td>
<td>A. de Jong</td>
<td>47</td>
</tr>
<tr>
<td>IAEA-CN-155-015</td>
<td>Materials aging management programs at nuclear power plants in the United States</td>
<td>T. Griesbach</td>
<td>48</td>
</tr>
<tr>
<td>IAEA-CN-155-016</td>
<td>Aging management in the Slovak Republic NPP’s</td>
<td>L’ Kupčaa</td>
<td>49</td>
</tr>
<tr>
<td>IAEA-CN-155-017</td>
<td>Applications of lifetime management for mechanical SSC in NPPs</td>
<td>E. Roos</td>
<td>51</td>
</tr>
<tr>
<td>IAEA-CN-155-018</td>
<td>Review and development of aging management programs of the main components at Paks NPP</td>
<td>P. Pálfi</td>
<td>53</td>
</tr>
<tr>
<td>IAEA-CN-155-019</td>
<td>Surveillance over the ageing effects and ability for the long term operation at the Krsko NPP</td>
<td>S. Šavli</td>
<td>54</td>
</tr>
<tr>
<td>IAEA-CN-155-020</td>
<td>Plant life management models with special emphasis to the integration of safety with non-safety related programs</td>
<td>P. Contri</td>
<td>56</td>
</tr>
<tr>
<td>IAEA-CN-155-021</td>
<td>Plant life management experience at Tarapur Atomic Power Station</td>
<td>V. Thattrey</td>
<td>58</td>
</tr>
<tr>
<td>IAEA-CN-155-022</td>
<td>The synergies of PLiM, PLEX, and power uprates: lessons learned from recent BWR experience</td>
<td>C. Nichols</td>
<td>60</td>
</tr>
<tr>
<td>IAEA-CN-155-023</td>
<td>Ageing management program to reactor pressure vessel internals components in a BWR nuclear power plant</td>
<td>C.R. Arganis</td>
<td>61</td>
</tr>
<tr>
<td>IAEA-CN-155-024</td>
<td>Aging management for TEPCO’s BWR reactor internals</td>
<td>N. Yamashita</td>
<td>65</td>
</tr>
<tr>
<td>IAEA-CN-155-025</td>
<td>Ageing management, in service inspection and exceptional maintenance</td>
<td>D. Dallery</td>
<td>68</td>
</tr>
<tr>
<td>IAEA-CN-155-026</td>
<td>In-service inspection of critical components as a key tool for the life management programmes</td>
<td>R. Martínez-Oña</td>
<td>70</td>
</tr>
<tr>
<td>IAEA-CN-155-027</td>
<td>Establishing a new ISI for PAKS NPP</td>
<td>P. Trampus</td>
<td>71</td>
</tr>
<tr>
<td>IAEA-CN-155-028</td>
<td>Onsite inspection experience of electric equipment in license process of the continued operation of Kori Unit One</td>
<td>O.P. Zhu</td>
<td>72</td>
</tr>
<tr>
<td>IAEA-CN-155-029</td>
<td>Network on use of PSA for evaluation of aging effects to the safety of energy facilities activities and results</td>
<td>A. Rodionov</td>
<td>74</td>
</tr>
<tr>
<td>IAEA-CN-155-030</td>
<td>The proposal evaluation approach of the risk informed-ISI and the result of trial evaluation</td>
<td>T. Azuma</td>
<td>77</td>
</tr>
<tr>
<td>IAEA-CN-155-031</td>
<td>Application of dynamic system reliability methods for incorporation of age-dependent reliability parameters and data into the PSA</td>
<td>G. Petkov</td>
<td>79</td>
</tr>
<tr>
<td>IAEA-CN-155-032</td>
<td>Analysis of the replacement need for the containment anchoring bolts of the Loviisa NPP estimated by the strength assessment</td>
<td>P. Varpasuo</td>
<td>82</td>
</tr>
<tr>
<td>IAEA-CN-155-033</td>
<td>SSC design modification, modernization, refurbishment and replacement: as a part of life extension of KANUPP</td>
<td>W. Butt</td>
<td>84</td>
</tr>
<tr>
<td>IAEA-CN-155-034</td>
<td>Material degradation management of the reactor coolant system at the Point Lepreau Generating Station</td>
<td>J. Slade</td>
<td>85</td>
</tr>
<tr>
<td>IAEA-CN-155-035</td>
<td>Considerations related to CANDU 6 lifetime management</td>
<td>M. Cojan</td>
<td>87</td>
</tr>
<tr>
<td>IAEA-CN-155-036</td>
<td>Life assessment experience for continued operation of a CANDU NPP in Korea</td>
<td>I.S. Jeong</td>
<td>90</td>
</tr>
<tr>
<td>IAEA-CN-155-037</td>
<td>Prevention of SCC occurring in an expansion transition region of steam generator tubing by Ni-plating in PWRs</td>
<td>J.S. Kim</td>
<td>92</td>
</tr>
<tr>
<td>IAEA-CN-155-038</td>
<td>Development of damage evaluation method considering radiation induced stress relaxation</td>
<td>Y. Kaji</td>
<td>94</td>
</tr>
<tr>
<td>IAEA-CN-155-039</td>
<td>Management of stress corrosion cracking in pressurized water reactors</td>
<td>M. Herrera</td>
<td>95</td>
</tr>
<tr>
<td>IAEA-CN-155-040</td>
<td>Failure analysis on primary water stress corrosion cracking of alloy 600 plugs for steam generator tube at a Korean NPP</td>
<td>M.H. Song</td>
<td>96</td>
</tr>
<tr>
<td>IAEA-CN-155-041</td>
<td>Periodic remaining life evaluation program of PWR presurizer surge line accounting for thermal stratification effect</td>
<td>B. Liang</td>
<td>97</td>
</tr>
<tr>
<td>IAEA-CN-155-042</td>
<td>Ageing management of the secondary circuit and related components of Embalse NPP</td>
<td>R. Saucedo</td>
<td>99</td>
</tr>
<tr>
<td>IAEA-CN-155-043</td>
<td>Structural integrity evaluation of cast austenitic stainless steel reactor Coolant piping for continued operation of nuclear power plants</td>
<td>P.E. Juhn</td>
<td>101</td>
</tr>
<tr>
<td>IAEA-CN-155-044</td>
<td>Progress of steam generator ageing management of Chinese NPPs</td>
<td>L. Hongyun</td>
<td>103</td>
</tr>
<tr>
<td>IAEA-CN-155-045</td>
<td>Development of cable ageing management program and equipment qualification improvement for Laguna Verde Nuclear Power Plant</td>
<td>R. M. Vázquez-Cervantes</td>
<td>104</td>
</tr>
<tr>
<td>IAEA-CN-155-046</td>
<td>Experimental relationship of break-elongation and indent data for ageing degradation of CSP and CR cable jacket</td>
<td>J.S. Kim</td>
<td>106</td>
</tr>
<tr>
<td>IAEA-CN-155-047</td>
<td>Improving regulatory practices through the OECD-NEA stress corrosion cracking and cable ageing project (SCAP)</td>
<td>A. Yamamoto</td>
<td>109</td>
</tr>
<tr>
<td>IAEA-CN-155-048</td>
<td>The collection of information, data and materials samples from concrete structures on nuclear facilities under decommissioning for ageing and degradation evaluation</td>
<td>L.M. Smith</td>
<td>111</td>
</tr>
<tr>
<td>IAEA-CN-155-049</td>
<td>Results of German investigations on damage due to material ageing and corrosion</td>
<td>R. Gersinska</td>
<td>112</td>
</tr>
<tr>
<td>IAEA-CN-155-050</td>
<td>Mitigation of degradation of high energy secondary cycle piping due to flow accelerated corrosion and life management of high energy piping in India nuclear power plants</td>
<td>T.M. Moolayil</td>
<td>114</td>
</tr>
<tr>
<td>IAEA-CN-155-051</td>
<td>Failure analysis on erosive wear of RCW heat exchanger titanium tubes</td>
<td>Y.F. Wang</td>
<td>116</td>
</tr>
<tr>
<td>IAEA-CN-155-052</td>
<td>New approaches for flow-accelerated corrosion estimation</td>
<td>M. Bakirov</td>
<td>118</td>
</tr>
<tr>
<td>IAEA-CN-155-053</td>
<td>Fatigue monitoring for demonstrating fatigue design basis compliance</td>
<td>D. Gerber</td>
<td>119</td>
</tr>
<tr>
<td>IAEA-CN-155-054</td>
<td>Corrosion monitoring system in the Slovak Republic nuclear power plants</td>
<td>M. Brezina</td>
<td>120</td>
</tr>
<tr>
<td>IAEA-CN-155-055</td>
<td>Vibration assessment method and engineering applications to small bore piping in nuclear power plant</td>
<td>F. Xue</td>
<td>122</td>
</tr>
<tr>
<td>IAEA-CN-155-056</td>
<td>WWER pressure vessel life and ageing management for NPP long term operation in Russia</td>
<td>V. Vasiliev</td>
<td>123</td>
</tr>
<tr>
<td>IAEA-CN-155-057</td>
<td>IAEA coordinated research projects on irradiated reactor pressure vessel structural integrity</td>
<td>W. Server</td>
<td>124</td>
</tr>
<tr>
<td>IAEA-CN-155-058</td>
<td>Aspects of operational life management of nuclear power plants</td>
<td>A. Ballesteros</td>
<td>126</td>
</tr>
<tr>
<td>IAEA-CN-155-059</td>
<td>Radiation embrittlement and neutron dosimetry aspects in WWER-440 reactor pressure vessels life time extension</td>
<td>D. Erak</td>
<td>127</td>
</tr>
<tr>
<td>IAEA-CN-155-060</td>
<td>Methodology research on prediction for operating lifetime of PWR RPV</td>
<td>Y. He</td>
<td>128</td>
</tr>
<tr>
<td>IAEA-CN-155-061</td>
<td>Conformity between LR0 mock-ups and VVERs NPP RPV attenuation</td>
<td>D. Kirilova</td>
<td>130</td>
</tr>
<tr>
<td>IAEA-CN-155-062</td>
<td>Neutron activation of reactor components during operation lifetime of a NPP</td>
<td>G. Pretzsch</td>
<td>132</td>
</tr>
<tr>
<td>IAEA-CN-155-063</td>
<td>RTPTPS re-evaluation of Kori-1 RPV beltline weld by master curve tests</td>
<td>B.S. Lee</td>
<td>135</td>
</tr>
<tr>
<td>IAEA-CN-155-064</td>
<td>Strategies for sustaining current nuclear assets</td>
<td>K. Huffman</td>
<td>137</td>
</tr>
<tr>
<td>IAEA-CN-155-065</td>
<td>Research in automation, risk analysis, control rooms, and organizational factors; applications to plant life management</td>
<td>B. Wahlström</td>
<td>139</td>
</tr>
<tr>
<td>IAEA-CN-155-066</td>
<td>New plant life management for Dukovany NPPs.</td>
<td>P. Kadecka</td>
<td>142</td>
</tr>
<tr>
<td>IAEA-CN-155-067</td>
<td>Canadian approach on regulatory Issues regarding ageing management, long term operation and plant life management</td>
<td>T. Viglasky</td>
<td>143</td>
</tr>
<tr>
<td>IAEA-CN-155-068</td>
<td>Regulatory framework for continued operation in Korea</td>
<td>Y.W. Park</td>
<td>144</td>
</tr>
<tr>
<td>IAEA-CN-155-069</td>
<td>Russian regulatory approach to extension of nuclear power plant service life</td>
<td>E. Vasileva</td>
<td>145</td>
</tr>
<tr>
<td>IAEA-CN-155-070</td>
<td>Regulatory approach to the long term operations of Czech nuclear power plants</td>
<td>M. Šváb</td>
<td>147</td>
</tr>
<tr>
<td>IAEA-CN-155-071</td>
<td>Activities of OECD/NEA in the regulatory aspects of plant life management</td>
<td>A. Blahoiuanu</td>
<td>152</td>
</tr>
<tr>
<td>IAEA-CN-155-072</td>
<td>Lessons learned from the OECD study on the ‘impacts of nuclear power plant life management on long term operation’</td>
<td>P. Kovács</td>
<td>154</td>
</tr>
<tr>
<td>IAEA-CN-155-073</td>
<td>Instrumentation and control (I&amp;C) modernization at KANUPP</td>
<td>Z. Rabbani</td>
<td>157</td>
</tr>
<tr>
<td>IAEA-CN-155-074</td>
<td>Steps towards a large-scale I&amp;C modernization at Paks NPP to serve the planned plant service time extension</td>
<td>T. Túri</td>
<td>158</td>
</tr>
<tr>
<td>IAEA-CN-155-075</td>
<td>Managing I &amp; C obsolescence for GE BWR nuclear plant life extension</td>
<td>L.L. Chi</td>
<td>160</td>
</tr>
<tr>
<td>IAEA-CN-155-076</td>
<td>Case histories and lessons learned from the design, development, planning and implementation of new I&amp;C systems, including effective integration with existing systems and processes</td>
<td>D. Greene</td>
<td>163</td>
</tr>
<tr>
<td>IAEA-CN-155-077</td>
<td>Assessment of crack-like flaw in ex-service Monel 400 steam generator tube removed from Pickering Unit 4 steam generator 12</td>
<td>X. Duan</td>
<td>167</td>
</tr>
<tr>
<td>IAEA-CN-155-078</td>
<td>Nuclear power plant life management: Materials and components, research, human resource, radwaste, and regulatory aspects</td>
<td>P. Tipping</td>
<td>169</td>
</tr>
<tr>
<td>IAEA-CN-155-001P</td>
<td>Life Assessment of primary heat transport system feeders at Embalse NGS</td>
<td>C. Schiersmann</td>
<td>171</td>
</tr>
<tr>
<td>-----------------</td>
<td>-------------------------------------------------</td>
<td>---------------</td>
<td>-----</td>
</tr>
<tr>
<td>IAEA-CN-155-002P</td>
<td>Refurbishment of main condenser circulating water piping of Embalse N.P.P.</td>
<td>D. Quinteros</td>
<td>172</td>
</tr>
<tr>
<td>IAEA-CN-155-003P</td>
<td>Inverter system replacement at Embalse N.P.P.</td>
<td>P. Acevedo</td>
<td>173</td>
</tr>
<tr>
<td>IAEA-CN-155-004P</td>
<td>Replacement of flux detectors at Embalse N.P.P.</td>
<td>H. Mangold</td>
<td>174</td>
</tr>
<tr>
<td>IAEA-CN-005P</td>
<td>Condition assessment of main transformers and replacement criteria.</td>
<td>R. Bonelli</td>
<td>175</td>
</tr>
<tr>
<td>IAEA-CN-155-006P</td>
<td>Modernization of instrumentation in water demineralization plant at Embalse N.P.P.</td>
<td>L. Perez</td>
<td>177</td>
</tr>
<tr>
<td>IAEA-CN-155-007P</td>
<td>Helicoidal spacers in the moderator heat exchanger at Embalse N.P.P.</td>
<td>L. Alvarez</td>
<td>178</td>
</tr>
<tr>
<td>IAEA-CN-155-008P</td>
<td>Management activities for long term operations of Atucha II nuclear power plant</td>
<td>S. Brunatti</td>
<td>179</td>
</tr>
<tr>
<td>IAEA-CN-155-009P</td>
<td>Pipe whip restraints-protection for safety related equipment of WWER nuclear power plants</td>
<td>Z. Plocek</td>
<td>181</td>
</tr>
<tr>
<td>IAEA-CN-155-010P</td>
<td>Design review of the VVER-440 NPP main components of the base of ASME BPVC</td>
<td>S. Dancsó</td>
<td>183</td>
</tr>
<tr>
<td>IAEA-CN-155-012P</td>
<td>Preliminary comprehensions about RPV and RVI ageing management and assessment technology</td>
<td>Y. Yang</td>
<td>184</td>
</tr>
<tr>
<td>IAEA-CN-155-013P</td>
<td>Rapid determination of histories of SIF distributions along 3-D crack fronts of RPV subjected to PTS by universal weight function and finite variation method</td>
<td>Y.L. Lu</td>
<td>185</td>
</tr>
<tr>
<td>IAEA-CN-155-015P</td>
<td>Role of research in materials development, mitigation strategies and non-destructive evaluation for plant life management (PLiM) in the Indian nuclear power programme</td>
<td>B. Raj</td>
<td>188</td>
</tr>
<tr>
<td>IAEA-CN-155-018P</td>
<td>The use of rotor diagnosis for the analysis of high vibration experience at turbine-generator system in nuclear power plants</td>
<td>G. Shin</td>
<td>191</td>
</tr>
<tr>
<td>Document Code</td>
<td>Title</td>
<td>Author</td>
<td>Page</td>
</tr>
<tr>
<td>---------------</td>
<td>-------------------------------------------------------------------------------------------------</td>
<td>---------------------</td>
<td>------</td>
</tr>
<tr>
<td>IAEA-CN-155-019P</td>
<td>The specific surveillance program in the plant life management of Laguna Verde NPP</td>
<td>M. Gachuz</td>
<td>193</td>
</tr>
<tr>
<td>IAEA-CN-155-020P</td>
<td>Continuing systems/components reliability process</td>
<td>C. Muscaloiu</td>
<td>196</td>
</tr>
<tr>
<td>IAEA-CN-155-021P</td>
<td>Evolution of WWER-1000 RPV materials nano-structure under irradiation and post irradiation annealing</td>
<td>A. Chernobaeva</td>
<td>197</td>
</tr>
<tr>
<td>IAEA-CN-155-022P</td>
<td>Application and ISO principles for regulatory practices and safety culture of nuclear installations being built to Russian designs</td>
<td>V. Kozlov</td>
<td>200</td>
</tr>
<tr>
<td>IAEA-CN-155-023P</td>
<td>Identification of optimum parameters of annealing for the VVER-1000 RPV materials with high level of NI</td>
<td>D. Zhurko</td>
<td>202</td>
</tr>
<tr>
<td>IAEA-CN-155-024P</td>
<td>Prevention of resonances between flow parameters oscillations and structure vibrations is the principal reserve of nuclear power plant life management</td>
<td>K. Proskuryakov</td>
<td>203</td>
</tr>
<tr>
<td>IAEA-CN-155-027P</td>
<td>Experience of NPP I&amp;C management of ageing</td>
<td>M. Yastrebenetsky</td>
<td>20</td>
</tr>
<tr>
<td>IAEA-CN-155-028P</td>
<td>Replacement of heavy water supply controllers in fuel handling system</td>
<td>E. Binetti</td>
<td>207</td>
</tr>
<tr>
<td>IAEA-CN-155-029P</td>
<td>Indian regulatory requirements with respect to plant life management for renewal of operation authorisation</td>
<td>Y.K. Shah</td>
<td>208</td>
</tr>
<tr>
<td>IAEA-CN-155-030P</td>
<td>Failure probability assessment of a PWR primary system piping subcomponents under different loading conditions</td>
<td>D. Datta</td>
<td>210</td>
</tr>
<tr>
<td>IAEA-CN-155-031P</td>
<td>Methods of evaluation of operational experience for regulating decisions support</td>
<td>M. Lankin</td>
<td>212</td>
</tr>
<tr>
<td>IAEA-CN-155-032P</td>
<td>Comparative analysis of irradiation conditions of surveillance specimens and RPV for lifetime extension of ROVNO-1 and ROVNO-2</td>
<td>V. Kochkin</td>
<td>214</td>
</tr>
<tr>
<td>IAEA-CN-155-033P</td>
<td>Application of the &quot;leak before break&quot; concept on NPP units of the first generation with the WWER-440 reactors. Improvements of leak detection systems considering new elaborated approaches</td>
<td>M. Bakirov</td>
<td>215</td>
</tr>
<tr>
<td>Code</td>
<td>Title</td>
<td>Author</td>
<td>Page</td>
</tr>
<tr>
<td>--------------</td>
<td>-------------------------------------------------------------------------------------------</td>
<td>-------------------</td>
<td>------</td>
</tr>
<tr>
<td>IAEA-CN-155-034P</td>
<td>Problems of development of a prediction technique of a residual lifetime of heat exchanging tubes of WWER NPPs</td>
<td>M. Bakirov</td>
<td>216</td>
</tr>
<tr>
<td>IAEA-CN-155-035P</td>
<td>Effect of the RPV cladding properties on the WWER-440 reactors lifetime</td>
<td>F. Gillemot</td>
<td>217</td>
</tr>
<tr>
<td>IAEA-CN-155-036P</td>
<td>Development of aging monitor for operating nuclear power plants</td>
<td>H. Kim</td>
<td>218</td>
</tr>
<tr>
<td>IAEA-CN-155-037P</td>
<td>DACAAM-NET - An intranet application for management of ageing related data</td>
<td>K. Baumann-ne Tanits</td>
<td>219</td>
</tr>
<tr>
<td>IAEA-CN-155-038P</td>
<td>Application of general methodology PLiM in the life assessment of emergency core cooling system heat exchanger at Embalse ENGS</td>
<td>A. Bergara</td>
<td>220</td>
</tr>
<tr>
<td>IAEA-CN-155-039P</td>
<td>The lessons and findings from the fuel channel lifetime management study</td>
<td>K.S. Lee</td>
<td>222</td>
</tr>
<tr>
<td>IAEA-CN-155-040P</td>
<td>Ageing assessment of RBMK-1500 fuel channel in case of delayed hydride cracking</td>
<td>G. Dundulis</td>
<td>224</td>
</tr>
<tr>
<td>IAEA-CN-155-041P</td>
<td>Piping evaluation under the degradation mechanism caused by control valve frequent action inducing vibration</td>
<td>G. Li</td>
<td>226</td>
</tr>
<tr>
<td>IAEA-CN-155-042P</td>
<td>From the fire alarm system to improve fire safety of nuclear power plant</td>
<td>C. Hang</td>
<td>228</td>
</tr>
<tr>
<td>IAEA-CN-155-043P</td>
<td>Ageing management review for reactor internals of PWR nuclear power plant</td>
<td>M. Zhang</td>
<td>230</td>
</tr>
<tr>
<td>IAEA-CN-155-044P</td>
<td>Ageing research for upgrades using digital I&amp;C systems of nuclear power plant</td>
<td>P. Chao</td>
<td>232</td>
</tr>
<tr>
<td>IAEA-CN-155-045P</td>
<td>Vibration fatigue analysis for mainstream pipelines on a NPP</td>
<td>X. Liang</td>
<td>234</td>
</tr>
<tr>
<td>IAEA-CN-155-046P</td>
<td>Ageing management of carbon steel pipings in 300MWe PWR secondary systems</td>
<td>X. Shi</td>
<td>237</td>
</tr>
<tr>
<td>IAEA-CN-155-047P</td>
<td>Aging degradation problems and some countermeasure considerations of PWR RPV and RVI</td>
<td>X. Xu</td>
<td>240</td>
</tr>
<tr>
<td>IAEA-CN-155-048P</td>
<td>Neutron monitoring system and rod control system upgrades for plant life extension</td>
<td>Z. Wang</td>
<td>242</td>
</tr>
<tr>
<td>Code</td>
<td>Title</td>
<td>Author</td>
<td>Page</td>
</tr>
<tr>
<td>------------</td>
<td>----------------------------------------------------------------------</td>
<td>-----------------------</td>
<td>------</td>
</tr>
<tr>
<td>IAEA-CN-155-050P</td>
<td>Regulatory issues in spent nuclear fuel management in Belarus</td>
<td>V.M. Paliukhovich</td>
<td>244</td>
</tr>
<tr>
<td>IAEA-CN-155-051P</td>
<td>Regulative aspects and decisions at licensing contemporary I &amp; C changes of NPP with WWER reactors in the context of PLiM</td>
<td>V. Krastanov</td>
<td>246</td>
</tr>
<tr>
<td>IAEA-CN-155-052P</td>
<td>Appraisal and countermeasures of pressure pipelines of CI and BOP</td>
<td>K. Li-Zhong</td>
<td>251</td>
</tr>
<tr>
<td>IAEA-CN-155-053P</td>
<td>Design of earthquake resistance enhancements of the Dukovany nuclear power plant building structures</td>
<td>Z. Plocek</td>
<td>253</td>
</tr>
<tr>
<td>IAEA-CN-155-055P</td>
<td>Regulatory control of ageing research reactor in Indonesia</td>
<td>Z. Ilyas</td>
<td>256</td>
</tr>
<tr>
<td>IAEA-CN-155-056P</td>
<td>Analysis and evaluation on hydrogen blistering of titanium tubes on tube sheet of RCW heat exchanger</td>
<td>Y.F. Wang</td>
<td>257</td>
</tr>
<tr>
<td>IAEA-CN-155-057P</td>
<td>Preliminary study on strategy of plant life management for the first nuclear power plant in Indonesia</td>
<td>S. Nitiswati</td>
<td>259</td>
</tr>
<tr>
<td>IAEA-CN-155-058P</td>
<td>Activity in Kazakhstan related to NPP life management</td>
<td>O. Romanenko</td>
<td>262</td>
</tr>
<tr>
<td>IAEA-CN-155-060P</td>
<td>Grade management of PWR steam generator potential ageing mechanism based on risk</td>
<td>G. Chun</td>
<td>264</td>
</tr>
<tr>
<td>IAEA-CN-155-061P</td>
<td>General method of thermal stress calculation on pressurized parts in nuclear power plant</td>
<td>L.P. Pang</td>
<td>266</td>
</tr>
<tr>
<td>IAEA-CN-155-064P</td>
<td>Maintenance and life assessment of steam generators at Embalse Nuclear Station</td>
<td>P. Luna</td>
<td>268</td>
</tr>
<tr>
<td>IAEA-CN-155-066P</td>
<td>Development of regulatory guidelines for the continued operation of CANDU reactor in Korea</td>
<td>Y.H. Choi</td>
<td>272</td>
</tr>
<tr>
<td>IAEA-CN-155-068P</td>
<td>Role of organizational leadership in plant life management</td>
<td>R.K. Mohindra</td>
<td>274</td>
</tr>
<tr>
<td>IAEA-CN-155-069P</td>
<td>PNRA Experience with licensing of KANUPP beyond design life</td>
<td>M. Tanweer Khan</td>
<td>276</td>
</tr>
<tr>
<td>IAEA-CN-155-070P</td>
<td>Ageing related programmes in Angra NPP Units 1 and 2</td>
<td>L. Dias Batista Ferrari</td>
<td>279</td>
</tr>
<tr>
<td>IAEA-CN-155-071P</td>
<td>The large projects at Kozloduy NPP - With focus on long time operation and ageing management</td>
<td>V. Popov</td>
<td>282</td>
</tr>
<tr>
<td>IAEA-CN-155-072P</td>
<td>Perspectives of plant life management with trans-boundary effects</td>
<td>G.H. Weimann</td>
<td>286</td>
</tr>
<tr>
<td>IAEA-CN-155-073P</td>
<td>PWR-LTO and Ageing: A study of available public material</td>
<td>K.A. Arlamovsky</td>
<td>287</td>
</tr>
<tr>
<td>IAEA-CN-155-074P</td>
<td>Fracture mechanics–based life management of structural materials operating at elevated temperatures</td>
<td>K. Nikbin</td>
<td>289</td>
</tr>
</tbody>
</table>
IAEA ACTIVITIES ON PLANT LIFE MANAGEMENT FOR LONG TERM OPERATIONS

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For the past couple of decades there has been a change of emphasis in the world nuclear power from that of building new Nuclear Power Plants (NPP) to that of taking measures to optimize the life cycle of operational plants. National approaches in many countries showed an increase of interest in Plant Life Management (PLiM), both in terms of plant service life assurance and in optimizing the service or operational life of NPP.

The safety considerations of a NPP are paramount and those requirements have to be met to obtain and to extend/renew the operating license. To achieve the goal of the long term safe, economic and reliable operation of the plant, PLiM programme is essential. Some countries already have advanced PLiM programmes while others still have none. The PLiM objective is to identify all that factors and requirements for the overall plant life cycle. The optimization of these requirements would allow for the minimum period of the investment return and maximum of the revenue from the sell of the produced electricity.

Recognizing the importance of this issue and in response to the requests of the Member States the IAEA Division of Nuclear Power implements the Sub-programme on "Engineering and Management Support for Competitive Nuclear Power". Three projects within this sub-programme deal with different aspects of the NPP life cycle management with the aim to increase the capabilities of interested Member States in implementing and maintenance of the competitive and sustainable nuclear power.

Although all three projects contain certain issues of PLiM, there is one specific project on guidance on engineering and management practices for optimization of NPP service life. This particular project deals with different specific issues of PLiM including aspects of ageing phenomena and their monitoring, issues of control and instrumentation, maintenance and operation issues, economic evaluation of PLiM including guidance on its earlier shut down and decommissioning. The paper describes in detail the full scope IAEA activities on different issues of PLiM and some of its achievements in this field during the nearest past as well as plans for the future.
PROACTIVE PLANT LIFE MANAGEMENT - ONE OF TVO'S KEY SUCCESS FACTORS

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Teollisuuden Voima Oy (TVO) is a Finnish utility owned by several Finnish industrial and public companies. All electricity produced by TVO is supplied at cost to the owner companies. The company’s investment-intensive operations are based on consistent planning, which takes the long-term span of the operations into account. TVO aims at preventing any incidents from occurring that might affect safety or the availability of electricity. This is achieved by keeping the units in good condition and up-to-date and by ensuring personnel’s expertise, a good working atmosphere and well-being at work.

TVO commissioned two identical BWRs of 660 MWe, Olkiluoto 1 (OL1) and Olkiluoto 2 (OL2), in 1979 and 1982, respectively. The third unit, a PWR unit Olkiluoto 3, is under construction and scheduled to start commercial operation at the turn of year 2010/2011.

As a part of the long term planning, TVO started the modernization of the OL1 and OL2 units in the mid-1990’s after an about 15 years’ operating period. The main goals of the extensive programme were: enhanced safety, increased production, extended lifetime and enlarged staff expertise. Following the modernization, the net output of both units was increased from 660 MW to 860 MW by uprating the thermal power and by increasing the thermal efficiency. This year, the company has decided to continue the programme by modernizing the turbine plant in 2010-2011, which is expected to increase the output further by some 25 MW per unit.

The average total capacity factor of the OL1 and OL2 units during (1988-1998) and after the modernization program (1999-2006) have been 93.5% and 96%, respectively. TVO aims at 95-96% long term capacity factors. The safety related and operational performance of the plant units after the modernization has been good. The modernization programme is expected to contribute in ensuring competitiveness and success in the future. It also forms a firm foundation for the coming programmes to keep the plant design modern.

Pertti Sakari Simola graduated (M.Sc. Power Engineering) from the Lappeenranta University of Technology, Finland, in 1973. He has held management positions since 1973 e.g. in the Finnish forest industry (e.g. Kaukas Oy, Oy Metsä-Ab, Kymmen Corporation, UPM-Kymmene Corp.) – both in Finland and abroad. Before joining Teollisuuden Voima Oy, Mr Simola was employed by UPM-Kymmene Corp. as Vice President, Energy, for seven years.

In 2004, P. Simola was appointed Teollisuuden Voima Oy's President and CEO, which position he received on 1 September 2004. P. Simola holds several board memberships in Finnish energy companies.
Introduction

As one of the main economic communities in the world and the most populous country, China is developing in a considerable speed. The needs for energy consumption as well as for environmental protection are very pressed. At present, nuclear power is the most suitable choice to compromise the above two needs. In recent years, China speeds up the development of nuclear power with a target to make its nuclear power capacity from around 9,000 MWe, about 2% of the whole electric power, in 2007 to 40,000 MWe, about 4% of the whole electric power, in 2020. With 11 multi-type units in operation at present and more than 30 units under construction, most of them being 3G units, in the next 10 years, China will definitely act as an important role in world nuclear power development in early of 21st century. How can we keep this proactive and progressive tendency and fulfill the target of clean electric power to satisfy the needs of more than 1.3 billion population? Ensuring safety while showing economic superiority of operational NPPs is one of the most important aspects to maintain the sustainable development. Plant life management is one of the important backbones to guarantee safe of operational NPPs. It will provide support to some important aspects for NPP operation, including nuclear safety indication system, experience feedback system to design, PSA system, database system, efficiency and power uprating system, spare parts strategy system, etc.

This paper briefly introduces the current status and prospect of Chinese nuclear power plant development, emphasizing interdependent and strategic relations between maintaining safe and economic benefit of operational NPPs by effective PLiM and implementing ambitious plan for construction new NPPs. Examples on PLiM activities in China in various organizations are listed, illustrating much progress being attained in past 10 years. Some points for future development are also presented in the paper.

Main aspects of PLiM for Chinese NPPs

China started nuclear power operation in Dec. 15, 1991, when Qinshan-I, a 300 MWe PWR unit independently developed by Chinese own efforts, got into grid. Since then, many efforts by various organizations including nuclear safety regulatory body, NPPs as well as designers and technical supporters to NPPs, have been made in maintaining safe and availability of operational NPPs. Life management is one of the aspects in these efforts.

Nuclear safety regulatory system

Chinese nuclear safety regulatory system in ageing management basically follows IAEA requirements and guidelines. The main requirements are stipulated in two documents: Safety Requirements for Nuclear Power Plant Design (HAF 102) and Safety Requirements for Nuclear Power Plant Operation (HAF 103), which are very similar to IAEA NS R-1 and NS R-2, respectively, Following the two requirements, there are a series of guidelines (HAD). For
example, it is required that for operational NPPs, Periodical Safety Review should be carried out every 10 years and HAD 103/11 provides a guidance for PSR which is similar to IAEA SG NS-G-2.10.

**Activities by utilities**

The ageing management activities by NPP utilities include periodical safety review (PSR), corrective actions focused on weaknesses identified by PSR, development of overall ageing management program (AMP), component based AMPs and ageing management databases, etc. At present, two NPPs and 3 units, Qinshan-I and Daya Bay-I, -II, has experienced the first PSR. Some plant are preparing for their first PSR. After PSR, the NPPs are doing corrective actions focused on weaknesses identified. For example, Qinshan-I is establishing overall AMP as well as some component based AMP.

**Activities by designers and technical supporters**

Typical activities of designers and technical supporters focus on making guidelines and standards as guidance for ageing management practice, establishment of assessment methodologies, investigating ageing degradation mechanisms of materials, developing inspection and monitoring methodologies and systems. For example, as a pioneering NPP designer for Chinese first, Qinshan-I, Shanghai Nuclear Engineering Research and Design Institute (SNERDI) carried out systematic R&Ds on ageing management and lifetime assessment for NPPs, as well as many technical services to NPPs.

**Role of international cooperation**

To effectively carry out PLiM and ensure safe while showing economic advantage for operational NPPs, it is important to establish international collaborative platform for experience feedback and information sharing. In recent years, IAEA provides many technical supports in life management to Chinese counterparts through various projects, such as TC and EPB. These activities widely spread conception of IAEA ageing management system and promote its practice in Chinese NPPs.

**Some points in the future**

Some points which should be focused on in the future towards Chinese PLiM system are listed as follows:
- Establishment and improvement of nuclear safety regulatory system;
- Construction of NPP life extension review system;
- Design requirements considering and facilitating PLiM
- Application of risk-informed technique
- AMP for critical components in secondary loop.

Dr. Zheng Mingguang graduated (Ph.D. Nuclear Engineering and Science) from Shanghai Jiaotong University. Dr. Zheng has been employed with Shanghai Nuclear Engineering Research and Design Institute (SNERDI) since 1982 and are working as vice president. His duties include the participation of Qinshan and Chashma 1 basic and detailed designs, SAR review and defense. He is responsible for the management of the detailed design of Hongyanhe NPP (4X1100MWe) in Liaoning Province of China and also for the operation supporting service of nuclear power plants such as strategic study, modification and back-fitting of system
and equipment, engineering and operation feedback for QNPC, TQNPC, and Chashma Nuclear Power Plant Unit I. His duties also include being an advisor for Ph. D. and Master Degree students on nuclear engineering and science.
PLANT LIFE MANAGEMENT OF A LARGE NUCLEAR FLEET

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Plant life management (PLiM) of the 58 EDF nuclear plants is in the core of an energy strategy which integrates durable development, supplying security, safety and competitiveness.

Three main stakes are linked to the EDF fleet PLiM:

- to assure permanently a high level of safety in operation, with a continuous increase of safety level,
- to increase the economic value of such an industrial tool, taking into account the advantage of a standardized fleet, and dealing with the potential vulnerability effect induced;
- to participate to the industrial solution required to permit a new built smoothing, in comparison with the high initial rythm, of nearly 50 Gwe, during the 80’s.

In this paper there are an analysis and descriptions of:

- the industrial stakes associated to lifetime management in particular questions which have to be investigated,
- the meaning, processus and content of the lifetime management approach within EDF:
  o 10 years safety reassessment, taking into account, on one hand, safety compliance demonstration and, on the other hand, safety reassessment, the aim of which is to upgrade safety level (seismic analysis, hazards, severe accidents,…). A cost/benefit methodology has been defined to rank the pertinence of plant modifications. The results concerning 3 loops plants thirty years decennial outage program are presented.
  o Ageing management principles, based on feed back experience of a large standardized fleet, including a dedicated process for obsolescence problems, and an anticipation approach (examples are given: alloys 600 areas, thermal fatigue, …)
  o Research and development support activities, in relationship with safety criteria, industrial decisions,
  o Skills management, internally for the company and those of industrial operators, subcontractors, in the field of design, manufacturing and maintenance :
    ▪ Evolution of job needs (engineering and operation),
    ▪ Medium and long term vision, in particular for sensible skills,
    ▪ Skills transfer because of high rate of retirement.

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engineering (7 years). He was safety engineer for the Super -Phenix project (fast breeder
reactor) for three years. He started his career in R&D activities, in the field of system
reliability and safety. He is graduated from Advanced Technics University Level College
and from Naval Engineering. He is an active member of French National Nuclear Society,
French Metallurgical Society and French Pressure Vessels Engineers Society.
REFURBISHMENTS & LIFE EXTENSIONS – LESSONS FROM THE PAST TO GET IT RIGHT IN THE FUTURE

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Over the last two decades the nuclear power industry worldwide has worked hard to improve performance and regain public trust and support for nuclear energy. Decisions we make today around life extensions and refurbishments will set the foundation for our performance over the next several decades. Sustaining safe, clean, predictable energy production performance in extended life operation of plants is a necessary condition to earn the privilege of nuclear energy production. Currently nuclear support is strong but remains fragile. We will always be one major event away from another period of decline.

So as growing demand for clean energy intensifies and business pressures drive us all, what can we draw from our experience to set the industry up for success? The first major refurbishment of a CANDU reactor took place in the 1980’s at the Pickering A Station in Ontario, Canada. Following a ‘G16’ pressure tube failure the reactors were re-tubed. The first Unit of four took 56 months to complete. Approximately three years later the fourth Unit was completed in 19 months. What enabled the dramatic improvement in schedule? Was the outcome what was expected? What lessons from history can we learn to shape practices today and set us up for success?

There are many factors that must be considered today including setting the right scope, encouraging the right behaviour and beginning with the end in sight. This presentation is intended to provoke thought and highlight some critical areas that heavily influence outcome. The experience from earlier campaigns is used to illustrate the opportunities and the risks.

John Froats is President & Chief Executive Officer of the CANDU Owners Group (COG). COG is a private, not for profit Company which is voluntarily funded by its Members. It provides Member Services in the areas of CANDU Nuclear Research and Development, Operating Experience, Joint Projects and Regulatory Affairs. John brings over 32 years of Nuclear Generation experience to his role in COG.

Prior to joining COG John held a number of leadership roles in Ontario Power Generation (OPG). OPG is an Ontario-based electricity generator whose principal business is the generation and sale of electricity in Ontario and to interconnected markets, while operating in a safe, open and environmentally responsible manner.
Introduction

Programmes to manage the ageing effects of Nuclear Power Plants (NPP) have been launched in different countries to demonstrate the long term integrity of nuclear power plants in particular for plant life extension purposes often based on published IAEA-recommendations how to deal with the physical ageing of safety relevant systems.

As a consequence of these international developments, the ageing management aspect was introduced to Germany, too, although plant life extension is not the key subject in the German nuclear industry, today. The situation in Germany concerning long term integrity of safety relevant NPP-systems and components is determined by safety regulations and codes containing requirements for continuous precaution measures according to the current “state of the art” starting already with the plant commissioning (demands for redundant protection issues and for surveillance measures).

These precaution measures are performed under different names and may be grouped in surveillance measures, maintenance and in-service inspection (ISI) activities. In addition to these plant specific measures, potential changes of the general “state of the art” are considered.

The German utilities understand that ageing issues are comprehensively covered by the entirety of these different precaution and maintenance measures. However, as different institutions are involved in the ageing management business, divergent interpretations exist how to handle technical contents and administrative issues. Thus, a harmonised understanding is required. Consequently, the preparation of a specific German KTA-rule 2301 “Ageing Management in Nuclear Power Plants” has been initiated.

Bearing in mind that the above mentioned measures managing the ageing of NPP-components already exist, the objective of this KTA-work is to provide a framing guide line around these various activities under the headline “Ageing Management”. To do so, the basic criteria to manage ageing effects in NPP have to be defined and the sound application of the related safeguarding measures has to be demonstrated.

* Status Report 2007
Basic NPP Ageing Management Issues

Basically, the “Management of Ageing” in NPP comprises conceptual, technological and physical ageing issues. Conceptual ageing aspects address potential changes in the design philosophy of NPP-systems. Technological ageing issues cover relevant changes in the state-of-the-art (e. g. in codes and standards). These long-term topics are mainly covered in periodical safety reviews (PSR). If necessary, relevant conceptual/technological aspects will be considered in the short term management of physical ageing. Thus, the managing of physical ageing in NPP is covered by

- the “Plant Life Management (PLIM)” of the entire nuclear power plant for safety and availability reasons primarily on the utility responsibility and
- the “Ageing Management (AM)” for the safety relevant systems, structures and components (SSCs) supervised by the responsible safety authority based on the issued licence of the plant.

As conceptual or even technological ageing issues usually lead to preventive plant safety/availability related precaution measures carried out by the utility voluntarily, these measures may be considered as part of the PLIM, even if they are safety related. Consequently, the same applies for the ageing management of physical degradation mechanisms. This leads to the fact that PLIM and AM of safety relevant systems/components are not strictly separated in practical NPP application. Very often utilities apply the same high level precaution for non safety relevant systems to ensure the plant availability as for safety significant SSCs. Hereby, one has to bear in mind that all power plant components are designed and fabricated with an appropriate quality level to fulfil the required tasks over the intended life time whether they are safety relevant or not.

Basic PLIM/AM-Concept

The safeguarding of quality requirements during plant operation lifetime is carried out by proactive and reactive measures depending on the graduated safety relevance (Group 1-3). The proactive measures cover the monitoring of root causes of potential operational degradation mechanisms (e.g. operational loadings, water chemistry). Hereby, the proactive approach, applied for Group 1-components, tries to avoid/minimise premature degradation effects. The reactive surveillance of consequences of potential operational degradation mechanisms deals with degradation effects after they have already occurred and been detected (e.g. by NDT, ISI, preventive maintenance). The component classification and the related PLIM/AM-activities are fixed in plant operation manuals and other documents approved by the responsible safety authority. The PLIM/AM-related precaution measures are surveillance measures, maintenance activities and ISI.

The PLIM-measures applied for SSCs during the plant operation to ensure the required quality usually correspond to the AM-measures. E.g. valves of the same quality are installed in safety and in availability relevant systems, which gives an extended data base about potential ageing effects.

Plant specific PLIM/AM-application

To compile all relevant ageing related information in one document, plant specific PLIM/AM-documentations have been prepared:
• “Basic Report” containing all general criteria plant specific component classification and the present status concerning the ageing management issues
• “Periodic Status Reports” (annual) documenting the “Delta”-information compared to the basic report.

Hereby, the plant specific “Basic Report” contains the following topics:
• Scope of evaluation component classification in Groups 1-3,
• Identification of relevant degradation mechanisms,
• Depiction of PLIM/AM-related measures,
• Assessment of effectiveness of PLIM/AM-measures,
• Reports/Documentation, extent of information data base.

The annual “Periodic Status Report” contains relevant results from in-service inspection, maintenance and surveillance measures.

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CONTROL OF PERMANENT SET STRUCTURES: CONDITION AT LIFETIME EXTENSION OF NUCLEAR POWER PLANTS

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At lifetime extension of nuclear power plants control of permanent-set structures state and their materials properties change forecasting is of crucial importance. For nuclear power plants RBMK-type – it is graphite stack, for nuclear power plants VVER-type – reactor vessel. Solution of this problem requires representative system of monitoring of such structures and reliable, physically justified models of main materials properties changes under the action of exploitation factors, which define lifetime of corresponding structural elements.

Justification of safe exploitation of nuclear power plants permanent-set structures beyond the scope of designed lifetime leads to necessity of development and implementation of principally novel methods of their condition control, such as

- Cutting of small pieces of metal (templates) from the non-cladded inside surface of the first generation VVER reactor vessels;
- Manufacturing and research of small-scale specimens from surveillance specimens, located in the higher containers of surveillance specimen assemblies, situated on the level of active zone edge;
- Reampouling of single-store 4-6 sets of VVER-1000 surveillance specimense and creation of novel, modernized programs of surveillance specimens;
- Drilling-out of specimens out of graphite bricks.

Of high importance is also implementation of advanced system of adjustment of radiation loading on the nuclear power plants elements, in particular, location of dosimeter monitors system in out-of-vessel area.

Moreover, when justifying models of changes of exploitation properties of structural elements beyond the scope of designed lifetime it is needed to conduct additional research for insertion of corresponding additions and qualified reports for these materials, confirming possibilities of materials use beyond the scope of fixed in qualified reports maximal fluences and materials exploitation time. All the relevant technical solutions, used at justification of lifetime extension, will be covered in the present report.

Dr. Strombach has worked in RRC “Kurchatov Institute” since 1972, he is currently the director of the Institute of Reactor Materials and Technologies of RRC “Kuchatov Institute” Dr. Strombach is involved in the research and development in the field of atomic energy with specific interests in the field of reactor material science, radiation solid state physics and radiation nanotechnologies. He is also engaged in the research and development in the field of creating new materials for NPPs and industry. The research and technology support of exploitation, modernization and lifetime extension of operating NPPs is also an area of scientific interest of Dr. Strombach. He is a member of the E-10 ASTM Committee and is the author of more than 100 publications in domestic and international issues.
The move to digital systems in petro-chemical, process and fossil fuel power plants is enabling major advances to occur in the instrumentation, controls and monitoring systems and approaches employed. The adoption within the nuclear power community of advanced on-line monitoring and advanced diagnostics has the potential for the reduction in mandated surveillances, more accurate cost-benefit analysis, ‘Just in time” maintenance, pre-staging of maintenance tasks, move towards true “operation without failures” and a jump start on advanced technologies for new plant concepts, such as those under the International Gen IV Program [1].

There has been a growing recognition that nuclear plant condition assessment based on NDT at the time of fabrication, followed by intense inspections during outages, requires the adoption of conservative assumptions with regard to addressing detected indications and intervention. With aging plants there is the risk of “surprises” at outages, that can cause extended down time. The move to on-line monitoring and condition based maintenance has the potential to increase operator situational awareness, enhance safety and provide significant cost savings.

There are significant opportunities when upgrades are implemented at existing facilities. The economic benefit from a predictive maintenance program can be demonstrated from a cost/benefit analysis. An example is that for the Palo Verde Nuclear Generating Station [2]. An analysis of the 104 US legacy systems has indicated potential savings at over $1B per year when applied to all key equipment [3].

For new plants the are even greater opportunities for improved operation, enhanced capacity factor and reductions in O&M costs through the adoption of true condition based maintenance philosophies, on-line monitoring and diagnostics. Adoption of digital I&C and new system technologies using on-line monitoring for more assemblies, wireless sensors and integrating total cost of ownership models can result in fewer surprises at outages, better planning for maintenance and many fewer unplanned outages.

Advances in technologies in other industries can potentially benefit new nuclear power plants, particularly when advanced on-line monitoring and diagnostics for condition based maintenance and in the future prognostics. The current status for various system elements is shown in Fig. 1.

New designs for advanced nuclear power plants, such as those within the Gen IV program, will require longer (potentially 4 years) between scheduled outages, and also shorter outages. To achieve such performance enhanced on-line monitoring is essential.
To move from periodic inspection to on-line monitoring for condition based maintenance and eventually prognostics will require advances in sensors, better understanding of what and how to measure within the plant; enhanced data interrogation, communication and integration; new predictive models for damage/aging evolution; system integration for real world deployments; quantification of uncertainties in what are inherently ill-posed problems and integration of enhanced condition based maintenance/prognostics philosophies into new plant designs, operation and O&M approaches.

<table>
<thead>
<tr>
<th>Diagnostic/Prognostic Technology For:</th>
<th>AP</th>
<th>A</th>
<th>I</th>
<th>NO</th>
</tr>
</thead>
<tbody>
<tr>
<td>Basic Machinery (motors, pumps, generators, etc.)</td>
<td>D</td>
<td></td>
<td>P</td>
<td></td>
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<tr>
<td>Complex Machinery (Helicopter Gearboxes, etc.)</td>
<td>D</td>
<td>P</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Metal Structures</td>
<td>D</td>
<td>P</td>
<td></td>
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<tr>
<td>Composite Structures</td>
<td>D&amp;P</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Electronic Power Supplies (Low Power)</td>
<td>D</td>
<td>P</td>
<td></td>
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</tr>
<tr>
<td>Avionics and Controls Electronics</td>
<td>D</td>
<td>P</td>
<td></td>
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<tr>
<td>Medium Power Electronics (Radar, etc.)</td>
<td>D</td>
<td>P</td>
<td></td>
<td></td>
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<tr>
<td>High Power Electronics (Electric Propulsion, etc.)</td>
<td>D&amp;P</td>
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</tbody>
</table>

**Fig. 1.** Diagnostics and prognostics: technology maturity matrix [4].

- **AP** – technology currently available and proven effective
- **A** – technology currently available, but V&V not completed
- **I** – technology in process, but not completely ready for V&V
- **NO** – No significant technology development in place

Dr. Bond is a Laboratory Fellow at the Department of Energy's (DOE's) Pacific Northwest National Laboratory (PNNL). He is an internationally recognized expert in applied physics with more than 25 years experience in measurement sciences. He joined Pacific Northwest National Laboratory in 1998 and became a Laboratory Fellow in 2000. He was seconded to the Idaho National Laboratory in February 2005 to be Director, Center for Advanced Energy Studies and returned to PNNL January 1, 2007.

His academic career started at University College London in 1979, when he was appointed to a faculty position. He held various positions and was promoted to “Reader in Ultrasonics.” In 1990 he moved to the US National Institute of Standards and Technology (NIST) and University of Colorado at Boulder, USA. His current research is focused on measurements for nuclear power systems. He was principal investigator for a DOE NERI Project (1999-02) to develop prognostics for nuclear power systems and co-IP for a 2002 NERI project to develop QA/QC tools for TRISO particles. He was the lead for PNNL’s Laboratory Level Advanced Nuclear Science and Technology Initiative (ANSTI) and was a major contributor to the development of PNNL’s Integrated Nuclear Strategy.

He has served on numerous conference committees including the American Nuclear Society 4th, 5th and 6th Topical Meeting on Nuclear Plant Instrumentation, Control and Human-Machine Interface Technologies (NPIC&HMIT). He is a US delegate to the International Electrotechnical Commission, WG 45, nuclear instrumentation. He was a member of the working group that recently submitted an instrumentation and controls, human machine interface technologies roadmap for advanced reactors to the US DOE.
Dr Bond earned his Ph.D. in physics from The City University, London, in 1978. He is a Fellow of the Institute of Physics (UK) and a Senior Member Institute of Electrical and Electronics Engineers (IEEE). He is Director-Elect IEEE Region 6 (’07-08). He has published more than 250 papers and holds 9 patents.

REFERENCES


NEEDS IN RESEARCH AND DEVELOPMENT SUPPORTING NUCLEAR POWER PLANT LIFE MANAGEMENT

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Due to current social and economical framework, in last years many Electric Utilities and Nuclear Power Plants adopted a framework for an improved coordination of both safety and non-safety programs, called Plant Life Management (PLIM). Some of them also started a process of Long Term Operation (LTO) for their older nuclear facilities, showing that a PLIM framework is particularly effective in such processes. The implementation of the PLIM framework followed many different approaches, being intrinsically dependent on the national regulatory framework and technical tradition. In particular, the LTO process has many nuclear safety implications, other than strategic and political ones, and therefore in recent years the need for tailoring the available safety assessment tools to such applications has become very urgent.

The analysis of the experience in countries operating Nuclear Power Plants (NPPs) suggested that most of the differences affecting the LTO programs are mainly related to the regulatory process (typically in the use of the periodic safety review), while the main components of the LTO program and its basic technical tasks are shared among most of the countries.

Most of these tasks are rather general; however in many cases they need reshaping in a PLIM framework, with special focus to the safety implications of the LTO. Their standard features, developed for plant still in their design life, need some modifications to support a long term decision.

This is why R&D tasks are needed in this phase, not only in the long term (i.e. more than 10 years of the standard periodicity of the Periodic Safety Review process) extrapolation of the component integrity and behaviour, but a also in new management strategies at the plant (PLIM), able to address organisational issues, spare part management, staff ageing, component obsolescence, etc.

The European Commission, especially in the framework of the EURATOM programme, identified specific R&D priorities to be mainly addressed by “direct actions” managed by the Joint Research Centers (JRC). The best chance for implementation is given by the newly launched Framework Programme 7 (FP7), where large coverage is given also to the nuclear safety issues. This paper, making reference to these priorities identified in the FP7, describes the R&D tasks that could effectively support Plant Life Management Models (PLIM) at NPPs, more directly affecting the decision for a long-term safe operation of a nuclear facility. In particular the paper provides an analysis of the research actions already in progress at the European Commission-Joint Research Center (EC-JRC) and their preliminary results.
Michel Bièth joined the EC-Joint Research Center of the European Commission in 1990. His current assignment after being Head of Unit, Technical and Scientific Support to TACIS PHARE from June 2001 until March 2007, is Head of Unit, Nuclear Operation Safety. The scope of work covers the full range of nuclear safety for Russian designed reactors and facilities: On-Site Assistance, Design Safety, Off-Site Emergency Preparedness, Safeguards, Industrial Waste Management and Decommissioning, Regulatory Assistance. He also is in charge to develop and implement inside the EC-R & D Framework Programme 7 (2007-2013) the JRC projects SONIS (Safety of Operating Nuclear Installations including the European Networks SENUF), and NUSAC (Nuclear Safety Clearinghouse for Operational Experience Feedback).

REFERENCES

1. Background

Now, 55 nuclear power plants (hereinafter referred to NPPs) are commercial operation in Japan. The number of ageing NPPs which have been in operation more than 30 years will exceed 20 in 2010 and 30 in 2015.

Licensees are required to conduct ageing management technical assessment before 30th years of commercial operation and reassessment less than every 10 years thereafter as a regulatory requirement to ensure safe operation and integrity for long-term operation. The regulating agency, Nuclear Safety and Industrial Agency (hereinafter referred NISA) reviews the appropriateness of the ageing management assessments.

Ensuring safety and integrity of ageing NPPs, it is important to implement adequate preventive maintenance and replacement and apply new technologies for proper inspection and assessment of each system, structure or component (SSC). So, it is necessary to improve inspection system and ageing management and promote safety researches for securing the stability of ageing NPPs.

Meanwhile, ageing management issue has become of great concern to the local community and the general public because of the secondary system pipe rupture accident at Unit 3 of Mihama NPP, run by Kansai Electric Power Company, in August 2004. Therefore, NISA established the ageing management review committee in order to verify and improve ageing management measures and has been studying how the inspection system of government should perform in the future.

NISA submitted a report, “Improvement of Ageing Management for NPPs,” and the report was approved by the committee on August 31, 2005. Securing the stability of Ageing NPPs, the current status in Japan will be reported on improving inspection system and ageing management and planning of technological strategy for safety research basis.

2. Improving Inspection System

As is conventionally, the regulatory has inspected the safety of NPPs mainly as periodic inspection and periodic safety management review during annual plant shutdown period. NISA
has been studying the following measures for improving inspection system to apply not only plant shutdown but also operation periods.

(a) To formulate maintenance programs for improving and strengthening plant-by-plant maintenance activities through the formulation of maintenance programs and shifting from uniform to fine-tuned inspections, in accordance with individual plant characteristics.

(b) To introduce an inspection system focused on important steps to ensure safety for strengthening inspections of plants during both they are in operation and shutdown periods. The government confirms important steps to ensure safety, such as startup or shutdown, during safety inspections.

(c) To promote the establishment of guidelines for licensees to actively analyze the root causes of accidents and trouble due to human error or organizational factors, such as the accident at Mihama nuclear power plant unit 3.

3. Improving Ageing Management

The following issues have been confirmed as fundamental requirements to improve ageing management in the report, “Improvement of Ageing Management for NPPs”. By further implementing the following measures, it will be expected scientific and rational ageing management for long term operation.

(a) To assure transparency and effectiveness through clarification of basic requirements for ageing management and content of guidelines and standard review procedures, etc. by the regulator.

(b) To provide a technical information basis for supporting scientific and rational implementation of ageing management, including the expansion of international cooperation.

(c) To prevent non-physical degradation such as safety culture, organizational climate, and technical capabilities in maintenance and administration.

(d) To fulfill clear accountability to the general public to eliminate anxiety about “obsolescence” resulting from long-term operation.

4. Planning of Technological Strategy for Safety Research Basis


The roadmaps have been developed in the four major categories as follows.

(a) Establishment of information basis
(b) Technical development for ageing management
(c) Establishment of codes and standards
(d) Systematic maintenance management

The roadmaps are the excellent references for wide variety of safety research. Based on the roadmaps, industries, government and research organization including universities will promote safety researches effectively and cooperatively by considering their roles.
The roadmaps should be reviewed by reflecting technological strategy that will be reviewed continuously by a coordinating committee on ageing management established with members from industries, utilities, research organization and regulatory authorities.

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EUROPEAN RESEARCH NETWORK AIMING AT HARMONISED PLANT LIFE PREDICTION PROCEDURES

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The European Network of Excellence NULIFE (Nuclear Plant Life Prediction) has been launched with a clear focus on integrating safety-oriented research on materials, structures and systems and exploiting the results of this integration through the production of harmonised lifetime assessment methods. NULIFE will help provide a better common understanding of the factors affecting the lifetime of nuclear power plants which, together with associated management methods, will help facilitate extensions to the safe and economic lifetime of existing nuclear power plants. In addition, NULIFE will help in the development of design criteria for future generations of nuclear power plant.

NULIFE was kicked-off in October 2006 and will work over a 5-year period to create a single organisation structure, capable of providing harmonised R&D at European level to the nuclear power industry and the related safety authorities in the area of lifetime evaluation methods for structural components. While over half are from the research sector, NULIFE also involves many industrial organizations and, in addition to their R&D contributions, these take part in a dedicated End User Group. Joint research and development activity priorisation procedure and first pilot projects are defined.

Led by VTT (Technical Research Centre of Finland), the five-year project has a total budget in excess of EUR 8 millions, with partners drawn from leading research institutions, technical support organisations, power companies and manufacturers throughout Europe. A key element in the Network structure will be the four Expert Group activities, which will cluster “centres of expertise”. Key organisations, Contractors of NULIFE are:

- VTT, Technical Research Centre of Finland, a Coordinator of NULIFE
- Studiecentrum voor Kernenergie - Centre d'Etude de l'Energie Nucléaire, SCK•CEN from Belgium
- Nuclear Research Institute Rez plc, NRI from Czech Republic
- Commissariat a L'Energie Atomique, CEA and Electricité de France, EDF from France
- AREVA NP GmbH from Germany
- European Commission Directorate General Joint Research Centre, JRC, from EC
- British Energy Generation Ltd and Serco Ltd from UK and
- Forsmark Kraftgrupp AB from Sweden.

NULIFE’s vision is to create a virtual institute with 1) An integrated RTD platform embracing all European stakeholders within a completely new structure with improved and efficient use of public and private RTD funding, 2) Sustainable forum for realising harmonised technical procedures giving impact for Nuclear energy industry, National
regulators and European Regulatory Working Groups and 3) Service provider and sustainable source of qualified expertise for all customers in Nuclear energy field.

The path towards the vision is described in Figure 1, proceeding through different phases of integration evolution and finally reaching NULIFE Institute with customer-driven programme.

**FIG. 1. Planned organizational evolution during the five phases of NULIFE (2006-2011)**

Viable expert groups and coherent network structure with efficient communication methods will be key indicators of successful operation and integration of the first year. The NULIFE questionnaire for mapping expertise has been launched. Research organizations and service providers are asked to define the current expertise, and end users and vendors are asked to define strategic needs. This covers different reactor types, experimental conditions, system and component types as well as material types. Life management expertise is broken down into areas such as degradation modes, load effects, condition monitoring, inspection, integrity assessment and safety management. Testing facilities, organization/process related expertise, dissemination skills as well as design, manufacturing, operation expertise are also all being considered.

Nuclear plant life prediction procedures are an essential part of nuclear safety, therefore, the research focuses on both development of procedures but also on special fields as integrity and degradation mechanisms of key components. In Finland this research is traditionally carried out in national nuclear safety programmes, today in SAFIR 2010 programme, aiming at training of new scientist and linking international co-operation with the Finnish research as well. The main target is to maintain and improve knowledge needed for life management of the present power plants but also to be ready for the future needs. The programme is formed of eight areas where reactor circuit is one of the key topic consisting of research on: environmentally assisted cracking, structural integrity and monitoring of integrity. This work is in many parts complementary to research in European programmes.
RECENT PLIM ADVANCES FOR CURRENT OPERATION AND LONG LIFE

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While plant aging is inevitable, continuous improvements can be applied at various stages of the plant design and operations life cycle to ensure predictable, reliable and graceful “aging” behaviour. Comprehensive and integrated Plant Life Management (PLiM) programs and associated technologies are key management tools for identifying these improvements. PLiM technology has grown and advanced to be applicable to new plants as well as older ones, although the purposes and key outcomes vary.

For older plants, PLiM techniques are often used to ensure design life is successfully and reliably attained and to provide prognosis for life extension, including technical and cost inputs to life extension business cases. For new plants, PLiM is helping utilities develop effective plant programs in preventive maintenance, surveillance and inspection, both for active and passive components, and to start building Life Cycle Management plans for the most important Structures, Systems and Components. For new designs, the objective is to understand potential aging degradation for both passive and active components, incorporate operating experience, and ensure design margins and measures are taken to assure component reliability can be achieved. This is important to demonstrate that the design has been improved for high capacity factors and longer design life.

Atomic Energy of Canada Ltd. (AECL) has worked with many CANDU utilities on Plant Life Management over the last 10 years. However, several recent applications and developments have led to further advances in PLiM technology.

PLiM technologies have been applied to Life Extension of AECL’s NRU reactor and, most recently worked with Nucleoeléctrica Argentina SA (NA-SA) to improve the aging assessment process for the Embalse NPP in Argentina. Examples of these advances are:

- For NRU, the systematic assessment process has been enhanced, including refinements in the reactor screening and decision process, to meet different needs of the facility operator.
- For Embalse, the challenge was to have both utility and AECL engineering groups generate consistent aging assessments. This involved advanced training on all aspects of PLiM, further refinements to the decision making process, improved definition and control of information flow, and improvements in detailed information for evaluation of various Age Related Degradation Mechanisms in process/mechanical, electrical, instrumentation & control, and civil components.

A further area of improvement in CANDU PLiM is related to active component preventive maintenance (PM) assessment. Even utilities with relatively new plants, such as China’s TQNPC Qinshan 1&2, are interested to improve their PM program effectiveness. While plant PM is sometimes not directly associated with PLiM, increased focus on the effectiveness of maintenance programs to deal with aging, both long and short term, is driving the nuclear
industry to better understand the overall system maintenance strategy. For instance, the INPO AP-913 Equipment Reliability guidelines and also IAEA guidelines involve detailed equipment reliability assessments. To assist with a systematic and rigorous assessment of PM program effectiveness and to guide the assessment process to achieve consistent and well documented basis, AECL has developed the SYSTMS tool and the AECL Maintenance Template Database for specific application to CANDU plants.

As systematic assessment of both long term degradation and short term degradation (such as wear) have advanced, AECL have developed techniques and methodologies to handle both active and passive components into a single systematic approach. This has many advantages, including improvements for important components such as major rotating machinery. With this type of equipment, both passive mechanisms (eg. those affecting the pressure retaining function) and active mechanisms (eg. shaft rotation and the associated wear/fatigue) are important potential degradation mechanisms to ensure are adequately dealt with.

Another important application of PLiM has been in outage frequency improvement (extending the time between planned outages, for instance from 2 years to 2-1/2 or 3 years). These durations are possible in CANDU reactors because of on-line refuelling. However it also places greater reliance on the effectiveness of predictive and preventive maintenance, as it implies fewer opportunities are available to do corrective maintenance (hence there is less tolerance of any inadequacies in the maintenance strategy). Systematic aging understanding and rigorous assessment (such as that from a comprehensive PLiM program) and a well documented technical basis are key inputs into an effective preventive maintenance strategy that supports the longer time between outages. For instance, if systematic and rigorous assessment (ie PLiM) is applied to the key Structures, Systems and Components, then understanding the potential impact on aging and the associated implications are simpler and quicker to identify.

It is also anticipated that the evolution and advances of PLiM to assist in improvements in design and operation will continue in the years ahead. One such advance is to move PLiM aging assessment to "quantitative" risk-based assessment (such as risk-based inspection - RBI). Detailed and refined assessment processes are being further adapted to reflect RBI requirements. Aging assessments on large passive components identify the specific areas and sub-component regions that should receive special age management attention, often by increased inspection effort. There is also the potential to focus in-service inspection effort to these age-sensitive locations and hence potentially reduce inspection effort. The key to implementing PLiM results and recommendations in plant decision making on changes to the inspection program is to answer the question “what is the change to risk?”. Risk based techniques can be a useful means of weighing the options and defining the next steps to manage the risk eg. uncertainties leading to higher risks on components.

The confidence regarding the level of understanding of the degradation contributing to the risk assessment is important. Protocols for this risk assessment that have been developed in various non-nuclear industries are being investigated for potential adaptation to existing CANDU PLiM assessment techniques. The result would be a well substantiated risk basis for inspection.

In summary, there are several areas of advances in recent CANDU PLiM technology and applications. This paper will further detail these advances and provide examples.
KEY ELEMENTS OF LONG TERM OPERATION OF WWER-440/213 UNITS AT PAKS NPP

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The WWER-440/213 units at Paks NPP are approaching to their last decades of licenced and designed term operation. From technical and economical point of view the extension of operational lifetime is feasible and from the legal point of view renewal of the operational licence is possible in accordance with Hungarian regulation. The plant decision is to renew the operational licence and operate the plant 20 years additionally to the existing one.

Systematic preparatory work for long-term operation (LTO) of Paks NPP WWER-440/213 units had been started in 2000 and it is going on now. The first formal step is already done: The environmental impact study of LTO is completed and the formal procedure of the environmental licesing is passed. The second and most decisive step ahead is the development of the Programme of LTO, which has to be submitted for approval to the Hungarian nuclear regulatory authority by the end of 2008. The formal application for licence renewal (LR) has to be made in 2011.

Parallel with the preparation of LTO the second periodic safety review (PSR) is going on. The report on PSR shall be submitted to the regulators by the end of 2007. According to the Hungarian regulation PSR is a current licensing basis requirements and it is not a tool for LR or LTO. Nevertheless the PSR is also a source of technical information applicable for the development of the Programme for Long-Term Operation due to principal redundancy between PSR and LR goals and essential overlapping between the scope of the PSR and LR. An other parallel activity is the update of the Final Safety Analysis Report which has been completely renewed in 2004, consists design base information and corresponds to the actual configuration of plant/units.

The main result of strongly interrelated and partly redundant activities mentioned above is the identification and understanding of key technical elements of LTO of WWER-440/213 units generally and particularly at Paks NPP.

Core tasks of the preparation of LTO are directly linked to the requirements of the relicensing: assessment of the plant status, review and demonstration of adequacy of plant ageing management programmes (AMPs) with focus on ageing management of safety related, long-lived, passive and non-replaceable structures and components, as well as review of time limited ageing analyses (TLAAs). Review of both the existing AMPs and TLAAs may result in development of new or modification of existing AMPs or other plant actions.

Concerning the core tasks of licence renewal following findings are important:

(1) The number of structures and components within the scope of LR is very large. This is caused on one hand by evolutionary character and very complex design features of WWER-440/213 type on the other by the deterministic way of safety classification, which obligatory classifies in a large number of components with apparently marginal
contribution to core melt frequency. Magnitude of the scope multiplies all the effort needed for preparation of LTO and LR.

(2) Although the plant has AMPs for most important structures and components, the review performed shows necessity of development of new programmes for covering the scope and also upgrading of existing ones. Because of the extent of the scope definition of commodities and structuring the AMPs on the basis of proper understanding of ageing phenomena became very important. Specific IT tools have been developed for supporting the management of the activities related to the scope.

(3) A very important aspect of justification of safety of LTO is the review and validation of the TLAAs. In the specific case of Paks NPP the designer made TLAAs are missing or inadequate because of lack of proper documentation or obsolescence. For the solution of the issue the necessary analyses (stress calculations, PTS, fatigue, etc.) have to be performed by state of the art methods, which means also checking of the design calculations in accordance with ASME BPVC requirements of most important structures and components. In the paper the mentioned above specific aspects of LTO and LR of WWER-440/213 units are discussed in detail. Measures and actions for solving the problems arisen during mentioned above preparatory work are part of the Programme of LTO.

Programme of LTO is a frame of all plant activities ensuring safety of operation beyond designed lifetime of the plant. Plant programmes (programmes for ISI, maintenance, EQ and reconstruction) have to be reviewed and their adequacy for ensuring safe LTO has to be demonstrated, modified, upgraded if necessary. These plant programmes and the measures defined during their review are also part of Programme of LTO. The Programme covers also human resource and knowledge management aspects of LTO. In the paper an overall picture of development of the Programme of LTO of Paks NPP is given.
NPP LONG TERM OPERATION IN SPAIN – FIRST APPLICATION FOR LICENSE RENEWAL

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In the operation of the Spanish nuclear power plants (NPP), safety is always the prime consideration. Plant Life Management Programmes have been set up with the strategic objective to operate the NPPs as long as they are considered safe and reliable. The safety of each NPP is reviewed by the Spanish nuclear regulatory authority (CSN) under a continuous process. In addition, experience is gained from operating the plants and from exchanges with operators of similar units.

Current Spanish regulatory framework for renewing NPP operating licenses requires performing a Periodic Safety Review (PSR) to be performed every 10 years and submitted when applying for a new renewal of the NPP operating license.

A few years ago, CSN issued a document regarding the licensing requirements that nuclear power plants should meet in order to be granted with an operating license for long term operation (i.e., operation beyond the original plant design life, typically 40 years).

Besides the traditional PSR requirements, specific requirements regarding to long term operation (LTO) include:

- An Aging Management and Evaluation Program, including the identification and evaluation of Time Limited Aging Analysis (TLAA).
- An updated Radiological Impact Study.
- A review and assessment of regulation/standard applicability.

Garoña NPP (GE, BWR/3 design) operated by Spanish utility Nuclenor from 1971 has a current operating license up to 2009. A decision was made to apply for a new operating license, being Garoña plant the first one in Spain to face with the new long term operation requirements.

The paper will provide an overview of the methodology used in Spain to address and perform the required analyses to support the LTO application for the operating license renewal. In particular, focus will be paid on the project developed in Garoña (2002–2006) whose result has been the first Spanish application for License Renewal for LTO. Also it will be reported the ongoing work necessary to be completed before the beginning of the new extended life period (Implementation Program) and the issue on the applicability of new standards.
IMPLEMENTATION OF THE RULES FOR THE CONTINUED OPERATION IN KOREA

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Most of the countries do not grant license term when they permit operation of the nuclear power plants. With exception, The US solely state the initial license term for forty years in Atomic Energy Act, as amended in 1954. Many other countries issue new operating licenses upon completion of each refueling outage or Periodic Safety Review (PSR) during the life of the plants. Accordingly, licensing renewal processes are different from country to country. Hereinbefore, the life of the plants is regarded as upper bound lifetime of the passive and long lived structures, systems, and components. In Korea, the initial operating license loses validity with expiration of the design lifetime stated in Final Safety Analysis Report (FSAR). The legislation on the Continued Operation (CO) beyond design lifetime has been completed in 2005, by amendment of the current PSR rule. The revised rule allows the continued operation for another ten years beyond the design lifetime.

Basically, the CO regulation is an extension of PSR, in that two rigorous requirements were added: one is aging management programs including time-limited aging analysis, and the other is assessment of radiological impacts on the environments. PSR is a useful tool to evaluate the safety level of the operating plants. The principles of the PSR requirements are same as those delineated in IAEA safety Guide 50-SG-O12. The PSR should be conducted for all operating plants every ten years with eleven safety factors. Each safety factor should be reviewed using the currently effective safety standards and practices to the plant of interest, which is defined as current licensing basis.

With expiration of the design lifetime, the licensee can submit applications to the government for the CO. Licensing renewal for the continued operation expires every 10 years. It is not restricted that how many times licensing renewals can be permitted. In order to apply for the CO, the extended PSR including aging management programs with time limited analysis and assessment of radiological impacts on the environment, should be performed. Licensee should submit the review report to regulatory body for safety evaluation. Regulatory body may bring up safety related issues and recommend its implementation to licensee as a result of safety evaluation. Implementation scope and process of the rules for the continued operation in Korea are shown in the figure.
FIG. 1. Flow Chart of Implementation process for the CO

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NUCLEAR POWER PLANT LIFE MANAGEMENT: STRATEGY FOR LONG TERM OPERATION OF THE BEZNIAU NPP UNIT 1 AND 2

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Both Beznau nuclear power plants (KKB) are of the Westinghouse two loop PWR type with a rated electrical output of 380 MW and have been operated on base load demand since 1969 and 1972 respectively. The design base uses rather conservative assumptions. By performing selective measures, such as large investments in backfitting, the safety of the plants has been continuously enhanced. Through focussed modernisations and careful maintenance, the overall condition of both Units is excellent. A comprehensive ageing management programme (AMP) for all safety related structures, systems and components (SSC) was set up and started 15 years ago, according to the requirements of the Swiss Federal Nuclear Safety Inspectorate (HSK).

The AMP contains all essential actions of evaluation and control of the ageing concerning the material and also conceptual aging. The permanently increasing demand for electrical energy in Switzerland, as well as economic factors, led NOK to evaluate long term operation (LTO) of KKB. Beside hydro power, nuclear power covers 40 % of the demand in Switzerland. A new nuclear law in Switzerland allows unlimited operation of a NPP as long as safety goals are met. The performed analysis and conclusions for a potential LTO is presented in the paper, including aspects such as technical issues, fuel, radwaste, elusive risks, personel mangement and economics to assure LTO with the best achievable safety level.

The necessary actions to address and control the ageing mechanisms of the civil strucures have been established in line with the AMP. The procedures for implementing these actions are in place and are continously executed. The technical feasibility for the essential actions required for a LTO up to 60 years is given.

A systematic registration of the actions to control ageing of all electrical systems and components occurs in line with the AMP. The main challenge is not the material related ageing but the availability of products and manufacturers and the conceptual aging due to obsolescence. Backfitting and modernisation of complex systems are permanent tasks. Also in the electrical area, the technical feasibility for the essential actions for LTO up to 60 years is given.

The AMP attests generally good conditions for the major components of the nuclear steam supply system (NSSS). Exceptions are some materials, such as Alloy 600, used in the reactor pressure vessel. Special attention is required in the area of embrittlement and fatigue of selected components. Selective replacement of several components is necessary. Maintenance activities require more resources especially for the qualified test methods requested by the authorities. The investigations and analysis show also for the NSSS the technical feasibility for the essential actions for a LTO up to 60 years is possible.
Most of the systems and components of the balance of plant have been replaced in the past years. Starting in 1993 with the HP-turbines, the condensers and the pre-heaters followed by the entire turbine controls. The last big component, the moisture separator/re-heater, will be replaced in the 2007 outage. The balance of plant is ready for LTO up to 60 years.

In the area of fuel acquisition, storage and disposal, long term contracts until 2020 are in effect. Spent fuel elements are first stored in the fuel-pools and later stocked in storage containers in a facility on the site. A national company owned by the utilities is assigned to search for an underground final disposal repository.

Elusive risks, such as a nearby hydro plant, that is part of the emergency power supply of KKB, new earthquake analysis for all NPP’s in Switzerland and aspects of the environment in the sector of nuclear energy are covered by the study for a LTO of KKB.

The workforce of KKB, currently 490 people, will show a slightly increasing tendency due to increasing maintenance activities. The management of spare parts has to be in line with LTO.

The electricity market in Europe and particularly in Switzerland as well as external influencing factors such as CO\textsubscript{2} issues are covered by the study. For the economic aspects, the entire study was carried out in terms of three scenarios; optimistic, realistic and pessimistic cases, for 50 years as well as for 60 years of full-power operation. A comparison between the production costs versus the market prices shows good perspectives.

The KKB was operated gently for base load only for the past 37 years, the number and magnitude of transients show only “half time” compared to the design. The safety status can be compared with new plants. If the identified actions can be realised, LTO is technically as well as economically feasible. At the end of 2005, the NOK executive board assigned the management of KKB to initiate all necessary steps and measures for LTO of at least 50 years.
PLANS FOR LONG TERM OPERATION IN ANGRA NPP

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Located in Angra dos Reis, in the south of the State of Rio de Janeiro, Almirante Álvaro Alberto Nuclear Power Station or, simply, Angra NPP, comprises two generating units in operation. Unit one, operating commercially since 1985, is a Westinghouse designed PWR, generating 657 MW, while Unit two, designed by Siemens, is a 1350 MW PWR in operation since 2000. The production of both units accounts for about 3% of the total electricity produced in Brazil and 40% of the Rio de Janeiro State consumption.

Over the last years, Eletronuclear has been concentrating efforts to provide the basis for an effective management of its plants life cycle, coordinating the implementation of a systematic ageing management and the development of a tailored computational system for economic analysis.

Unit 1, being the oldest plant of the Station, has been chosen as the host plant for the development of the first Ageing Management Programme (AMP) at the company. Angra 1 AMP has been established with the following objectives: coordinate long term operations, maintenance and engineering actions to control, under acceptable limits, the effects of ageing on integrity and functional capability of systems, structures, and components important to safety; comply with regulatory requirements for Periodic Safety Reviews; and proactively prepare for License Renewal.

In conjunction with the current maintenance, operations and engineering programmes, the following ageing related programmes, in course or being implemented in Angra, are credited to support long term operation plans: Steam Generators AMP, Reactor Pressure Vessel AMP, Fatigue Monitoring Programme, Flow accelerated Corrosion Programme, Obsolescence Management Programme, Alloy 600 Programme, and Boric Acid Leakage Corrosion Programme.


The structures and components subject to ageing management review are limited to those that perform an intended function without moving parts or without a change in configuration or properties and those that are not subject to replacement based on a qualified life or specified time period.

Plans for long term operation requires, however, the definition of a more comprehensive scope to include systems, structures, and components important to economy not addressed in the scope of the AMP. Observing that the implementation of the Ageing Management Programme for Angra 1 is still in course and a systematic process for ageing management in
Angra 2 has not yet been introduced, the long term plans currently in development are based on experience and on specific studies. The most important sources of information for those studies are: operational experience of Angra 1 and Angra 2, external operational experience, ageing assessment, obsolescence assessment, safety analysis, corporate goals, and economic analysis.

Long term plans for Angra power plants are classified in accordance with the management goals they meet, which are: keeping plant performance at design required levels, production enhancement, and life extension.

Investments to keep plant performance at design required levels are those aimed to prevent generation discontinuity, power reduction, performance degradation or, even, the premature decommissioning of the plant. The production enhancement may be obtained by power uprate or by any other modification in design or procedures that enable any reduction of outage duration. Life extension requires the implementation of long term programs, design modifications, and performance of technical assessments that meet the requirements of a License Renewal process.

A computational system developed by Eletronuclear is used in economic analysis. Investments can comprise a set of repairs, replacements, design modifications, safety analysis or any improvement to the station, combined to support the aforementioned goals. Economic analysis, based on the Net Present Value method, provides estimates for the price of the KW-h that renders the investment viable.

The most significant investments for Angra 1 and 2, evaluated for each management goals, are summarized in the following table:

<table>
<thead>
<tr>
<th>Angra 1</th>
</tr>
</thead>
<tbody>
<tr>
<td>Keeping plant performance at design required levels</td>
</tr>
<tr>
<td>Replacement of the steam generators, replacement of the reactor pressure vessel head, repair or replacement of parts in alloy 600, refurbishment of turbine rotors, replacement of valves</td>
</tr>
<tr>
<td>Production enhancement</td>
</tr>
<tr>
<td>Design modifications in the secondary circuit for 40MW uprate, replacement of turbine rotors</td>
</tr>
<tr>
<td>Life extension</td>
</tr>
<tr>
<td>Licensing, replacement of components, replacement of I&amp;C equipment, acquisition of a simulator</td>
</tr>
<tr>
<td>Angra 2</td>
</tr>
<tr>
<td>------------------------------------------------------------------------</td>
</tr>
<tr>
<td>Keeping plant performance at design required levels</td>
</tr>
<tr>
<td>Improvements in the electric generator, improvements in the transformers,</td>
</tr>
<tr>
<td>replacement of valves, replacement of mechanical components, design</td>
</tr>
<tr>
<td>modifications</td>
</tr>
<tr>
<td>Production enhancement</td>
</tr>
<tr>
<td>Safety analysis for a 65MW uprate, modifications in secondary circuit</td>
</tr>
<tr>
<td>for a 15MW uprate, replacement of turbine rotors for an uprate of 45MW,</td>
</tr>
<tr>
<td>design modifications to reduce outage duration</td>
</tr>
<tr>
<td>Life extension</td>
</tr>
<tr>
<td>Licensing, replacement of components, replacement of I&amp;C equipment</td>
</tr>
</tbody>
</table>

*Most Significant Investments Considered in Angra 1 and 2 Long Term Plans Economic Analysis*

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AGING MANAGEMENT OF EDF NPP: FROM THE DESIGN PHASE UP TO END OF LIFE

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This paper will present the different step in EDF NPP, from design phase to take care of potential ageing mechanisms, to end of life of the plant to assure continuously safe and economical operation with ageing of the plant. It will consider:

- the potential changes in regulation,
- the design phase with regulatory, Code and User requirements
- the maintenance plan at design phase
- the monitoring, surveillance, ISI and data collection
- increase in knowledge considerations
- the periodic safety review
- the maintenance optimization with plant ageing consideration
- repair, replace or continue to inspect, an important challenge
- the role of Codes & Standards in this process

All the major actions that are important and relevant to assure periodic justification of our Ageing Management Program will discussed as a conclusion.
MAINTENANCE MEASURES RELATED TO PLANT LIFE MANAGEMENT TAKEN BY JAPANESE PWRS

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1. Current status of ageing nuclear power plants in Japan:

In Japan, 13 of 55 commercial light water reactors in operation will have operated for 30 years at the end of 2007. Besides them, additional 21 reactors will mark the 30th year of operation within the coming decade. Accordingly, plant life management (PLM) for nuclear power plants has become an urgent issue to be dealt with.

2. Preparation of utilities’ PLM technical evaluation reports and future prospects:

The Japanese utilities are supposed to implement plant life management technical evaluation assuming a 60-year operation period before their plant marks the 30th year of operation, and prepare a long-term maintenance plan for the coming 10 years. In 2005, related laws and regulations were established to require the utilities to prepare and submit a report describing PLM technical evaluation results. The reports for 12 reactors had been prepared and submitted to the government as of the end of 2006. In addition, the utilities are required to submit the long-term maintenance plans as well as their results of subsequent inspections.

The Japanese utilities should enhance their maintenance activities on a continuous basis while complying with the requirements regarding the presentation of the report specified by the laws and regulations. The industry, government and academia have been working closely in formulating the strategies regarding the development of technologies and codes and standards in view of enhancing preventive maintenance and plant reliability. Those strategies are to be developed as a roadmap. We as the industry has been actively participating the discussion and plans to take actions in compliance with the strategies.

3. Maintenance measures implemented for representative PWR components:

Under the circumstances mentioned above, the Japanese PWR utilities including Kansai Electric are implementing maintenance measures from the viewpoint of plant life management as follows:

(1) Replacement of major components

One of the main maintenance options for the components for which failures and events have been reported at home and abroad or the possibility of damage cannot be ruled out due to severe service conditions is the integrated replacement. For example, the reactor vessel head, which is one of the major PWR components, has been already replaced with new one at some plants considering the possibility of stress corrosion cracking (SCC). Regarding the primary pipe, the utilities have been working on the maintenance of the parts with possibilities of failure due to fatigue or SCC. In recent years, some Japanese
NPPs have also experienced fatigue failure in the pipe sections where hot and cold fluids are mixed or with the possibility of cavity flow. Therefore, the utilities are also required to address the pressing issue by taking actions against such failure. This section discusses the background, policies, and current status regarding the replacement of major components performed by the PWR utilities including Kansai Electric.

(2) Inspection and preventive maintenance of major components

For the components which are exposed to severe service conditions and difficult to be replaced, confirming the integrity by inspections and taking preventive maintenance measures are a suitable maintenance option. This section discusses the preventive maintenance techniques, background, application records and plans and inspection details mainly about Inconel 600 parts.

(3) Development of technologies and codes/standards

Enhancement of the database accumulated through technology development and establishment of codes and standards are essential in determining the timing of inspection and replacement. In this section, an improved equation to predict neutron irradiation embrittlement of the reactor vessel and recent developments in establishing the codes and standards regarding technologies to regenerate surveillance test specimens are presented. In addition, developments in the revision of the codes and standards regarding the secondary pipe wall thinning management for which the utilities have been continuously working on since Kansai’s Mihama-3 secondary pipe rupture accident in 2004 are also described.
THE ROLE OF LICENSE RENEWAL IN PLiM FOR U.S. NUCLEAR POWER PLANTS

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At the 1st International Symposium on Nuclear Power Plant Life Management (PLiM) in 2002, it was reported that the NRC had approved renewal of operating licenses for eight nuclear units, which would allow operation for 60 years (i.e., an additional 20 years from the original 40-year license term). Of the 103 operating nuclear units in the U.S. in 2002, it was anticipated that over 90% would eventually pursue license renewal. At that time, it was also concluded that the regulatory process was stable and predictable for license renewal, and that successful PLiM activities were helping to ensure the safety, economic, and political factors in the U.S. remained favorable for continued success with license renewal.

The status of license renewal in 2007 is even better than it was in 2002. As of August 2007, the NRC has approved renewal of the operating licenses for 48 nuclear units and has applications under review for 12 more units. In addition, nuclear plant owners of 26 more units have announced plans to submit license renewal applications over the next few years. This brings the total of renewed licenses and announced plans for license renewal to over 80% of the 104 currently operating nuclear units in the U.S. The prediction that over 90% would eventually pursue license renewal has nearly been achieved, and it now appears that 100% of the U.S. nuclear units will likely pursue license renewal. This positive trend for license renewal in the U.S. is attributed to: (1) the success of PLiM activities in achieving an excellent safety record of the nuclear power industry and in ensuring on-going positive economics for nuclear plant operation, and (2) the stable and predictable regulatory process for license renewal.

The NRC and the nuclear industry guidelines and processes for license renewal have continued to mature and become better documented since the original license renewal applications were submitted in 1998. For example, the NRC has updated and revised NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (now Rev. 1); NUREG-1801, Generic Aging Lessons Learned (GALL) Report (now Rev. 1); and the Nuclear Energy Institute (NEI) has revised NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 – the License Renewal Rule (now Rev. 6). These revised guidance documents now incorporate a number of lessons learned from the initial license renewal reviews and ensure greater consistence for future license renewal applications. In addition, the NRC has modified the regulatory review activities to focus more attention on on-site inspections and audits.

Long term operation of the U.S. nuclear plants is dependent on excellent PLiM that ensures continuing safe and cost effective plant operation. Safety remains the top priority for operating nuclear plants and the safety record in the U.S. continues to improve in parallel with cost effective operation. For example, the accident rate for nuclear plant workers is approximately 0.12 accidents (i.e., accidents resulting in lost work, restricted work, or fatalities) per 200,000 worker hours, which is well below the average for the electric utility industry at 2.0 and the manufacturing industry at 3.5. Another measure of safety is the
number of unusual events report by nuclear power plants. This indicator shows a significant trend downward from the 1980’s and 1990’s to a relative stable low number in the 2000’s.

This on-going record of safe nuclear plant operation has also helped in public opinion survey results. An NEI sponsored national public opinion survey report in April 2007 concluded that 81% of the U.S. public agree that we should renew the license of nuclear power plants that continue to meet federal safety standards.

Cost effective plant operation is also a vital part of plant life management. Though safety is the top priority for nuclear plant operation, cost effective operation is also necessary in order to allow long term operation. Data continues to confirm that the safest nuclear plants are also among the most economical to operate. This positive correlation between safety and cost effectiveness is one of the main reasons for the nuclear renaissance in the U.S. The industry's average production costs (including expenses for fuel, operation, and maintenance) were an all-time low of 1.66 cents/kwh in 2006. Average production costs have been below 2 cents/kwh for the past several years, which is highly cost competitive with other electricity sources that are capable of reliably producing large amounts of electricity.

The average capacity factor for the U.S. nuclear industry in 2006 was almost 90%. This average has been near 90% for the past several years. In the early 1980’s the average was less than 60%. Steady improvement in plant operation from the 1980’s through the 1990’s now results in the U.S. nuclear industry consistently averaging about 90%. In addition, the top 25% of the operating nuclear plants consistently reach greater than 95% annual capacity factors. These high average capacity factors are a major reason for the low cost of electricity produced by nuclear power plants.

In summary, the outlook for long term operation of the U.S. nuclear power plants is very positive. License renewal is a prerequisite for operating beyond 40 years and due to successful PLiM, the option to operate for 60 years is a realistic goal. In addition, preliminary PLiM information supports a second round of license renewal applications in the U.S. beginning in 2009. This would provide an option to operate many of the U.S. nuclear powers plants for 80 years and to keep the option open for even longer periods of operation based on high levels of safety and economic performance.
THE EXPERIENCE OF SERVICE LIFE PROLONGATION OF NPP UNITS OF THE FIRST GENERATION

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During the last 6 years in Russia there were executed works with regard to a substantiation of the service life prolongation for NPPs of the first generation with WWER-440, RBMK-1000, EGP-6 power units, which have 30 years service lifetime initially prescribed by the design.

For the purpose of NPPs lifetime prolongation there was developed an approach which enables to estimate the actual state of NPP components at the end of the design service life. The approach is based on the analysis of the operation history, design-experimental assessment of metal condition in zones with maximum damaging, determination of ageing mechanisms in such zones, selection of optimal methods and devices for non-destructive testing, carrying out the on-site inspection and providing verification calculation of components’ strength using all collected data. In accordance with the obtained results a conclusion about technical condition and residual lifetime is being performed.

By substantiation of lifetime management the NPP main equipment were analyzed from the point of view of the base mechanisms of operational ageing of materials. Also it the frame of service life prolongation work there were used special new devices intended for carrying out the assessment of NPP equipments’ metal condition, including devices for mechanical properties measurement, inspection of metal flow-accelerated corrosion wear, measurement of metal magnetic properties.

In the report there are presented the results of comprehensive researches of the pipelines’ and equipments’ metal condition, as well as modern inspection procedures are performed. Also there are reviewed data of original experimental researches of thermal-deformational ageing of full-scale test-samples used for prediction of the metal mechanical properties to the end of extra service life period till 45 years of unit operation.

Also the methodology of on-site inspection of metal irradiation embrittlement of the reactor power vessels is presented. To evaluate a technical condition and residual lifetime of NPP building units there were elaborated corresponding regulatory documents, being applied during on-site inspection of the actual state of NPP building units by use of modern procedures and techniques.

Regulatory documents, reviewed in the report, describe the procedure and the way of implementation of the technical state estimation and residual lifetime period forecast for NPPs pipelines and equipments outside the 30-years lifetime period.
EMBALSE NGS PLEX OVERVIEW

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Embalse Nuclear Generating Station (ENGS) is a CANDU 6 type with 648 MWe output with the First Criticality in 1983, and the Commercial Start up in January 1984. In the last 10 years it has shown an excellent performance with an average 89.27 % Capacity Factor (CF). Since 1992, planned outage programs at 18 month operation intervals have been implemented.

The end of the design life is foreseen for 2011. An initial decision about the intention for the ENGS PLEX has been taken.

A typical program for a CANDU PLEX has three main phases. Phase 1: Pre-project definition, Phase 2: Project Engineering and preparation and Phase 3: Project Implementation.

Phase 1 of the Refurbishment and Life Extension (RLE) project for the Embalse Nuclear Power Station consists of all preparatory activities that are required to define the refurbishment scope and costs, for input into the utility business case for the RLE project.

Organization:

ENGS has designated an organization to deal with the PLEX which is independent of the utility operation. Because of the different nature of the activities to be performed the organization chart will be changing over the phases of the project.

In April 2007 were hired 43 engineers to the staff of the PLEX project. After a training, which include a participation in the walkthrough performed by own and external experts, they started to work on the different assessment indicated bellow in the Aging Evaluation Section.

Current state – Phase I: Project Description

Four areas are distinguished:

Aging Evaluation

According to IAEA methodology and TECDOC [1], a systematic review of the plant is being carried out to determine what equipment refurbishment or replacement will be required due to aging or obsolescence of plant equipment. It should be define which refurbishment and / or replacement should be implemented during the so called “refurbishment outage” and which may be performed during “normal maintenance outages”.

They will also provide a health prognosis for continued operation of the SCCs for life attainment and life extension beyond the refurbishment outage, and may identify changes which are necessary and sufficient in order to deal with issues related to equipment obsolescence and aging effects.
Under the Aging Evaluation Program all the System, Structures and Components (SCC) of the plant are screened in order to determine which of them must be analyzed in order to assure the safe and economical Long Term Operation of the plant.

A first definition of the important system was based on the Point Lepreau and Gentilly II Nuclear Power Plant, on the AECL experience and on the judgement of experienced staff from ENGNS.

Given the complexity of the task it is necessary the participation of many actors (AECL, ANSALDO, B&W, etc.). For that reason a Division Of Responsibility (DOR) matrix was developed. This matrix defines the roles and responsibilities of each participant and also the relationship between the different assessments.

This Plant Condition Assessment (PCA) provides a structured approach by following these main steps:

1. Determination of systems important to nuclear safety or power production,
2. Generation of complete lists of constituent Structures, Components and Commodities (SCCs) in each system,
   Determination of those SCCs that are adequately addressed by normal maintenance activities,
3. Aging Assessment of those SCCs not addressed by normal maintenance, (Condition and Life Assessment)
4. Assessment to determine items that are obsolete and
5. Development of health prognosis and recommendations for items that need to be refurbished or replaced and the timing for these activities.

Digital Control Computers (DCC), electrical and I&C components

Because of the different required skills to deal with the electrical & IC degradation mechanisms was created a different department. Also the DCC arise as an important component to be refurbished and a Condition Assessment (CA) has been done.

Retubing

is the most relevant independent work to be performed during the refurbishment outage. Due to that the Phase 2 for that component has already been initiated.

For that issue a contract with the plant designer is under discussion for the assistance in that very important task.

For this activity as well as for the others to be performed during the refurbishment outage is the NASA intention to maximize the local participation. That is in design, engineering and spares provision.
Safety Assessment (SA)

It includes Licensing requirements, probabilistic and deterministic assessment (e.g. trip coverage), analysis of the design changes performed in other and news CANDU, and the review of the application of a PSR methodology analysis.

The paper will describe the Embalse Refurbishment Phase 1, the novelties and improvements of the aging assessment methodology and progress made to date in each of the four areas identified above. It will also include a general consideration about the organization and the future steps into the refurbishment.

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AGEING MANAGEMENT AT THE NPPS OF ENBW IN GERMANY

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Ageing Management (AM) at the NPPs of the EnKK (GKN and KKP) is to be reflected in respect to the political situation in Germany. The political decision to shut down the NPPs step-by-step and not to allow the new building of NPPs is the absolute opposite direction of the world-wide-trend of life-time-extension or building of new NPPs. AM is at the moment legally based on the RSK-recommendation “Management of ageing processes at nuclear power plants”, which was a result of the international discussion. Based on this recommendation and the experience in the implementation process of the AM process in German NPPs the experts are formulating a new technical guideline (KTA). Simultaneously the legal requirements in Germany are going to be developed, initiated by the BMU (Federal Ministry for the Environment, Nature Conservation and Nuclear Safety) and the WENRA-Process to harmonize the European regulations.

Based on these conditions the licensee EnKK created an Ageing Management Programm. Objective of this Programm is to ensure the high safety standard in our NPPs by applying standardized criterions and collimating informations to analyse them systematically (eg in a detailed analysis or a systematic weak-point analysis). Fundamental of this AMP is an integral approach.

The AMP in the NPPs of EnBW is based on the national RSK-recommendation which is integrating the national and international experience. Our AMP covers the technical issues (mechanical engineering, I&C, structural engineering and operating supplies) and is now being implemented at all NPP-sites. The non-technical issues like AM of personnel or documents, which are required by the RSK-recommendation, are treated separately. All the information about ageing effects are merged in the knowledge basis of the NPP which is itself the base for the AMP. This knowledge base includes all the documents, databasis and the knowledge of the personnel.

This paper is concentrated on the AMP of mechanical SSC as the most advanced issues.

For the AMP the SSC are classified on their relevance to safety. For mechanical SSC three groups are defined.

Group 1 addresses mostly passive SSC which are subjected to the integrity-concept. For them the quality is to be ensured and malfunctions are not allowed. Ageing effects have to be obviated.

Group 2 addresses the safety relevant SSC. The quality of this SSC is if possible to be ensured and if necessary to be reconstructed. Malfunctions are allowed in singular cases. Ageing effects should be minimized and systematic failures should be excluded. The required quality is ensured by preventive maintenance.
Part of group 3 are all the other SSC. SSC could fail and they are replaced after failure. The licensee is solely responsible for these SSC. In the AMP only the first and the second group are to be treated.

After the SSC are classified and the scope is defined the AMP - as a part of the management system – is attached to the different processes of the NPP. In the AMP the information and feedback of the operating experience (eg preventive maintenance programmes; in-service inspection, surveillance, testing and monitoring programmes; corrective maintenance) and so the AM-relevant information out of the relevant processes for countervailing ageing effects is bundled. Together with the information of transferable external events and research programmes the informations are systematically and analysed in detail by experts again. If the analysis and evaluation necessitates to further measures, these are communicated to the responsible person of the SSC and they are executed in the affected processes. These lead to a closed PDCA-Cycle, which is shown in Figure 1.

![PDCA-Cycle](image)

**FIG. 1. PDCA-Cycle**

On account of the numerousness of the SSC of the group 1 and 2 EnBW decided to implement a computer-assisted procedure. Therefore collectives of components were established that were identified because if there eg identical in construction, character of redundancy, same size and similar stresses (pressure, temperature and medium).

Documentation and status-sheets were prepared for the collection of data. Three different types of documentation and status sheets exist, which include the relevant data for the analysis in the AMP, such as description of the component or the collective, technical data, requirements, measures for preventive maintenance, supporting documents, methods for monitoring, history of the component and the results of the AM-analysis. These sheets are part of the knowledge base.
Furthermore these sheets are elements of the computer-assisted procedure where the information is flowing e.g. directly from the experience of the preventive maintenance in to the AMP to be analyzed completely in a systematic, reproducible way and independent of individuals.

In the last years EnBW has established a computer-assisted AM to execute advanced analysis of scheduled or unscheduled measures and events. The former, which is based more on case-by-case activity, is being changed to an integral approach of the NPP.

The realization of this concept is a relevant element for plant safety as a precaution to prevent damage.
AGEING MANAGEMENT AND LONG TERM OPERATION OF NPP BORSSELE

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NPP Borssele, a PWR already 34 years in operation as the only NPP in operation in the Netherlands. Although designed for at least 40 years the long term perspective for Borssele has changed from 30 to 40 to 60 years in only a few years time.

In this respect not so much technical influences but far more political, personal and economical factors were very important to reach the positive long term outlook.

Of course safety is priority nr.1 as a prerequisite for Long Term Operation and at the moment activities are going on to be able to reach at least 60 years at a high safety level.

In this paper an overview is given of NPP Borssele from start of operation to the current status. In this overview important upgrades/projects are briefly shown. A short description of ageing management during operation is given but also the performed Ageing Management reviews including an IAEA-AMAT review in 2003. A slight insight is given on the political struggle and the final unique agreement between Dutch government and company on LTO of Borssele including the requirement to belong to the safest 25% of Western Light Water Reactors. At the end a short overview on the current LTO-activities is given.
MATERIALS AGING MANAGEMENT PROGRAMS AT NUCLEAR POWER PLANTS IN THE UNITED STATES

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Recent events regarding degradation of materials in nuclear power plants have been very costly to the industry and have implications for reliability, safety, and performance. As a result, industry executives implemented a Nuclear Energy Institute Materials Initiative (NEI 03-08) for proactive management of materials aging issues at all plants in the United States. This materials initiative, which began in May 2003, will integrate and coordinate the industry resources devoted to the materials issues. Additionally, it requires that each plant supports this effort through funding and implementation of the applicable guidance documents with “mandatory” and “needed” actions. An overarching Materials Degradation Management Program must be in place at all plants (effective August 2006) which coordinates the many individual subprograms affecting the reactor coolant system materials.

The key areas to be included are: Alloy 600 management, reactor vessel integrity, PWR vessel internals, boric acid corrosion control, steam generator management program, primary and secondary system water chemistry, fuel reliability and BWR vessels and internals program. Other programs such as NDE and in-service inspection, flow accelerated corrosion, fatigue management, and leak management may also be incorporated under the overarching program.

The success of the NEI 03-08 Materials Initiative will depend on how well each utility performs and how well these programs are adopted by the sites. Improvements in the overall performance should show up in the Program Health Reports and in the individual plant performance indicators such as fewer unplanned outages, and improved reliability and capacity factors. Any deviations from the “mandatory” or “needed” program elements will require documentation with a justification and senior management approval. These are not regulatory requirements, but are commitments made by senior executives under the NEI Nuclear Strategic Issues Advisory Committee (NSIAC). The Industry Issue Program groups, coordinated through the Electric Power Research Institute (EPRI), will monitor all deviations, and the Institute of Nuclear Power Operations (INPO) will audit these programs for compliance.

This paper reviews the status of the NEI 03-08 Materials Initiative and implementation of aging management programs in the United States including recent activities and general approaches for managing materials degradation issues.
AGING MANAGEMENT IN BOHUNICE NUCLEAR POWER PLANT

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Aging management begun to be systematically implemented under provisions of Slovak Electricity Company environment since 1996, when Safety Analysis Report after 10 years of Unit 3 & 4 operation had been submitted to the national regulatory authority (ÚJD SR). By then, effects of known degradation mechanisms were assessed through the use of existing particular programs; e.g. RPVs’ irradiation embrittlement by a standard design-built surveillance specimen program and using neutron dosimetry calculations, low-cycle fatigue of main primary components/pipelines hot spots applying computational analyses as well as evaluation of an erosion-corrosion effect on wall thinning of critical secondary piping components.

As the world’s trends and progress still have continued further towards the use of more sophisticated software, hardware, diagnostics, analyses, procedures and computing tools, given methodologies have been modernized and the idea of operational loads on evaluated systems, structures and components – particularly their critical hot spots - has been several times revised.

After issuing of country’s nuclear authority safety guide No.II.5.X/2001 ‘Aging management of nuclear power plants. Requirements.’ in 2001, the project called ‘Aging management and lifetime optimization of nuclear power plants with WWER 440 units’ had been introduced. In collaboration with the research institute (VUJE) it has been developed within 2002-2005.

Project’s particular tasks were dealing with:
- research of aging mechanisms and their analysis,
- aging management programs,
- research on advanced monitoring systems of degradation processes,
- legal terms of reference for AMPs,
- AM database,
- evaluation of the plant’s longterm operation conditions efficiency,
- legal terms for an approval on the NPP operation beyond design lifetime.

The project set a basis for an implementation of an AM system under provisions of Slovak Electricity Company environment to resolve the following issues:
- defining an effective AM system,
- listing critical SSCs including their known degradation mechanisms,
- fundamentals of AMPs establishment,
- principles of AMPs evaluation - per annum and after 10years.
REFERENCES


FIG. 1. Proposed steps within aging management of critical SSCs in Bohunice nuclear power plant according to [1]
APPLICATION OF LIFETIME MANAGEMENT FOR MECHANICAL SYSTEMS, STRUCTURES, AND COMPONENTS (SSC) IN NUCLEAR POWER PLANTS

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In most countries it has been stipulated that the licensing of nuclear power plants and their subsequent operation is based mainly on proof of the plant safety (e.g. strength analysis for operational conditions, postulated accidents, etc.). In Germany the atomic energy act [1] requires that “every necessary precaution has been taken in the light of existing scientific knowledge and technology to prevent damage resulting from construction and operation of the installation”. This has been realised in guidelines and in the nuclear standards [2,3,4] with their indications and requirements for plant safety. According to these documents it has to be ensured that:

- safety with respect to the quality of the systems, structures and components (SSC) is provided by the design, the material and the manufacture,
- the quality of the SSC has to be guaranteed and documented throughout the lifetime (extensive quality assurance during manufacture, construction, and operation),
- the operational parameters (damage mechanisms) relevant for the integrity of the SSC are monitored and
- operational experience is recorded continuously and safety related information is evaluated.

Therefore, the guidelines and standards contain all the requirements for a safe operation throughout the lifetime (lifetime management), for the control of ageing phenomena (ageing management) as well as for proof of integrity (e.g. with the aim to demonstrate break exclusion) for mechanical SSC. Within this field the ageing management is a key element.

The first step within this scope is to select and arrange the SSC and to assign these to group 1, 2 or 3. The classification is according to the requirements of the nuclear codes and standards (RSK-guidelines, KTA) and if necessary according to plant-specific and safety-related factors. The plant operator is responsible for the classification and an expert has to check it on the basis of the current codes, standards and the state-of-the-art, Fig. 1.

- Group 1: Failure of the SSC shall be excluded to avoid subsequent damage, e.g. reactor pressure vessel (RPV) and main coolant lines (MCL). The required quality shall be guaranteed for subsequent operation. The causes of possible in-service damage mechanisms shall be monitored and controlled (proof of integrity) [5]. Implementing this “proactive approach” prevents damage.
- Group 2: For redundant SSC the failure of a single part is allowable from a safety relevant point of view. However, common mode failure shall be excluded. The present quality shall be maintained for subsequent operation. The consequences of possible in-service damage mechanisms shall be monitored (preventive maintenance, time- or condition-oriented)
- Group 3: There are no defined standards for the quality of the SSC concerning subsequent operation (failure-oriented maintenance).

Depending on the safety-relevance of the SSC under observation including preventive maintenance various tasks are required in particular to clarify the mechanisms which contribute system-specifically to the da-mage of the compo-nents and systems and to define their control-ling parameters which have to be monitored and checked. Appropriate continuous or discontinuous mea-sures are to be considered in this connection. The approach to ensure a high standard of quality in operation and the management of the technical and organisational aspects are demonstrated and explained.

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REVIEW AND DEVELOPMENT OF AGING MANAGEMENT PROGRAMS OF THE MAIN COMPONENTS AT PAKS NPP

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According to the License Renewal related Hungarian Atomic Energy Authority’s Nuclear Safety Regulations and Guidelines a systematic review and development of the aging management programs of all the safety related equipment has been performed for Paks NPP. The AMP Review methodology is based on US NRC 10 attributes (Scope, Preventive actions, Parameters monitored or inspected, Detection of aging effects, Monitoring and trending, Acceptance criteria, Corrective actions, Confirmation processes, Administrative controls, Operating experience) taking into account the typical VVER-440 constructional, operational, QA and other administrative control features. The presentation deals with the AMP review of the main components: RPV, RPV Internals, Steam Generators, Pressurisers, MCPs, Main Gate Valves and the Primary piping. All known and potential degradation mechanism was taken into consideration. All the aging management related programs and activities (ISI, Maintenance, Chemical control, Technical Condition Monitoring, Operational data monitoring, Operational experience monitoring, QA e.g.) together with the related NPP’s internal procedures have been identified for the review. The results of the review ascertain the strengths and weaknesses of the previous programs and activities as related to the need to understand and manage the effects of aging of the main components. The methodology and the main results of the review as well as examples of typical recommendations for development of the current programs will be presented.
SNSA SURVEILLANCE OVER THE AGEING EFFECTS AND ABILITY 
FOR LONG TERM OPERATION AT THE KRŠKO NPP

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The paper presents the Slovenian Nuclear Safety Administration (SNSA) tools used for verification the adequacy of management of ageing effects and assuring suitability for long term operation at the Krško NPP. In addition to tools commonly used as PSR (Periodic Safety Review), assessment of plant modifications, regular inspections, the SNSA applies some special methods like monitoring the condition of important plant structures, systems and components (SSC) through special designed software, review and assessment of important plant programmes and its own set of performance indicators.

Until the new Act on Ionizing Radiation Protection and Nuclear Safety was accepted in 2002, there was no legal basis for conducting a PSR of nuclear installations. After the NPP Krško modernization project in 2000 (including steam generators replacement, reactor power increase), the need for conducting a PSR was clearly recognized by both the NPP management and the SNSA.

The first PSR programme was prepared in 2001 in accordance with the IAEA requirements and European practice. The main tasks of the PSR were a review of plant status for each safety factor, a development of ageing and life cycle management programme, a review of seismic design and PSHA analysis and an update of regulatory compliance program. The review confirmed that the plant is as safe as originally meant and that there are no SSCs that could limit the plant life in the next ten years. Nevertheless, it identified a number of recommendations to further enhance the plant safety.

According to new legislation the PSR is obligatory for all nuclear installations. It is a fundamental prerequisite for the extension of the NPP operation licence for the next ten-year period.

Regular SNSA inspections in the facility are performed about twice a week. More strict inspections are performed during plant outages when additional inspections are performed by the SNSA experts from the Nuclear Safety Department, who are focused on results of in-service inspection programme (ISI), corrosion/erosion monitoring programme (CEMS), surveillance tests and important modifications. Technical Support Organisations (TSO) are also engaged in supervising parts of maintenance and testing.

Significant progress has been done in the field of review and assessment of new plant documents. The SNSA reviewed important plant programmes such as Ageing Management Programme (AMP), Environment Qualification Programme (EQ) and Maintenance Rule Programme (MR). Findings were discussed with plant managers and most of findings were taken into account by the plant specialists.

A special way to control suitability for long term operation at the Krško NPP is a development project which the SNSA started in 2005, basically to improve knowledge in the
ageing area and to set up a list of potential regulatory body activities in this area. The project consists with the following topics:

1. Overview of regulatory requirements and practices from other European countries and USA,
2. Theoretical basis of ageing processes,
3. Review of the AMP at the Krško NPP,
4. Development of the SNSA procedure for supervision of ageing processes at the Krško NPP,
5. Development of the software for monitoring the condition of important SSC at the Krško NPP.

The software for monitoring the condition of important SSC and the SNSA procedure governing its use has been just completed. The provider of the software has included only 20 representative SSCs in the database that will be gradually extended by the SNSA. On the basis of the data from surveillance testing, in-service inspection and maintenance activities, the database provides trending, comparison with allowable values and alerts, review of corrective and preventive actions. Transients important for fatigue evaluation are included into the database as well. The SNSA procedure comprises safety criteria to include new components into the database, criteria to rank components by their importance to safety, inspection parameters to monitor level of degradation, acceptance criteria and guidelines for obtaining data from the NPP.

In 2005, the SNSA started to collect and sort corrective work orders issued by the plant. Tracking of work orders enables early warning of systems or equipment degradation as well as efficiency of maintenance and corrective program. Database can be systematically searched and screened for reoccurring malfunctions and failures of equipment. Safety performance indicator for corrective work orders is followed on monthly basis. Trending of work orders with regard to their safety significance is preformed and SNSA actions toward facilities are triggered in the case of undesirable trend or prompt increase of work orders issued.

The future SNSA goals regarding ageing management and long term operation are to:
- complete and accept new regulations with IAEA and WENRA requirements,
- establish a suitable way for obtaining data from the plant for the above-mentioned database,
- start adding new SSCs into the database.
PLANT LIFE MANAGEMENT MODELS WITH SPECIAL EMPHASIS TO THE INTEGRATION OF SAFETY WITH NON-SAFETY RELATED PROGRAMS

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Due to current social and economical framework, in last years many nuclear power plant owners started a program for the Long Term Operation (LTO)/PLIM (Plant Life Management) of their older nuclear facilities. PLIM/PLEX has already been implemented in many countries (USA, Russia, etc.). This process has many nuclear safety implications, other than strategic and political ones. The need for tailoring the available safety assessment tools to such applications has become urgent in recent years and triggered many research actions.

In particular, a PLIM framework requires both a detailed review of the features of the main safety programs (Maintenance, ISI, Surveillance) and a complete integration of these programs into the general management system of the plant.

New external factors, such as: large use of subcontractors, need for efficient management of spare parts, request for heavy plant refurbishment programs demand for updated techniques in the overall management of the plant. Therefore new organisational models have to be developed to appropriately support the PLIM framework, integrating both safety related and non safety related issues.

Last year a network of European Organisations operating Nuclear Power Plants, SENUF (Safety of European Nuclear Facilities), under the coordination of the JRC-IE (European Commission, Joint Research Center, Institute for Energy), carried out an extensive questionnaire on maintenance practice in their facilities aiming at capturing the aspects of the maintenance programs where research is mostly needed.

This paper uses some results of the questionnaire, which was not oriented to LTO/PLIM, to draw some conclusions on how the current maintenance programs could support a potential LTO/PLIM, among the other programs running at NPPs. In this sense, it is spin-off of the SENUF WG on maintenance.

The paper aims at identifying the technical attributes of the maintenance programs more directly affecting the decision for a long-term safe operation of a nuclear facility, the issues related to their implementation and safety review. The paper includes an analysis of the questionnaire circulated among the SENUF participants and a discussion on the implications of optimised maintenance programs in existing plants. Some examples at VVER plants taken by sources other than SENUF complete the overview, with some proposals for solution of practical implementation problems.

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REFERENCES


PLANT LIFE MANAGEMENT EXPERIENCE AT TARAPUR ATOMIC POWER STATION

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The twin BWR reactors of Tarapur Atomic Power Station (TAPS) are in their 38\textsuperscript{th} year of successful operation and have generated more than 71 billion units of electric power. The plant has seen continuous evolution based on operating experience, feed back from overseas reactors, lessons learnt from nuclear incidents, accidents and fresh review of design basis and safety analysis of the plant due to efforts of upgradation, renovation and refurbishment.

The Plant Life Management involved establishing an Ageing Management Programme (AMP). The AMP involved identification of key systems, structures and components (SSCs) that may experience degradation due to ageing, and take corrective measures through maintenance, repair and / or replacement. The identified components were classified as major critical components, important systems and other critical components. The components were further classified as not replaceable, difficult to replace and replaceable on routine basis. The various degradation mechanisms (Stress Corrosion Cracking (SCC), Intra Granular Stress Corrosion Cracking (IGSCC), Trans Granular Stress Corrosion Cracking (TGSCC), Erosion Corrosion (EC), Flow Accelerated Corrosion (FAC), Temperature, Pressure, Humidity, Radiation, etc.,) were identified for critical components, their method of detection, methodologies followed for In-Service inspection and developmental activities to assess the integrity of nuclear reactor vessels, piping & components for continued service. For each component mode of degradation was identified, ageing assessment was done and action plan was finalized. A comprehensive examination was carried out on Structures, Systems and Components (SSCs) as part of plant ageing management programme.

The major critical components of TAPS are the Reactor Pressure Vessels (RPVs) and the Containment structures. The RPVs are designed for 40 effective full power year (EFPY) of operation and till date they have operated for less than 21 EFYP. The studies and assessment of material condition of the RPVs was done by testing surveillance coupons subjected to accelerated ageing. The test results conclude that there has been no deterioration in the mechanical properties of the RPV material during the operating life experienced, and the RPVs are fit for service for several more years. The condition of the containment and main plant buildings was assessed. Visual inspection, crack mapping, non-destructive test, chemical test, petrography etc were the tools employed for condition survey of Main plant building at TAPS. Based on the findings detailed repair and protective coating application on main plant building was done and they are now fit for long term service.

Important Systems (engineered safeguard system and other important support systems) were assessed to be in good condition. Replacement of some important equipment like Salt Service Water (SSW) pumps, Control rod drive (CRD) pumps, Emergency Condenser tube bundles, Station battery was done on the basis of condition monitoring.
The detailed review of the plant included the PSA studies, aging degradations of structures, systems and components, operating experiences, seismic studies and design review based upon the US NRC’s General Design Criteria, codes and guides. The reviews were carried out by expert groups and completed in about three years. The modifications implemented on basis of the review were mainly about the change in equipment layout and unit wise segregation of electrical and mechanical systems and replacement of the 3 x 50% capacity emergency diesel generators with 3 x 100% capacity diesel generators (to obviate common cause failure and to enhance the system reliability) apart from aging related inspections / replacements and seismic up grades. Samples of Safety related cables were subjected to residual life assessment (RLA) and Polarization Index (PI) checks and replacement action firmed up on the basis of the RLA findings. The shared systems such as control rod drive hydraulic system, reactor shut down cooling system and power supply to the neutron monitoring system have been segregated unit wise. About sixty new equipment and panels have been added and fifteen were relocated. To segregate the power and control circuits, two thousand meters new cable trays were installed and fifty thousand meters new cables had to be laid. About ten thousands meters cable was re-routed from the old distribution panels to the new distribution panels. A significant number of equipments, piping, electrical panels have been replaced as part of continuous plant life management & upgradation to meet current standards.

Schedules were prepared for each activity, reviewed and finalized. Multi disciplinary self sufficient Task Forces were appointed for completing all prerequisites before shutting down the reactors. Daily review of work by Station management, Additional resource mobilization as required, Design and engineering assistance availability on round the clock basis, detailed erection and commissioning procedures , regular inspections and QA checks were an integral part of the Plant Life Management related up gradations.

The preparation of revised schemes, design basis reports and operating procedures for the upgraded systems was part of the parallel activities.

Training of O&M personnel on the upgraded systems were planned and executed in parallel so that trained staff was available to operate and maintain the upgraded systems and new equipments immediately on their commissioning.

The Ageing Management Programme of the safety related components is in place to ensure that integrity and functional capability are maintained at par with current standards of safety throughout the service life of the plant. Programme of surveillance, repair and replacement exist to ensure that deterioration of SSCs with time does not reduce level of safety.

The stable performance of the plant stability in operation after safety upgradations indicates its capability to operate for several more years.
Increasing electricity demand due to population growth and redistribution, high oil and gas prices, concerns for greenhouse gas emissions, and a positive trend in public opinion and government support for nuclear power provide tremendous opportunity for growth in the nuclear industry throughout the world. This can be accomplished in two ways, (1) new plants can be built and (2) the performance of existing plants can be improved through increased reliability and increased in generation capacity known as power uprates. In addition, the operating and design life of existing units can be extended for twenty or even forty years through plant life extension (PLEX). However, these uprates and life extensions are only viable if the plant reliability and capacity factor gains made in recent years continue through the extended operating domain/period.

A growing number of US and international BWR’s have successfully improved economic viability while increasing generation within existing facilities through implementation of uprates and PLEX efforts. The evolution of these efforts is the transition to a synergistic approach of plant modernization (including digital I&C upgrades), Life Cycle Management (LCM), margin recapture, and reliability improvement included in the overall plan with power uprates and PLEX. As the experience base with these programs grows suppliers, like GE, and utilities continue to build on the experience of prior projects, improve project execution, and maximize the investment returns.

This paper will present the results of several recent GE BWR projects in the US that have implemented combined efforts of reliability management programs with strategic projects such as power uprates. The paper will focus on the lessons learned from these efforts to help plants prepare for planning and implementing their own integrated reliability programs. Specific areas to be discussed include project initiation and scoping, project planning, project team requirements, execution and results. Recommendations will be provided for utilities to enable them to commence their own projects or improve ongoing projects. Original power uprates, referred to as stretch power uprates (SPU) were mostly design bases documentation changes and involved minor equipment/set points modifications. Plants took advantage of inherent margins in systems and components (especially Balance of Plant Systems) to gain an extra five percent in generation. The original extended power uprates (EPU) in the 1990s were approached in the same manner. Only absolutely necessary modifications and upgrades were implemented to meet safety, licensing, and generation requirements. More recent EPU projects took the approach of combining the capital investment in the unit with a proactive approach to margin improvement, long term operation and maintenance, and plant life extension considerations.
AGEING MANAGEMENT PROGRAM TO REACTOR PRESSURE VESSEL
INTERNALS COMPONENTS IN A BWR NUCLEAR POWER PLANT

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Mexico has two identical units (GE) BWR 5 type reactor and nominal power is 682 MWe. Unit 1 and unit 2 have being operated since 1990 and 1995 respectively. The original licensed reactor thermal power was 1931 MWT and power was uprated to 2027 MWth in 1999. A new Extension Power Uprate (EPU) is planed for 2010 and finally the licensed expired in 2020 for U1, and in 2025 for U2, because it is for 30 years. In 2005 a Technical cooperation project was approved by the IAEA with the objective to prepare the Plant Life Management (PLiM) program for the integration of ageing and economic planning. In 2007 it was agreed to extend the on-going project focus in License extension of these BWR’s.

After that, the EPRI founded the BWR-Vessel Internal Project and the Utility obtained a membership and access to the BWR-VIP documents. In 1999, the ININ realized the first analyses of susceptibility to IGSCC of the Shroud of Unit 1, based in specific information and in BWR-VIP 76, and reprogrammed the IVVI program. Since this year the ININ had realized the analyses of susceptibility to IGSCC and the IVVI program for the principals RPVI’s components.

One of the pilots programs chose in 2005 to apply the Ageing Management Program methodologies was the Shroud and in the end of 2006, the AMP program was extended to all the Reactor Pressure Vessel Internals (RPVI’s) components in U1 and U2.

The main RPVI’s components were studied by groups: Shroud (including horizontal H1 to H7 and Verticals welds), core spray internal Piping, Core Spray Sparger, core plate, top guide, In core Housing-CRD housing-CRD Guide tube-Dry tubes, Jet Pumps, LPCI Coupling, Access Hole covers, Weld internals attachments including Surveillance Capsule Holder, Steam separator, Steam Dryer, Support Plate, Access Hole Covers and Instrumentation DP and SLC. The RPVI’s Safety function was take account, and a specification of the EPRI’s BWR guidelines used in every case and the programs for 2007 that finished with all the RPVI’s of both units were showed.
<table>
<thead>
<tr>
<th>Internal Components</th>
<th>Safety function</th>
<th>Material</th>
<th>BWR VIP Associated document.</th>
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<tbody>
<tr>
<td>Shroud</td>
<td>Yes</td>
<td>304L welds 308L Support ring alloy 600</td>
<td>76</td>
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<td>core spray internal Piping</td>
<td>Yes</td>
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<td>Yes</td>
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<td>core plate</td>
<td>Yes</td>
<td>304 welds 308 and 308L</td>
<td>25</td>
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<tr>
<td>top guide</td>
<td>Yes</td>
<td>304, 3304L welds 308L</td>
<td>26</td>
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<tr>
<td>In core Housing-CRD housing-CRD Guide tube-Dry tubes</td>
<td>Yes</td>
<td>Several</td>
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<td>Jet Pumps</td>
<td>Yes</td>
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<tr>
<td>LPCI Coupling</td>
<td>Yes</td>
<td>304 welds 308L</td>
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<td>Weld internals attachments</td>
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<td>Yes</td>
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<td>Yes</td>
<td>Several</td>
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*Programed in 2007  + Only in Unit  (1) not in Unit 2
The general description of the methodologies applied to the Internal Reactor Pressure Vessel components, which were designated like a pilot program within the scope of the Ageing Management Programs (AMP) and PLiM Program were discussed and they are show in Figure 1.

A specific example of Shrouds AMP was showed, including Water chemistry data, ECP calculus by CHECKWORK VIA software, the Inspection Program for the U1 Shroud, including changes did by Structural Integrity calculus and the Mitigation program of Hydrogen Water Chemistry and ECP monitoring in this unit.

Some discuss is made about the use of a conservative crack growth rate of 1 E-5 inch/hrs in Stainless steel, that produce very restricted time inspection periods. For example for H3 weld of shroud in U1, based in inspections data, has a longitudinal growth rate in 3 cycles of 3.997E-5 in/hr and a depth growth rate of 3.66E-6 inc/hr, and the Fracture Mechanical calculus was do with a value of 5E-5 in/hr in the longitudinal direction and the crack was consider through wall. That is a conservative assumption, and may be with the correct crack growth rate, we could obtain a better period of time to inspect this weld.

New research activities related to the RPVI’s is commented. The ININ is correlated the influence of fluence peaks on cracking, and the case of the new water chemistry (NMCA) the characterization of oxides deposits with Nobel Metal Chemical Additions +Hydrogen Water Chemistry in laboratories.
REFERENCES


AGING MANAGEMENT FOR TEPCO’S BWR REACTOR INTERNALS

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1. Experience of damage to reactor internals:

Tokyo Electric Power Company (TEPCO) has 17 BWR power plants, and has experienced damage to reactor internals at some of the plants, brought about by stress corrosion cracking (SCC) or fatigue. Typical examples include SCC of shrouds, CS sparger T-BOXs and ICM housings and fatigue damage to steam dryer drain channels, jet pump sensing lines. In these cases, damaged components were either repaired by installing tie rods and clamps or replaced.

2. Current condition of reactor internals:

a) Reactor internals

To prevent SCC of reactor internals, SUS316L is used for reactor internals at TEPCO’s power plants. Although SUS304 is still used for some components such as jet pumps at Fukushima-Daiichi Nuclear Power Plant Nos. 4 and 6, the material has been replaced with SUS316L on a scheduled basis, including shroud replacement.

At Fukushima-Daiichi Nuclear Power Plant Nos. 1, 2, 3 and 5, and at Kashiwazaki Kariwa Nuclear Power Plant Nos. 6 and 7, where the shrouds have been replaced, weld lines were reduced by using forged stainless steel, and stress measurements improved by N-stripping.

For joints in shrouds H4 and H7 that were evaluated as susceptible to crack growth once SCC occurs, stress improvement by peening was performed on a scheduled basis. In addition, at some plants, environmental mitigation by hydrogen injection and NMCA is in place.

b) Nickel-base alloy welds

At Fukushima-Daiichi Nos. 1 to 6, Fukushima-Daini Nos. 1 to 4, and Kashiwazaki Kariwa No. 1, SCC-sensitive material INCONEL 182 is used at nickel-base alloy welds of the shroud support.

At Fukushima-Daiichi Nos. 1, 2, 3 and 5, stress improvement by peening was performed on some shroud joints when the shroud was replaced. Since the CRD stubs at Fukushima-Daiichi No. 6, Fukushima-Daini Nos. 1 and 3, and Kashiwazaki Kariwa No. 1 were evaluated to have high potential for SCC due to the manufacturing process, stress improvement by peening was performed.

c) Jet pump sensing line

Clamps have been installed to the sensing line at Fukushima-Daiichi No.6, Fukushima-Daini Nos. 1, 2 and 3, and Kashiwazaki Kariwa Nos. 1, 2 and 5 that may resonate with
pressure pulses from the PLR pump during operation, in order to reduce stress and prevent resonance.

3. Inspection requirements for reactor internals

(a) Reactor internals

For shrouds, an MVT-1 inspection is required every 10 years by the NISA, and for jet pumps and CS pipes, an MVT-1 inspection is required within 25 years of actual operation under JSME Standards (Maintenance Standard). These components are relatively accessible, allowing visual inspection with an underwater TV camera.

(b) Nickel-base alloy welds

For nickel-base alloy welds of shroud supports and CRD stubs, an MVT-1 inspection is required within 25 years of actual operation under JSME Standards (Maintenance Standard).

These components are not easily accessible for inspection, requiring removal of the CR guide tube and the use of remote control devices, such as robots, to perform the inspection.

For H09 and H11, which are welded to the RPV, measurement of the depth of SCC is important to ensure the integrity of the RPV.

(c) Jet pump sensing line

The jet pump sensing line is not safety-significant and does not require inspection. However, an underwater visual inspection is performed during a jet pump inspection.

4. Further action

(a) Reactor internals

Shrouds can be continued to be used under the Maintenance Standard or can be repaired even if cracks are found. The material has been replaced with the SCC-resistant material and requires no further action.

However, jet pumps and CS pipes are difficult to replace. In addition, it takes about 14 days to inspect all 20 pumps, and the inspection of the jet pumps has a significant impact on the inspection process. Therefore, an efficient inspection method needs to be established.

(b) Nickel-base alloy welds

When the shroud was replaced at Fukushima Daiichi Nos. 1, 2, 3 and 5, reactor internals such as jet pumps were replaced and the shroud support was inspected. Repair welding was performed on the cracks found in the inspection.
The number of plants that will have been operating for more than 25 years will increase. At these plants, it will be necessary to inspect nickel-base alloy welds underwater, such as in the shroud support, and it will be important to establish an efficient inspection method, as well as a method to evaluate the integrity of welds when SCC is found.

Based on our experience in inspecting and repairing shrouds, remote inspection systems are being developed. The applicability of these systems to actual plants must still be evaluated. It is also necessary to evaluate the method of examining the integrity of reactor pressure vessels by UT inspection from outside the vessel, which has been used in the U.S.

(c) Jet pump sensing line

Clamps have been installed to prevent resonance with pressure pulses from the PLR pumps during operation. It will be necessary to evaluate resonance at startup and shutdown and then to study the implementation of resonance prevention measures, such as installing clamps.

(d) Health monitoring

Environmental mitigation by hydrogen injection and NMCA and measures to prevent vibration of the jet pump sensing line by installing clamps are in place. However, no quantitative evaluation has been performed at actual plants on the effectiveness and adequacy of scope of these measures. Therefore, it is necessary to perform corrosion potential and vibration measurements inside the reactor and to plan the development of technology to monitor the effectiveness of such measures.
AGEING MANAGEMENT, IN SERVICE INSPECTION AND EXCEPTIONAL MAINTENANCE

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In this paper, it is drawn up a panorama of the actions implemented by EDF in the field of maintenance in order to control the ageing of Nuclear Power Plants Structures, Systems and Components and to ensure, thus, an operation of its NPPs in compliance with safety requirements.

Routine maintenance, exceptional maintenance, inspections in service associated with an efficient monitoring and a thorough analysis with the experience feedback, contribute to this control and has to be adapted to each component, taking into account the ageing phenomena brought into play.

EDF PWR fleet consists of 58 Units for an installed capacity of 61 GWe; at the end of 2006, this fleet has an average age of approximately 22 years, the youngest unit 7 years and the oldest 29 years.

EDF maintenance policy is based on a routine maintenance to maintain the functional capacities of equipments and thus the availability of the units, in the respect of safety requirements.

Since 1995, the routine preventive maintenance and the associated surveillance are piloted and optimized compared to the stakes in safety and in competitiveness. After a first phase of optimization using RCM analysis, EDF develops for several years conditional preventive maintenance, leaning on strengthened diagnosis/prognosis and conditional preventive maintenance on a sampling basis, taking benefit of the standardized fleet of units.

In every case, the definition of this routine optimized preventive maintenance requires the knowledge of the functional consequences of the failures and the utilization of the experience feedback, taking into account lab examination of used components and validation of non destructive examination methods.

This routine maintenance results in various actions such as adjustments, replacements of wear parts, functional or equipment modifications, repairs, ... But to face certain ageing phenomena occurring in a irreversible way in the course of time, it is necessary to complete it by an exceptional preventive maintenance of bigger scale.

Exceptional maintenance consists of operations carried out only once (it is the objective) during the lifespan of the units. They are implemented to control proven or potential degradations, in anticipation, concern a significant part of the fleet and present important industrial stakes (costs, dosimetry, resources,…); they are thus planned on the fleet leve.

It relates primarily to the large components: steam generators, turbines, condensers, control systems, civil works,… and requires:
- a capacity of degradation forecast and repairability/replaceability anticipation,
- a capacity of decision and programming economically relevant in relation to the technical end of units lifetime and the possible increase of their performances; it is important to have a visibility for the next 20 to 30 years for the choices of large investments to be studied.

To illustrate this policy, several examples implemented by EDF are described:

- Reactor Vessel Heads replacement strategy,
- I&C systems long term maintenance strategy,
- Generator Stators refurbishment strategy.
IN-SERVICE INSPECTION OF CRITICAL COMPONENTS AS A KEY TOOL FOR THE LIFE MANAGEMENT PROGRAMMES

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The standard design life of a nuclear power plant is 30-40 years, however, based on the accumulated experience and multiple assessments carried out on the utilities, it is very likely that many plants will request to the authorities, and will be able, to operate in excess of their design lives. One of the key elements to assess the life of the plant, to apply for life extension, is to determine the integrity of structural materials more critically subjected to ageing degradation. One of the most powerful and comprehensive approaches is the in-service inspection.

Safety codes require periodic inspections either prescriptive or risk based informed of critical components. These components are those subjected to the most stringent environmental conditions such as radiation, corrosion, high temperature and pressure. In addition to these facts, the areas to be assessed have a limited and difficult access, the materials present complex geometries and the most critical parts are the welds and the heat affected zones.

To answer the requirements imposed by these limitations, in-service inspection systems would be designed with capabilities such as powerful manipulators with several degrees of freedom able to reach locations with limited access and robust enough to cope with high radiation doses, miniaturised multifunctional transducers as phased arrays to minimise interferences scanning the whole inspection volume generating abundant information, real time multi-channel data acquisition system, data analysis system graphics based, and qualified techniques that perform with high level of reliability.

There are solutions for harmonisation of regulatory requirements regarding in-service inspection of plants of different designs. The Spanish Nuclear Sector, including utilities, regulators, and service providers, offers a good example about the harmonisation of plants of PWR (from Westinghouse and Siemens) and BWR designs and ASME and KTA Codes. This experience could be valid for other countries with nuclear plants of different designs.

In this paper will be described the approach followed for in-service inspection of critical components, firstly, with standard equipment and, secondly, with advanced systems. Then, based on the experience gained on examinations carried out on different reactor types of more than 30 countries, specific examples of actual inspections implemented on critical components such as reactor pressure vessels (RPV), RPV head penetrations, and internals will be described.
ESTABLISHING A NEW ISI STRATEGY FOR PAKS NPP

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The owner of Paks NPP, Hungary, is aiming at reviewing and adjusting the plant’s ISI program to meet the ASME Code Section XI requirements. At the same time the IST requirements of the ASME OM Code are planned to be adapted, too. ISI and IST in conformance with ASME Code requirements will provide an opportunity to compare these activities with world wide acceptable safety requirements, which will help to reach the consent across Europe for Paks NPP’s operational life extension project. Meeting the Section XI requirements will allow the plant owner to extend the current four-year inspection interval up to an eight-year based ISI program, which will contribute to a more cost-efficient operation and maintenance. Establishment of a new ISI strategy is being done gradually. As a first step, studies to compare the current ISI program, mainly based on Russian normative documents, and the ASME Code Section XI based one were developed, and items needing special care were identified. These comparative studies justified the feasibility of the project. Based on this, the ISI program modification was accordingly modified. Of course, the existing NDE procedures were also transformed taking into consideration Section V requirements. Details of this part of the project were reported elsewhere [1]. Since prerequisite for application of Section XI is the meeting with Section III requirements, a check and, if needed, a re-design of selected components is being done to comply with the Section III requirements.

Meeting the ASME Code requirement is not limited to Sections III, V and XI as well as the OM Code, however, a complex assessment of the entire ASME system is necessary. The complex assessment includes the aspects of each Sections including Regulatory Guides, ASTM and ANSI standards referred to, and the legal background in both the US (10 CFR 50) and Hungary (Nuclear Safety Regulations). The concept of the ASME Code presumes that components inspected and tested according to Section XI and OM Code, were originally designed according to other relevant Code sections. It does not exclude, however, that ISI and IST requirements of the ASME Code can be applicable for components designed by another system (e.g. by Russian codes and standards).

As for the Hungarian rules, they do not determine explicitly the applicable codes neither for the design nor for the ISI. The only statement is that codes and standards must be authoritative, and the ASME code obviously meets this criterion.
ONSITE INSPECTION EXPERIENCE OF ELECTRIC EQUIPMENT IN LICENSE PROCESS OF THE CONTINUED OPERATION OF KORI UNIT ONE

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The methods of equipment qualification for the continued operation of Kori unit one are qualification by type test report, by analysis, by operating experience and the combined qualification. The original licensed document did not include the Environmental Qualification Reports due to either a turn-key based project or early 70’s requirements.

Total 1,537 items are additionally performed for license of the continued operation with following actions: Replacement (REPLACE), Analysis (ANAL), Environmental Qualification Reports (EQR), Partial Type Test (TEST), Relocation to mild environmental condition area (RELOCATE).

<table>
<thead>
<tr>
<th></th>
<th>REPLACE</th>
<th>ANAL</th>
<th>EQR</th>
<th>TEST</th>
<th>RELOCATE</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of items</td>
<td>733</td>
<td>462</td>
<td>331</td>
<td>12</td>
<td>9</td>
<td>1,537</td>
</tr>
<tr>
<td>Percentile of items</td>
<td>47.7</td>
<td>30.1</td>
<td>21.5</td>
<td>0.8</td>
<td>0.6</td>
<td>100</td>
</tr>
</tbody>
</table>

The time-line of works are as shown below as dated October 2006. Most of replacement and relocation will be completed by August 2007. It should be required to finish the regulatory review within 18 months after submission on June 2006.

<table>
<thead>
<tr>
<th></th>
<th>REPLACE</th>
<th>ANAL</th>
<th>EQR</th>
<th>TEST</th>
<th>RELOCATE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Dec. 2006</td>
<td>0%</td>
<td>100%</td>
<td>95%</td>
<td>100%</td>
<td>0%</td>
</tr>
<tr>
<td>Aug. 2007</td>
<td>100%</td>
<td>0%</td>
<td>5%</td>
<td>0%</td>
<td>100%</td>
</tr>
</tbody>
</table>

Submerged condition of Class 1E equipments are determined by floor level on these location. The 128 items of that condition will be performed by EQR (8 items), by replacement (110 items), and by an analysis (10 items). If the works complete, safety functions of each Class 1E equipment would not affected by flooding and there would be hardly need to demonstrate the equipment operability during submergence.

Main review concerns to date could be summed up:

- An aging management program regarding non-EQ electric cables and connectors
- Time limited aging analysis for inaccessible underground 4.16 Class 1E cables having 30 year qualified lifetime.
Condition monitoring program for the harsh environment particularly at hot temperature zone.

As an operating experience approach qualified the equipment for normal environments, an additional material degradation analysis is performed to qualify the equipment for DBE. Partial type tests on vital components, such as MOVs, are being provided in support of the operating experience method, in addition to PVC cables even though passive and long-lived items.

The inspection of the continued operation will be performed in three stages. The first onsite inspection completed in January 2006. The inspection activities in the field of electric equipment installed are (1) scoping and screening of submitted/ supporting documents, (2) walk-down of electric systems, and (3) reviewing the onsite procedure of AMP etc.

The key components found for additional review are such as (1) metal enclosed bus-ducts and fuse holders of AC power source for control rod drive mechanism, down stream from reactor trip breakers and (2) inaccessible underground 4.16 kV cables both AAC power source and Class IE from 154kV offsite power.

The interim remedial proposals for the issues above are (1) one through inspection for passive and long lived bus duct and (2) addition of procedures for fuse holders of feeder disconnecting switches.

The raceway with cables of AAC was run recently and the performance test of AAC power source has finished in March 2007. The aging would scarcely be concerned with new installed cables in 10 years if the properly monitored during rainy season. Such interim actions have been reviewed.

The following onsite inspections, before the continued operation of Kori unit one, will be performed and would find the unexpected problems during replacement work of obsoleted 733 equipments, and relocation work of 9 equipments to a mild environmental condition area. With respect to the integrated plant assessment, an improved performance of emergency diesel generator with replacement of key components, such as governor and AVR, will be inspected according to the current licensing basis standards.

REFERENCES

[3] Licensing Submissions by KHNP for Kori Unit One: Lists of EQ Equipments and Cables; Environmental Qualification Reports; UFSAR and procedures of AMP
EC JRO NETWORK ON USE OF PSA FOR EVALUATION OF AGING EFFECTS TO THE SAFETY OF ENERGY FACILITIES, ACTIVITIES, AND RESULTS

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This paper summarizes some results of research studies and followed discussions on the use of PSA for evaluation of SSC aging effect to the overall plant safety carried out in the frame of EC JRC Network activities on Incorporating Ageing Effects into Probabilistic Safety Assessment (Aging PSA) [1].

The basic concern to use the PSA for aging evaluation is coming from the requirement to accomplish the safety goals during the whole lifecycle of the nuclear installation (including the extended lifetime). In probabilistic terms, INSAG-12 [2] specifies a safety goal as follow: “The target for existing nuclear power plants consistent with the technical safety objective is a frequency of occurrence of severe core damage that is below about $10^{-4}$ events per plant operating year. Severe accident management and mitigation measures could reduce by a factor of at least ten the probability of large off-site releases requiring short term off-site response.”

So, for the units which approach to the end of initial design lifetime and especially for those which planned to extend the lifetime, it has to be demonstrated that the plant safety level, at least, will be remain in accordance with this target till the end of operation.

Additional motivation to use the PSA for aging assessment proceeds from the significant limitations of deterministic approach and needs to prioritize the Aging Management or Life Extension actions to maintain established safety goals.

Objectives of the Ageing PSA Network are:

- to identify the way how the PSA could be applied for aging related activities as NPP’s Periodical Safety Examination, Aging Management, Maintenance Optimization and Lifetime Extension programs,
- to show what PSA tasks have to be addressed and performed to take into account aging effects and to demonstrate their impact to the overall plant safety,
- define the new approaches needed and priority for their development and demonstration in the frame of Ageing PSA Network.

Presently, aging evaluation related activities are realized or in the way of realization in the frame of the following programs:

- Periodical Safety Review,
- Aging Management,
- Maintenance Optimization,
- Lifetime Extension.
There are number of standards and guidelines available on the national and international level [3], but all of them in general based on the deterministic approach.

Only few elements concerning the possible application of PSA for aging evaluation were found in IAEA TECDOC 1511 [4].

Two main directions of PSA application could be identified:

- PSA as a tool to demonstrate and to monitor current level of safety,
- PSA as a predictive evaluation of risk.

PSA as a demonstration of current safety level will require:

- analyze aging trends for safety important components,
- identify components susceptible for aging, but not modeled in PSA,
- update reliability parameters (l=const) for PSA components with recent operating experience and input them into PSA,
- estimate reliability parameters and introduce them into PSA model,
- perform sensitivity analysis of aging issues.

PSA for predictive extrapolation the following steps could be done:

- review of PSA scope and assumptions,
- review of initiating events,
- analyze aging trends for safety important components,
- identify components susceptible for aging, but not modeled in PSA,
- elaborate age-dependent reliability models (statistic and physic),
- introduce them into PSA (consider, where necessary, test and maintenance effectiveness and periodicity),
- perform time dependent quantifications and sensitivity analysis.

The paper will present a brief summary of EC JRC Aging PSA network activities and futures actions planned, as well as some results obtained from feasibility studies and benchmark exercises.
REFERENCES


THE PROPOSAL EVALUATION APPROACH OF THE RISK INFORMED-INSERVICE INSPECTION AND THE RESULT OF TRIAL EVALUATION

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Nearly a decade has passed since risk evaluation in form of risk informed – inservice inspection (RI-ISI) was introduced for maintenance of plants in U.S and Europe. In such a situation, if Japanese plants employ RI-ISI in future, an evaluation approach applicable to Japanese plants is desired to be applied. Some approaches have been already considered as a RI-ISI evaluation approach, and each has strong points and demerits. Application of existing approaches to the Japanese plants without modifications is considered to be difficult since there are some differences between U.S and European plants and Japanese plants with respect to their degradation mechanisms, piping materials and circumferential environmental conditions, etc. Therefore, the RI-ISI evaluation approach was studied for domestic plants by making most of the advantages of the existing approaches in consideration of Japanese plant conditions, and implemented for representative systems as a trail.

The RI-ISI evaluation approach consists of five steps in this proposal:

Step 1- Evaluation of affection and segment classification (large classification)
The range, which the affection to the plant by fracture becomes the same on a piping isometric drawing, is classified as a segment, and the affection category by fracture of each segment is evaluated.

Step 2- Possibility evaluation of failure and a segment classification (small classification)
The range in which is anxious about degradation possibility on a piping isometric drawing is established, and, moreover, the segment, which is classified in Step 1, is classified in the range with the same degradation possibility. Degradation possibility evaluation of each segment also considers into consideration a piping material and an environmental condition, and is considered as five steps of evaluation. (in Table-1)

Step 3- Classification of a risk category
The combination of the affection category and the failure possibility category evaluated in Step 1 and 2 estimates a risk category, and the inspection proportion demanded is decided. Since it is thought that the part estimated by risk evaluation as “High” has a high failure concern degree, and is a critical area as which the maintenance of repair, exchange, etc. is demanded, an inspection demand proportion is established to 100%. (in Table 2)

Step 4- Establishment of an inspection part and the inspection range
According to the inspection demand proportion, an inspection part is specified from each segment. In this step the inspection part, the inspection volume, and the inspection approach according to the degradation mechanism assumed are established.
Step 5- Evaluation of a risk change amount

According to the introduction of the RI-ISI evaluation approach, it estimates a risk change amount of the plant.

It performed trial evaluation in RCS system and CVCS system with the above procedure. As a result of evaluating, as compared with present in-service inspection (ISI), inspection parts could be reduced about 45%, and it has confirmed that the risk to a plant could also be reduced.

Table 1 Likelihood of degradation classification for piping

<table>
<thead>
<tr>
<th>Category</th>
<th>Condition</th>
<th>Specific area (eg.)</th>
<th>Response</th>
</tr>
</thead>
<tbody>
<tr>
<td>I+</td>
<td>There is a possibility for rupture.</td>
<td>FAC</td>
<td>Repair/replacement, operational change, etc. are desired to perform as maintenance.</td>
</tr>
<tr>
<td>I</td>
<td>Since it may not result in rupture, however there is a possibility for leakage.</td>
<td>O₂SCC(SUS304), SCC from outer surface (Unidentified area)</td>
<td>Integrity is checked by performing periodic inspection.</td>
</tr>
<tr>
<td>II</td>
<td>There is a possibility for crack (degradation) occurrence.</td>
<td>Valve sheet leak type thermal stratification, thermal fluctuation (RHR cooler)</td>
<td></td>
</tr>
<tr>
<td>III</td>
<td>There is a slight possibility for degradation.</td>
<td>O₂SCC(SUS316), thermal fluctuation (MCP charging nozzle)</td>
<td>Areas arbitrary inspected by the present ISI.</td>
</tr>
<tr>
<td>-</td>
<td>There is no possibility for degradation.</td>
<td>No possibility for degradation, SCC from outer surface (identified)</td>
<td></td>
</tr>
</tbody>
</table>

Table 2 Inspection requirements for RI-ISI procedure

<table>
<thead>
<tr>
<th>Failure potential category</th>
<th>Impact assessment</th>
<th>Impact assessment</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Low</td>
<td>Medium</td>
</tr>
<tr>
<td>+ *</td>
<td>0%</td>
<td>5%</td>
</tr>
<tr>
<td>*</td>
<td>%</td>
<td>0%</td>
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<td>1</td>
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<tr>
<td>1 1 1</td>
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</tbody>
</table>
APPLICATION OF DYNAMIC SYSTEM RELIABILITY METHODS FOR INCORPORATION OF AGE-DEPENDANT RELIABILITY PARAMETERS AND DATA INTO THE PSA MODEL

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One of the tasks of Ageing Probabilistic Safety Analysis (APSA) is an incorporation of age-dependent reliability parameters and data of certain safety-related System, Structure and Component (SSC) in to the PSA model and interpretation of its results. The aging process could take place as a gradual degradation or improvement of characteristics of materials. Usually the improvement of one characteristic is accompanied by degradation of other characteristics. Sometimes a forced ageing of materials is applied for improvement and stabilization of certain characteristics. The aging can affect the systems and structures only through their components. Consequently the SSC availabilities may decrease due to the aging of components. The following categories of components could be identified:

1. **Unrepairable or irreplaceable components** (Category 1)

The Category 1 components are those generally considered irreplaceable (e.g. RPV - reactor pressure vessel). They have binary (2 possible) states – normal (success) and failed (failure) but it is possible to take into account some partial restoration by unsteady change of the failure intensity at a given moment (RPV annealing) – restorable-irreplaceable components.

2. **Repairable components**

   **Hard-to-replaceable (replaceable but costly) components** (Category 2). The Category 2 components are replaceable but costly (e.g. SG – steam generator). They have 3 possible states success, outage/capital repair and failure. The partial restoration is also possible to include similar to Category 1 components (restorable-hard-to-replaceable components).

   **Replaceable on a routine basis** (Category 3). The Category 3 and other components are the ‘key’ ones in terms of safety and reliability but susceptible to aging. They are replaceable on a routine basis or according to specific reconstruction/modernization program. If they will be replaced two or more times then they will have two or more aging states.

The plant availability and safety may decrease due to the aging of unrepairable and repairable components. The resulting increase in the overall plant unavailability and risk could be reduced by different maintenance measures: replacements and upgrading of replaceable and restorable components during repairs or changing of surveillance intervals of repairable components. The simplest way to quantify PSA models consists to use the hypotheses that SSC failure and repair rates are constant. However, all these measures are implemented in time and consequently they are age-dependent. That is why the dynamic aspects of aging and decision-making in NPP lifetime management becomes increasingly notable and more advanced tools are needed for their analysis. On the Figure 1 the state transition diagram of gradually aging component with multiple failure modes is shown.
This state transition diagram could be defined by two matrices and column: matrix of failure intensities \( \Lambda(i,j) \), matrix of repair intensities \( M(i,j) \) and column of replacement intensity \( -\mu_i \).

\[
\Lambda \equiv \{\lambda_{i,j}\} \equiv \begin{pmatrix}
\lambda_{11} & \lambda_{12} & \ldots & \lambda_{1m} \\
\lambda_{21} & \lambda_{22} & \ldots & \lambda_{2m} \\
\vdots & \vdots & \ddots & \vdots \\
\lambda_{n1} & \lambda_{n2} & \ldots & \lambda_{nm}
\end{pmatrix}
\]

\[
M \equiv \{\mu_{i,j}\} \equiv \begin{pmatrix}
\mu_{11} & \mu_{12} & \ldots & \mu_{1m} \\
\mu_{21} & \mu_{22} & \ldots & \mu_{2m} \\
\vdots & \vdots & \ddots & \vdots \\
\mu_{n1} & \mu_{n2} & \ldots & \mu_{nm}
\end{pmatrix}
\]

\[
\mu \equiv \{\mu_i\} \equiv \begin{pmatrix}
\mu_1 \\
\mu_2 \\
\vdots \\
\mu_n
\end{pmatrix}
\]

where \( i=1\ldots n \) and \( n \) is number of possible aging states and replacements; \( j=1\ldots m \) and \( m \) is number of possible failure modes.

The paper presents the GO FLOW and Analysis of Topological Reliability of Digraphs (ATRD) approaches to extend the fault and event trees (FT-ET) methodology for aging systems. Both approaches are used for preparation of comparable aging process control system models of the 3-train residual heat removal system (RHRS) of a Russian-design pressurized water reactor (VVER-1000). The possible extensions of these methodologies are compared with the equivalent FT models of this system.
REFERENCES

ANALYSIS OF THE REPLACEMENT NEED FOR THE CONTAINMENT ANCHORING BOLTS OF THE LOVIISA NPP ESTIMATED BY THE STRENGTH ASSESSMENT

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The type of the containment building is a double-containment, the purpose of which is to produce the isolating protection between the processes of the reactor building and the environment. The outer concrete shell of the containment building gives the external protection to the process and to the inner steel shell against the future effects of the environment and the purpose of the inner steel shell is to prevent the emissions from getting directly into the environment in any process situation. The free-standing inner steel shell has been anchored from its bottom on the ring plate on the elevation +9.60 which from the middle part extends downwards as the reactor pit reaching the reactor pit base slab on the bedrock. These reinforced concrete structures together with the vertical structures, which support the material air lock and the elevation +9.60, form the entity which in this report is called the reinforced concrete part of the containment building.

The loads used in the original design of the containment have been given in Table 1.

Table 1. Original design loads

<table>
<thead>
<tr>
<th>DESIGN CONDITION</th>
<th>TEST CONDITION (Pneumatic Test)</th>
<th>MAXIMUM OPERATING CONDITION</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design Internal Pressure</td>
<td>Design Temperature T_D</td>
<td>Max. internal pressure P_MAX</td>
</tr>
<tr>
<td>P_D</td>
<td>0.70 kp/cm²</td>
<td>0.70 kp/cm²</td>
</tr>
<tr>
<td>Design External Pressure</td>
<td>Design Temperature T_D</td>
<td>Max. wall temperature T_MAX</td>
</tr>
<tr>
<td>P_e</td>
<td>0.035 kp/cm²</td>
<td>70 °C</td>
</tr>
<tr>
<td>Test Pressure P_T</td>
<td>Test Temperature T_T</td>
<td></td>
</tr>
<tr>
<td>0.875 kp/cm²</td>
<td>0 °C</td>
<td></td>
</tr>
</tbody>
</table>

The containment model including the elevation +9.60 and the reactor pit structures as well as the steel liner on top of these structures is described in the following section. Further, the supporting reinforced concrete structures beneath the elevation +9.60 have been taken into consideration in the modeling.

The limit capacity of anchor bolts in Abaqus analysis was 0.5 MN/m² of internal overpressure, and 185 °C of internal temperature. In that case the most stressed bolt had been reached the yield stress of 700 MN/m² and the tension of the least stressed bolt was 662 MN/m². The yellow curve marked by squares depicts the most stressed bolt stress history in the Nastran analysis. The green curve marked by triangles depicts the least stressed bolt stress.
history in the Nastran analysis. The internal overpressure in the Nastran analysis rises from zero to the value of 0.41875 MN/m² and the internal temperature rises from the value of zero to the value of 157 °C. The material of anchor bolts is elasto-plastic and the yield limit of the material is 700 MN/m². The anchor bolts in the Nastran analysis are not bolts are not pre-stressed. In the limit state according to Nastran analysis the most stressed bolt is in yield stress of 700 MN/m² and the tension of the least stresses bolt is 411 MN/m². The bolt stress histories of Abaqus and Nastran analyses are given in Figure 1.

FIG. 1. The anchor bolt stress histories

One can conclude from the results that the limit capacity of the anchor bolts with respect to the containment steel shell can be expressed with the aid of the quotient 0.41785MN/m²/0.203125MN/m², which equals to 2, in other words the capacity of bolts with respect to the steel shell of the containment is double.
Due to the increased risk for energy security, rising environmental concerns and persistent energy poverty, nuclear power nowadays is being considered a viable option. Since construction of a new nuclear power plant is a costly business, plant life extension (PLEX) is also being done all over the world if permitted by the cost economics.

The Karachi Nuclear Power Plant (KANUPP), a Pressurized Heavy Water Reactor (PHWR), started its commercial operation in 1972 and completed its nominal design life of 30 years in 2002 with effective full power years (EFPY) of 10.75. Due to this, adequate safety margin existed in the system, structure and component (SSC) as revealed through international reviews and inspections. Plant life extension was therefore, decided. With the strong support from IAEA, KANUPP strived hard to combat problems of ageing and obsolescence. In order to combat ageing and obsolescence for sustained operation by meeting minimum acceptable current safety standards and for plant life extension (PLEX) beyond design life, IAEA Technical Assistance Project “Safe Operation of KANUPP” (SOK) was started in 1992.

KANUPP was shutdown in December 2002 and in December 2005 to fulfill the regulatory requirement for re-licensing of the plant. During these outages jobs related to safety upgrades, replacement, design modification, obsolescence, refurbishment etc. were carried out.

As a part of SSC design modification & modernization CO₂ Annulus Gas System (AGS) and Redundant and Forced ECI system were installed. Refurbishment of SSC include dump valves while replacement of SSC include neutron power instruments. Various safety upgrades e.g. seismic retrofits, installation of systems to handle various LOCA scenario were completed as regulatory requirements. Replacement of I&C e.g fuel handling/regulating computers, control loops were also completed to combat the problem of ageing and obsolescence.

Successful completion of the jobs related to SSC design modification, modernization, refurbishment and replacement enabled life extension and re-licensing of KANUPP ensuring clean, safe and reliable supply of electricity to the grid of Karachi city.
MATERIAL DEGRADATION MANAGEMENT OF THE REACTOR COLLANT SYSTEM AT THE POING LEPREAU GENERATION STATION

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In 2003, New Brunswick Power Nuclear (NBPN) and Atomic Energy of Canada Ltd. began developing a new approach to manage the degradation of major components in the reactor coolant system at the Point Lepreau Generating Station (PLGS). These are zirconium alloy fuel channels, carbon steel feeder piping, and steam generators with alloy 800 tubing. The objectives of this approach were to correct inefficient processes identified from review of PLGS experience and to meet the intent of the US industry initiative following the Davis-Besse experience and outlined in NEI 03-08.

Based on previous experience, NBPN recognized that industry guidelines for the development of programs to manage materials degradation resulted in processes that were not effectively implemented and maintained by station staff. As a result, operations and maintenance activities were not designed to optimize the reduction of the risks associated with the materials degradation. This was exacerbated by an increasing severity of some degradation and effort required, and an increasing demand and decreasing supply of technical resources to manage it.

This paper describes the approach used by NBPN to develop a set of program plans to are used to manage the multi-discipline, multi-component aspects of the reactor coolant system at Point Lepreau. The primary objective for this approach was to have concise management plan documents that are easy to use and update by station staff to manage very complex components with complex degradation mechanisms. Details of the key components of the main program plan and sub-program plans are included, such as:

- Regulatory commitments
- Pre-determined response plans for inspection results
- Clear acceptance criteria, e.g. for inspection results and chemistry excursions
- Maintenance activities and outage schedules based on operational assessments
- System-level plan used to manage issues that affect more than one component, e.g.:
  - Identifies interfaces between station work groups
  - Identifies operating restrictions for the system that are required to maintain structural integrity of a specific component
  - Includes a consistent method to evaluate risk-reduction to select management activities and balance resources between components
  - Uses severe core damage frequency as a common unit to compare the risks of degradation between components in the system

The importance of collaboration with stations of similar design to share resources and co-ordinate management approaches is also discussed.
REFERENCES

CONSIDERATIONS RELATED TO CANDU 6 LIFETIME MANAGEMENT

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There are 10 CANDU 6 units in operation and 1 unit in commissioning (Cernavoda NPP Unit 2). As in Canada and Korea where programs regarding lifetime management were implemented, the PLiM Program is already implemented in Romania where Cernavoda NPP Unit 1 was put into commercial operation on the 2nd of December 1996 and the Cernavoda NPP Unit 2 will be in operation in 2007. Over ten years of operation the Cernavoda NPP Unit 1 has been supplying 53 934 218 MWh to the national power grid. After 10 years of commercial operation the average capacity factor is 87.43%. Accordingly, the Cernavoda-1 NPP ranks 4th in the performance top of the similar CANDU 6 plants [1]. The time scheduling of the Cernavoda NPP PLiM Program is presented in Figure 1.

![FIG. 1. Time scheduling of the PLiM Program applicable to Cernavoda NPP](image)

Over the past 6 years, INR Pitesti (Institute for Nuclear Research – Romania) has been working on R&D Programs to support a comprehensive and integrated Cernavoda NPP Life Management Program (PLiM) that will see the Cernavoda NPP successfully and reliably through the design life and beyond.

The multiphase program, proposed to be applied at Cernavoda NPP, is supported both by the experience of CANDU 6 owners and by the results of research conducted within INR Pitesti. Thus, the first step of Phase 1 has been covered, referring to the studies on the assessment of CSSCs operation, encompassing the methodology related to the definition of critical SSCs [4] ÷ [6]. The works have been performed between 2000÷2006, within the INR R&D Program on "Process Systems and Equipment" [3] ÷ [6]. This program deals with the increase of performances of NPP systems and components; their upgrading based on the evaluation of their operation behavior. Another objective of this program is assessing and increasing of reliability and maintenance of process systems and equipment in relation with the Plant Lifetime Management.
In order to attain all the objectives of Phases 1 & 2, INR has been initiated other four R&D programs for the evaluation of “ageing” and the capability to carry on safe operation within the limits of nuclear safety (“fitness-for-service-assessment”) of the key critical components in the Cernavoda NPP, such as: “Fuel Channel”, “Steam Generator”, “Chemistry, Chemical Control”, “Instrumentation and Control” [2] ÷ [5].

Recently, on the 29th September 2006, the INR Pitesti has become a partner in the European Network of Excellence Nuclear Plant Life Prediction (NULIFE) coordinated by Technical Research Centre of Finland (VTT), http://nulife.vtt.fi The goal of this NoE is to create a single organisational structure capable of working at European level to provide harmonised R&D in the area of lifetime evaluation methods for structural components to the nuclear power industry and the relevant safety authorities. INR Pitesti is involved in NULIFE for the following work packages [2], [3]:

- **WP IA-1 “Mapping of partner RTD expertise and competences”** to provide the INR expertise and competences in application of ageing management in CANDU 6 PLiM / PLEX programs;

- **WP IA-2-3 “Lifetime evaluation”** to provide advice on/develop the lifetime evaluation tools used by the INR and to provide benchmarking of specific lifetime evaluation tools (experimental and analytical procedures, evaluation criteria);

- **WP IA-2-4 “Safety, risk information and reliability”** to review the practical applicability of various probabilistic methods for assessment of structural reliability, ageing and residual life of NPP components, identify R&D needs in this area and assess the limitations of structural component modeling in PSA and their importance for risk-informed decision making.

**NULIFE** will be the future focal point and umbrella for INR R&D activities in support to Cernavoda NPP PLiM Program.

**REFERENCES**


LIFE ASSESSMENT EXPERIENCE FOR CONTINUED OPERATION OF A CANDU NUCLEAR POWER PLANT IN KOREA

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The first PHWR (pressurized heavy water reactor) nuclear power plant in Korea, Wolsong Unit 1 reaches its 30 years design lifetime by 2012. As the plant approaches its design life, it is considered to maintain a high level of plant safety and to provide aging management programs. In this regard, "Wolsong Unit 1 Lifetime Management Study (I)" ('00.07~'03.01) was conducted to assess technical and economic feasibility for the continued operation beyond design life. KHNP (Korea Hydro and Nuclear Power) decided to perform the second phase of the study, "Wolsong Unit 1 Lifetime Management Study (II)" ('04.12~'07.05) on the basis of the results of the phase 1 study. This study covers an in-depth lifetime evaluation for systems, structures and components (SSCs) and establishment of aging management programs for continued operation.

The second phase of Wolsong Unit 1 Life Management Study evaluates technical matters for the continued operation based on the first phase feasibility study. The technical evaluation includes scoping and screening of SSCs, verification of aging status of the entire facilities through field inspection, in-depth lifetime evaluation, and establishment of aging management programs for SSCs for the extended period of continued operation. The results of this will be utilized as an input to aging evaluation part of succeeding periodic safety review (PSR) and as a technical basis for the continued operation application.

Intended functions of SSCs that are required to be maintained for the period of continued operation are confirmed by reviewing functions and design requirements of SSCs with design data of systems and structures. In addition, plant data of tests, operation and maintenance since the first criticality are reviewed and field walk-downs have done to verify current physical condition and aging status of the SSCs. The followings are general methods for the process.

Data collection and review of design, manufacture and installation documents and plant operation and maintenance history.
Determination of physical or functional boundary for SSCs
Grouping and screening of sub-components within the scopes
Aging analysis for the screened groups and sub-components
Technical recommendations and management programs based on the aging evaluation results.

Besides reviewing design or operation data, a number of on-site tests were performed like visual inspections, wall thinning measurements, performance tests of heat exchangers, temperature measurements of cables operating environments, and chemical ingredient analysis for soils in order to verify current physical condition of SSCs.
Because rules for the continued operation of PHWR is not provided yet in Korea, 10CFR54 (License Renewal) of U.S. NRC was applied to the aging evaluation and management of this study. For main technical references of life evaluation and aging management are NUREG-1800 and 1801 and Canadian experiences.

This paper introduces a Korean experience on the process and method of life evaluation and aging management programs for the continued operation of CANDU nuclear power plant.
PREVENTION OF SCC OCCURRING IN A EXPANSION TRANSITION REGION OF STEAM GENERATOR TUBING BY Ni-PLATING IN PWRs

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It is well known that Ni-base Cr-Fe alloys, especially, Inconel (or Alloy) 600 used as steam generator tubes is highly susceptible to stress corrosion cracking (SCC) in the primary and secondary side cooling water conditions in operating pressurized water reactors (PWRs) as nuclear power plants (NPPs). A Ni-plating technique has been applied to repair the cracked tubes in operating NPPs,[1,2] because pure Ni is proven to be immune to stress corrosion cracking (SCC) occurring under operating nuclear power plant environments.[3]

In this study, a method to prevent stress corrosion cracking occurring in the expansion transition region of steam generator tubing around the top of a tubesheet in PWRs was investigated by using Ni-plating. Ni-plating from both ends of steam generator tubes up to above the regions where tube expansion will be done is carried out before manufacturing the steam generators, and subsequently the Ni-plated regions of the tubes are expanded by using an explosion or hydraulic expansion method after they are inserted into the holes in the tubesheet. In order to verify the applicability of the Ni-plated Alloy 600 HTMA tubes to steam generators, the integrity of the expanded Ni-plated layers were examined and their susceptibility to SCC was investigated by using C-ring and SSRT specimens in simulated primary and secondary side operating conditions.

Commercial Alloy 600 HTMA steam generator tubes were used in this study and their surfaces were Ni-plated by using electroplating techniques. A Ni strike layer was plated first to about 5 μm in thickness before a Ni layer deposition to increase the adhesion force between the Ni-plated layer and the mother tube surface. The optimum thickness of the Ni-plated layer was determined to be about 20 μm from the hydraulic expansion method. A more detailed description of the procedures for the strike layer deposition, Ni-plating, and hydraulic expansion processes will be presented during the presentation at the meeting. After the hydraulic expansion, the soundness of the deformed Ni-layer and the interface between the Ni-layer and the tube surface were examined and evaluated by using an optical microscope (OM), secondary electron and transmission electron microscopes (SEM, TEM). At the same time, stress corrosion cracking tests were performed in pure water containing 1200 ppm B, 2.2 ppm Li, 5 ppb O₂ and 30 cc/kg H₂, at 330°C for simulating the primary side operating conditions, and in a 40% NaOH solution at 315°C and 200 mV above the corrosion potential for the secondary side operating conditions. From these tests, it was confirmed that Ni-plating on tube surfaces (in this study, Alloy 600 HTMA tubing) before the construction of steam generators can be applied to prevent SCC occurring in the expansion transition regions of steam generator tubes.

This technique has been applied for patent and the developed technology will be transferred to a steam generator manufacturing company in Korea.
REFERENCES


DEVELOPMENT OF DAMAGE EVALUATION METHOD CONSIDERING RADIATION INDUCED STRESS RELAXATION

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When stainless steel components have been in service in light water reactors (LWRs) for a long period, irradiation assisted stress corrosion cracking (IASCC), irradiation creep and dimensional changes by swelling are occurred. It is difficult to predict precisely initiation of IASCC based on the present knowledge, because IASCC is caused by acting intricately multiple radiation damage phenomena. These damage phenomena are radiation hardening (RH), radiation induced segregation (RIS), swelling and radiation induced stress relaxation (RISR). Regarding to the dose dependence of these phenomena, RH and RIS increase with increasing of dose, swelling increase rapidly over a certain threshold dose level. On the other hand, residual stress near welding portion decrease with increasing of dose by RISR.

It is concluded from analyses of damage examples of IASCC that degradation of materials under irradiation and applied stress over a certain tensile stress level are important on the IASCC initiation. If we can develop the new evaluation method for materials damage considering these change with the passage of time, prevention of rupture of structural materials is possible not only by development of new materials but also by a new design method. Figure 1 shows the change of materials degradation phenomena with the passage of time and the concept of new materials damage evaluation method. Embrittlement by RH and degradation of corrosion resistance by RIS increase with time lag with increasing of dose (operation time of reactor). Residual stress by welding, however, decrease by RISR with increasing of dose. Furthermore, new stress increase over a certain dose by swelling. Though the effect of RISR on damage evaluation is not considered in the present design method, proper damage evaluation is possible by considering the effect of RISR on damage phenomena.

In this study, we derived the dependence of dose and irradiation temperature on RH, RIS, RISR, swelling and corrosion characteristics of ion-irradiated materials using ion accelerators radiation with and without bending displacement constraint. For example, the dependence of dose on RH of irradiated materials with and without bending displacement constraint at 330°C is shown in Figure 2. Blue data show the data irradiated neutron at 250 to 340°C. The RH with bending displacement constraint is found to be smaller until about 50dpa than that without bending displacement constraint. Therefore, considering the effect of bending displacement constraint on damage phenomena, new evaluation method for materials damage is possible to apply to the structural design of components.

By investigation of both literature and experimental data using the ion accelerators radiation, the characteristics of radiation rate dependence for each damage phenomena were also modelled. Furthermore, integrated model was developed for prediction of macroscopic materials damage phenomena by the integration of elementary models.
MANAGEMENT OF STRESS CORROSION CRACKING IN PRESSURIZED WATER REACTORS

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Stress Corrosion Cracking (SCC) has recently become a significant issue for pressurized water reactors (PWR) in the United States (US). Until recently, SCC in nuclear reactors was thought to be an issue only for Boiling Water Reactors (BWR) caused by the aggressive BWR environment, susceptible material and high stresses, all of which occurred in the vicinity of pipe and nozzle weldments. However, with the discovery of cracking at several PWR plants over the past 5 years, the first being at V.C. Summer in South Carolina, the issue has become a significant issue for the US PWR industry.

Based on the discovery of more SCC at dissimilar metal welds (welds connecting low alloy steel to stainless steel or Ni-Cr-Fe), the PWR industry and the United States Nuclear Regulatory Industry initiated work to better understand the PWR SCC mechanism and to formulate approaches that would assure plant life management and plant safety. From this work, a significant amount of requirements, guidelines recommendations have been established that would support the reliable continued safe operation of PWRs. This includes augmented inspections, beyond those required by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME BPV) that would characterize the condition of all dissimilar metal welds. This also includes proactive mitigation of susceptible welds and preparation for contingency repair before inspections during an outage.

Because of the significant amount of experience in managing SCC over the past few years, it clear that implementing a well planned proactive aging management program can be important in minimizing interruptions to plant operation due to unexpected degradation. Implementation of aging management programs (Plant Life Management) can help minimize the impact of SCC on plant availability and safety. An effective plant management program considers the mitigation, monitoring, inspection, repair and replacement activities related to PWR SCC.

This paper will summarize the PWR SCC issue in the United States including historical, regulatory, industry activities and general recommendations/approaches to manage potential degradation.
FAILURE ANALYSES ON PRIMARY WATER STRESS CORROSION CRACKING OF ALLOY 600 PLUGS FOR STEAM GENERATOR TUBE AT A KOREAN NUCLEAR POWER PLANT

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Recently some boron deposits were detected on the bottom surface of the tube-sheet of steam generator at an operating Korean nuclear power plant. The following precious visual inspection was performed and as the result of that, two leaking indications of steam generator tubes were confirmed. After inspection work, plugs of 5 tubes in the hot-leg side were proved to be damaged and therefore all plugs with cracks were replaced.

In this paper, the failure analysis of steam generator tube plugs pulled out from a pressurized water reactor was introduced. The material of failed plugs was the nickel based alloy 600 and cracks were identified at the plugs by liquid penetration test (PT). To investigate the cause of cracks, micro-structural and fracto-graphic examination were performed by optical microscope (OM) and scanning electron microscope (SEM). An elasto-plastic analysis using finite element techniques was also conducted to analyze the residual stress applied to the failed plug.

The examinations showed that the cracks were inter-granular cracking propagated along minimal (i.e., less than semi-continuous) inter-granular carbide precipitation. On the fracture surfaces examined showed carbide was precipitated. Residual stress analysis showed that axial cracks resulted from the hoop stress by seal welding and circumferential crack from bending tensile stress by compulsive inserting occurred during the installation of plugs into tubes. Based on these results, it was concluded that the failure mechanism was primary stress corrosion cracking and poor carbide precipitation induced by a low mill annealing temperature and residual stresses remained after plug welding resulted in inter-granular cracking of tube plugs.
PERIODIC REMAINING LIFE EVALUATION PROGRAM OF PWR PRESSURIZER SURGE LINE ACCOUNTING FOR THERMAL STRATIFICATION EFFECT

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Pressurizer surge line contains high temperature, high pressure and high radioactive reactor coolant. It is of nuclear class I and seismic category I. It is one of the important equipment keeping the integrity of primary pressure boundary. Generally, the fatigue usage factor of surge line is comparatively high, owing to its operating temperature and pressure transients and its thermal stratification cyclic loads. NRC issued Bulletin 88-11 relating to surge line thermal stratification problem almost twenty years ago. The purpose of that bulletin is to (1) request that addressees establish and implement a program to confirm pressurizer surge line integrity in view of the occurrence of thermal stratification and (2) require addressees to inform the staff of the actions taken to resolve this issue. And the addressees are all holders of operating licenses or construction permits for pressurizer water reactors(PWRs). Meanwhile, lots of research has been done and prove that: thermal stratification of fluid in the surge line is likely to occur when the flow velocity during an insurge or outsurge is low and temperature difference between the pressurizer and the hot leg is large. Therefore the potential for thermal stratification is greatest during heat-up and cooldown because the difference between the pressurizer and hot leg temperature is then largest. Consequently, SNERDI established a program to perform temperature measurement for surge line and fatigue re-evaluation of it accounting for thermal stratification effect for a 300MWe PWR NPP during its start of commissioning. Refer to the temperature records of 5 measuring points for each of the four sections of surge line relating to four times of plant heatup and cooldown, thermal stratification phenomenon can clearly be observed. And then we modified the original design thermal transients for the surge line, included the thermal stratification and thermal stripping effects, and re-evaluated it following ASME III NB-3650. Its maximum design-basis fatigue usage factor during a 40-year operating life is bellow 1.0. There is an example of its temperature record shown in the Figure 1 bellow.

From the design safety point of view, it is enough to evaluate the surge line according to the maximum design-basis transients including thermal stratification, thermal stripping and thermal shock effect. But from the LTO & AM point of view, the actual fatigue usage factor which is a function of plant operation rather than the design assumptions is important and necessary for the preparation of the plant operation beyond its design life. Owing to the difficulty to replace the surge line, it is very important and useful to reevaluate the surge line remaining life each a certain period according to the result of on site temperature measurement during the commission. And it can be one of the foundations of the surge line life management. Accordingly, SNERDI implemented another program to perform temperature measurement for surge line and fatigue reevaluation of it accounting for thermal stratification effect for a PWR NPP of several years of operation. First, we established a temperature measurement program and system for its surge line, and obtained successfully the temperature record which shows an obvious thermal stratification existence. There is also an example of its temperature difference record shown in the Figure 2 bellow. Secondly, we established an analysis model which includes the whole RCS main loop(including piping and
equipment), surge line and pressurizer. Thirdly, we analyzed and concluded the thermal transients for both the surge line and RCS main loop. The transients includes heatup and cooldown with thermal stratification, other normal and upset transients with thermal stratification except heatup and cooldown and thermal shock transients. On the basis of temperature measurement and transients analysis, more dependable temperature difference and temperature change rate can be obtained. Fourthly, we calculate the thermal stratification globle bending stresses, cyclic through-wall bending stresses and thermal stripping stresses. Finally, we perform the fatigue evaluation for the surge line and its nozzles using the complicated model.

Subsequently, based on the temperature measurement program, existing thermal stratification data base and improving analysis methods, the periodic remaining life evaluation program of PWR Presurizer surge line concerning thermal stratification effect can be successfully achieved.

**FIG. 1. An example of temperature record**

**FIG. 2. An example of temperature difference record**
AGEING MANAGEMENT OF THE SECONDARY CIRCUIT AND RELATED COMPONENTS OF EMBALSE NPP

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Nuclear Stations designed in the 70's and that began commercial operation in the early 80's have to deal with some conceptual design features including materials and chemistry control not entirely suited to cope with phenomena like flow accelerated corrosion (FAC) whose importance has appeared later and is today worldwide recognized. Although since 1984 Embalse has been carrying out a program of piping thickness inspections, it was mainly guided by experience but not on a basis of rationale. Also the lack of documentation in electronic form and equipment data sheets made hardly difficult the evaluation of the piping and components of the circuit.

On 2004, together with the start up of the PLIM-PLEX project, Embalse decided to begin with an entire program for the Secondary Circuit which includes:

- Construction of a detailed data base for piping, accessories, dimensions and materials.
- Process data of water/steam: Temperature, pressure, velocities, void fraction, for the calculation of mass transfer coefficients.
- Chemistry conditions: base concentration, pH at temperature, etc.

By applying the FAC theory a model for pipe wall thinning rate has been developed and the results compared and fitted to historic data of the plant, some published pipe thickness thinning data and new results collected in the outages of 2004 and 2005. Figure 1 shows the distribution of the error (predicted -measured) thinning rate vs. temperature.

![FIG. 1 Error vs. Measured thinning rate of some available inspected points](image)

Also, the chemistry control has been gradually modified by the elevation of the amine dosing in the feedwater looking for an elevation of the pH\textsubscript{T} in the liquid phase of water-steam streams to enhance the protection of some regions and also to improve protection of steam generator internals. Figure 2 shows the elevation of the amine concentration after December 2005 outage.
It is also planned the replacement of the amine by other compound more suited to the steam generator liquid phase.

![Graph showing pH in Feed water after increment in base dosing](image)

**FIG. 2. pH in Feed water after increment in base dosing**

Current job includes:

- Further enlargement of the lines/components data base and inclusion of new inspection points in the ISI program (like turbine extraction lines located inside the condenser).
- Detection of other potential degradation phenomena like jet impingement, droplet impingement and cavitation.
- Evaluation of expansion joints and pipe supports.
- Computational fluid mechanics modelling of some accessories and components to improve/shorten inspection time during outages.

Figure 3 shows the relative importance of the lines in terms of thinning rate (from red to green) after the calculations detailed above.

![Graph showing relative thinning rate at Embalse water/steam cycle](image)

**FIG. 3. Relative thinning rate (from red to green) at Embalse water/steam cycle**
STRUCTURAL INTEGRITY EVALUATION OF CAST AUSTENITIC STAINLESS STEEL REACTOR COOLANT PIPING FOR CONTINUED OPERATION OF NUCLEAR POWER PLANTS

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Cast austenitic stainless steel (CASS) materials have been used in the reactor coolant pressure boundary components, such as reactor coolant piping, fitting valve bodies, and reactor coolant pump casing, in the pressurized light water cooled reactor (PWR). However, CASS with about 14% to 20% ferrite phases has been known to experience in reduction of fracture toughness caused by thermal aging embrittlement, when being exposed to reactor operating temperatures at around 280°C to 320°C, for a long period of time, like more than 30 years. Thermal aging embrittlement may cause to decrease in ductility, impact strength and fracture toughness of the CASS materials. In this paper, fatigue crack growth and flaw tolerance phenomena have been evaluated considering thermal aging embrittlement in order to ensure the structural integrity of CASS reactor coolant piping during the continued operation of the PWR plant K in Korea, from 30 years of operation to 40 years of operation.

In order for ensuring the structural integrity of CASS reactor coolant piping in the PWR plant susceptible to thermal aging embrittlement, two evaluation options are recommended, according to NUREG-1801 [1]. One method is to demonstrate the integrity through enhanced volumetric examination for base metal and another one is to perform plant specific and component specific flaw tolerance evaluation considering material property changes due to thermal aging embrittlement. In this paper, the latter method was chosen. The final crack sizes of reactor coolant pipes at the completion time of 40 years of continued operation were calculated by using the fatigue growth evaluation procedures presented in ASME B&PV Code, Sec. XI, App. A, C and L.

Table 1 summarizes the result of calculated final axial crack depths for the various subparts of CASS reactor coolant piping of PWR plant K. As shown in this table, the fatigue crack growth over the continued operation of PWR plant K is not significant. Failure mode at the continued operation comletion time might be changed from fully plastic fracture (FPF) to ductile fracture (DF), because tensile strength may increase while fracture toughness may decrease due to thermal aging embrittlement phenomena. Since flaw tolerance is changed in accordance with failure mode, failure mode should be evaluated. The results of failure modes, which were evaluated by using ASME B&PV Code, Sec. XI, App. C, Article C-4000 are also presented in Table 1, for the various subparts of CASS reactor coolant piping.

Flaw tolerance at the completion time of 40 years of continued operation was evaluated by using limit load analysis procedures, according to ASME B&PV Code, Sec. XI, Subsec. IWB, IWB-3600 and App. C. As being summarized in Table 2, it may be concluded that the structural integrity of CASS reactor coolant piping of PWR plant K, which is susceptible to thermal aging embrittlement, be maintained over the continued operation periods, since the
final crack depth ratios for all subparts are much smaller than allowable crack depth ratios.

As a result of structural integrity evaluation of CASS reactor coolant piping, considering the changes in material properties due to thermal aging embrittlement, it may be concluded that structural integrity of CASS reactor coolant piping in the PWR plants be maintained over the continued operation period of 40 years.

Table 1. Final axial crack depth for various subparts of reactor coolant piping

<table>
<thead>
<tr>
<th>Subpart</th>
<th>Initial crack</th>
<th>Final crack depth</th>
<th>Max. SIF at final crack size $K_{I_{\text{max}}}$ (MPa m$^{0.5}$)</th>
<th>Min. fracture toughness $K_{IC}$ (MPa m$^{0.5}$)</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hot leg 50° fitting</td>
<td>12.36</td>
<td>8.7884</td>
<td>10.67</td>
<td>20.55</td>
<td>203.5</td>
</tr>
<tr>
<td>Straight pipe</td>
<td>12.36</td>
<td>8.7884</td>
<td>10.27</td>
<td>20.29</td>
<td>203.5</td>
</tr>
<tr>
<td>Crossover leg 40° fitting</td>
<td>11.7</td>
<td>8.9154</td>
<td>10.67</td>
<td>20.51</td>
<td>187</td>
</tr>
<tr>
<td>90° elbow</td>
<td>11.7</td>
<td>8.9154</td>
<td>11.18</td>
<td>20.83</td>
<td>187</td>
</tr>
<tr>
<td>90° elbow with splitter</td>
<td>11.7</td>
<td>8.9154</td>
<td>11.18</td>
<td>20.84</td>
<td>187</td>
</tr>
<tr>
<td>Cold leg 35° elbow</td>
<td>12.69</td>
<td>8.7122</td>
<td>10.16</td>
<td>34.61</td>
<td>164</td>
</tr>
</tbody>
</table>

Table 2. Flaw tolerance evaluation results for axial crack

<table>
<thead>
<tr>
<th>Subpart</th>
<th>Final crack depth ratio</th>
<th>Allowable crack depth ratio</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hot leg 50° fitting</td>
<td>0.150</td>
<td>0.7</td>
<td>Integrity may maintain over the continued operation period</td>
</tr>
<tr>
<td>Straight pipe</td>
<td>0.144</td>
<td>0.7</td>
<td></td>
</tr>
<tr>
<td>Crossover leg 40° fitting</td>
<td>0.140</td>
<td>0.7</td>
<td></td>
</tr>
<tr>
<td>90° elbow</td>
<td>0.147</td>
<td>0.7</td>
<td></td>
</tr>
<tr>
<td>90° elbow with splitter</td>
<td>0.147</td>
<td>0.7</td>
<td></td>
</tr>
<tr>
<td>Cold leg 35° elbow</td>
<td>0.148</td>
<td>0.44</td>
<td></td>
</tr>
</tbody>
</table>

REFERENCES

PROGRESS OF STEAM GENERATOR AGEING MANAGEMENT OF CHINESE NPPS

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In China, the first two NPPs have operated over 15 years (QNPC) and 13 years (Dayabay). Now, most of Chinese NPPs have started their ageing management project. Steam generator (SG) is one of key safety important components for PWR and CANDU, which is selected as the first group of components to develop component ageing management system. QNPC and TQNPC have made some progress in this area. The basic SG ageing management system has been set up, and some application has been under way.

To set up SG basic ageing management system, we start with
1) SG ageing mechanism analysis
2) Developing SG ageing management program
3) Developing SG ageing management information system, SGAMDB
4) Ageing degradation assessment and remained life evaluation

SG ageing mechanisms are classified as function degradation and material/structure degradation. Tables with unified format are used to analyze the ageing locations, ageing behave, existing inspection measures and their frequency, inspection measures need to be complemented, existing mitigating measures and their frequency, mitigating measures need to be complemented, existing operation monitoring and controlling measures and their frequency, operation monitoring and controlling measures need to be complemented, and R&D suggestions. The numbering requirements for ageing mechanism are developed to manage them in ageing information system. All ageing mechanisms are graded I&II&III based on its feasibility, impact on safety and operation experience, so that the key mechanisms are managed more strictly.

SG ageing management program (SGAMP), which is fundamental document for ageing management, provides the requirements for working process, what to do and when to do. SGAMP reviewing and improving requirements are as well included.

SG ageing management database (SGAMDB) is the first component ageing management database for Chinese NPP, which have been installed and are in operation in QNPC and TQNPC. SG ageing mechanism analysis result, SGAMP and other guides for ageing management, and data of design, manufacture, operation, inspection and maintenance are stored in SGAMDB.

SG ageing degradation assessment and remained life evaluation are based on ageing mechanism, which includes integrity of tube bundle, integrity and remained fatigue life of key assemblies, and heat transfer capability degradation. It should be noticed that the impact of corrosion on tubes is primarily analyzed on operation experience of the tube material in the world.
DEVELOPMENT OF CABLE AGING MANAGEMENT PROGRAM AND EQUIPMENT QUALIFICATION IMPROVEMENT FOR LAGUNA VERDE NUCLEAR POWER PLANT

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Laguna Verde Nuclear Power Plant (LVNPP) is owned by the Mexican Federal Government. LVNPP is a BWR plant type Mark IV with two Units 675 MWe each one. Unit 1 is in operation since 1989, Unit 2 in 1990, designed by General Electric (USA) and sited in Veracruz, Mexico. The EQ Group/LVNPP and the Nuclear Research National Institute (ININ) are working on the project to perform activities to improve the Equipment Qualification Maintenance Programme (EQM Programme). Also to develop the cable aging management program (AMP) as part of the technical basis to extend the operational life, through the plant license renewal. The LVNPP EQ Program complies with the 10CFR50.49 regulation, covering required aspects and performing the related tasks, following approved procedures and documents. The EQ specifications and the EQ List were prepared, the I&C Class 1E equipment were qualified by type testing. The specific EQ plant evaluation (EQ vendor reports revision) has already done. The EQM requirements have been established and the plant has performed some tests and studies for the life extension of some equipment involved in the EQ List.

The main objective of this project is to improve the EQ implemented actions, by developing a more efficient method to follow on successfully the EQM Programme. This project is focused to following activities: a) To review the EQ applicable documentation. b) To verify and improve the communication lines among the personnel involved in the EQ process. c) To identify susceptible points of improvement in the current EQM Programme, taking as reference features and EQ practices established in standards and technical documents. Those technical documents were prepared by research organizations, other nuclear plants, IAEA, EPRI, etc. [1]. It has been performed the revision and updating of EQ procedures and related. New EQ documents has been prepared (EQ manual, procedures, etc.). The EQ data base is being updated and the establishment of complementary programs is being prepared (qualified life extension, environmental monitoring and obsolescence studies).

The EQ Group perform analysis and tasks for the continuous improvement of the EQM Program and life extension studies. These tasks are supported with current plant information from inspections, testing and monitoring over I&C qualified equipment.

The selection of parameters that indicate real equipment degradation is being prepared. Those indicators are useful to evaluate the condition-monitoring of the equipment. With this activities we will establish failure trends, perform and cause-root analysis when corrective maintenance will be frequently. The plant and ININ are working to prepare EQ condition-monitoring programs. They are derived from: a) The identification of failure mechanisms not considered during accelerated aging, b) Data assessing to achieve trend failure studies in specific equipment, c) Identification of maintenance activities inducing degradation or malfunctioning.
To start pilot studies for cable AMP, samples of new safety related cables were taken from the LVNPP warehouse. These Cables have the same model as those instaled into the primary containment. The cables are qualified for 40 years plus a LOCA DBA in accordance with 10.CFR 50.49, Standards IEEE-323, IEEE-383. The specific environmental zones are described in the LVNPP FSAR. The working plan for Cables AMP is being developed as part of the IAEA PLIM Project. The main activities included are: a) Selection of representative types of cables, b) Evaluation of current degradation status, c) Selection of methodology for life extension, d) Prediction models of cable remaining life e) Design of the Cables AMP [2] [3].

An analysis of installed cables was made and some representative types of cables were selected and carry to ININ. They are used to start the necessary steps to determine condition monitoring techniques, the line base data and prepare degradation predictive models. With these activities the methodology for AMP cables is focused to the L VNPP license renewal. In this way a cable tests program is running at the EQ Lab in ININ and an accelerated aging until 60 years is being performed. Results of this test program integrate the base line data to match with the current cable status at the plant and validate the predictive aging models. This evaluation let us to identify required actions, monitoring programs and methods to estimate remaining life.

Important degradation mechanisms were understood for cables. About thermal ageing, Arrhenius methodology has been accepted to determine the projected cables service life. Normal service and DBA conditions are considered. The condition monitoring techniques selected were: Induced Oxidation Time, Reduced Oxidation Temperature, Insulating Resistance/Polarity Index, Indenter Module, Elongation at break and Hardness. ININ is preparing the methods and the procedures to apply these techniques during the accelerated tests program mentioned above. All these tasks are carried out following technical guides established in reports, standards and documents from the IAEA, EPRI, IEEE, ASTM, ANSI, NRC and from information gathered during the PLIM workshops organized by the IAEA in México.

REFERENCES


EXPERIMENTAL RELATIONSHIP OF BREAK – ELONGATION AND INDENT DATA FOR AGEING DEGRADATION OF CSP AND CR CABLE JACKET

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Attempts to extend the lifetime of NPP have become one of the major concerns in the world nuclear industry. Consequently, life evaluation and lifetime management of cable to survive over 40 years has become major topic of discussion.

For the life evaluation of cable, break-elongation test of ASTM D412 has been widely used. Since dumbbell specimen for break-elongation test has to be supplied from plant cable, it is not easy to apply this method to operating plant. One of alternatives is indent test which nondestructively measures the aging of cable jacket. Because indent test has no approved standard, we have to compare the test result with that of break-elongation test. Since two kinds of tests for one aging evaluation are time consuming, we decided to develop relational equation between elongation and indent which can deliver relative elongation value from indent data.

Break-elongation test is destructive aging evaluation method which measures the elongation length after elongates aged dumbbell specimens until they are broken. Dumbbell specimens were prepared in accordance with ASTM D412 ‘Die C’. We kept the speed of 500±50mm/min during break-elongation test [1].

Indent test is nondestructive aging evaluation method which measures the hardness of cable by pressing a cable surface with steel probe at the vertical direction [2]. Portable cable indenter was used for this test. 2mm round bar and 0.56mm edge of truncated con type probe was used for indenting. Pressing load was limited to 0.5kg to protect damage of cable jacket. Indent depth didn’t go over 0.7mm for all of cable specimens [3][4].

As result of experiment, it was found that elongation and indent data have proportional relation. FIG. 1 and FIG. 2 show relationship between elongation rate and indent modulus for CSP and CR cables aged at 116°C, 126°C, 136°C and 146°C during 48, 72, 77, 88, 93, 94, 96, 168, 264, 312 hours. FIG. 3 and FIG. 4 show relationship between elongation length and indent depth. Relational equations between elongation and indent are described as equation (1) ~ (4)

\[
\begin{align*}
\frac{e}{e_0} &= 3.9E^a(P/d_i)^{-2.5} \quad \text{for CSP cable} \quad (1) \\
\frac{e}{e_0} &= 2.2E^a(P/d_i)^{-2.5} \quad \text{for CR Cable} \quad (2) \\
e &= 310(h_i)^{2.57} \quad \text{for CSP cable} \quad (3) \\
e &= 520(h_i)^{4.96} \quad \text{for CR cable} \quad (4)
\end{align*}
\]
After making the general equation between elongation and indent, we analyze the deviation between actual data and equation data. 14.6% average deviation for CSP cable and 10% average deviation for CR cable at the equation of elongation rate and indent modulus is observed. 9.2% average deviation for CSP cable and 7.9% average deviation for CR cable at the equation of elongation length and indent depth are observed.

Some relational equations between elongation and indent were obtained but coefficients in the equation were different according to the material type. Good convergence was observed when indent depth was used as equation factor instead of indentering modulus. It was verified that elongation data can be delivered by indent data if we had enough amount of experiment result for various material type.

FIG. 1. Relation between elongation rate and indent modulus for CSP cable

FIG. 2. Relation between elongation rate and indent modulus for CR cable

FIG. 3. Relation between elongation and indent depth for CSP cable

FIG. 4. Relation between elongation and indent depth for CR cable
REFERENCES


IMPROVING REGULATORY PRACTICES THROUGH THE OECD – NEA STRESS CORROSION CRACKING AND CABLE AGEING PROJECT (SCAP)

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For regulatory authorities, it is important to verify the adequacy of ageing management methods applied by the licensees, based on reliable technical evidence. In order to achieve that goal, 14 NEA member countries have joined in 2006 to share the knowledge through SCC and Cable Ageing Project (SCAP). The International Atomic Energy Agency (IAEA) and the European Commission also participate as observers.

The project focused on two important safety issues, the stress corrosion cracking (SCC) and the degradation of cable insulation, due to their relevance for plant ageing assessments and their implication on inspection practices.

The main SCAP objectives are to:

1) Establish a complete database with regard to major ageing phenomena for SCC and degradation of cable insulation through collective efforts by OECD/NEA members

2) Establish a knowledge-base by compiling and evaluating collected data and information systematically, with regard to major ageing phenomena for SCC and degradation of cable insulation

3) Perform an assessment of the data and identify the basis for commendable practices which would help regulators and operators to enhance ageing management

The project is managed by a Management Board (MB) and consists of two Working Groups (WGs), one dealing with SCC and the other on cable insulation degradation. Participants at SCAP are experts from regulatory bodies, industry, research institutions and academia.

The 1st MB was held in June 2006. The Terms of Reference of the project were approved and each member country nominated the representatives in the two WGs. The MB is responsible of approving the programme of work to be carried out by the WGs, monitoring the project progress in terms of results and timeliness and supervise reporting within and outside the project.

The Cable WG and the SCC WG has met twice, late 2006 and early 2007, where the scope and organization of the databases were discussed. The databases format and content for the cable failure event and the SCC event has been discussed and agreed. The database is being set up with the technical support from a Clearing House.

Based on operating experiences, including databases at the NEA and R&D information provided by the member countries, the preliminary database structure was defined and covers the following items:
For the SCC database

- Event description (Background & summary)
- In-service inspection (ISI) history
- Root cause information (Analysis of contributing factors)
- Evaluation of SCC

For the Cable database

- Technical data of Cable (e.g. Cable specification, Operating environmental condition)
- Cable maintenance data (e.g. Cable inspection, Cable sampling, Cable Failure reports, cable repairing)
- Regulatory information (e.g. Regulatory requirement for cable ageing management)
- Cable condition monitoring method under development

This project will last for four years through a Japanese voluntary contribution and it is anticipated that the database definition and the collection of data from member countries would take approximately two years. The subsequent assessment and the commendable practice report are expected to take one year each.

The paper presents the experience gained in designing and using the data base, as well as how the information will be used to perform the assessment and thus support regulatory authorities.
THE COLLECTION OF INFORMATION, DATA AND MATERIALS SAMPLES FROM CONCRETE STRUCTURES ON NUCLEAR FACILITIES UNDER DECOMMISSIONING FOR AGEING AND DEGRADATION EVALUATION

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The collection of information from nuclear facilities under decommissioning can usefully inform operators of existing plant and designers of new facilities. The evaluation of the performance of concrete structures after a considerable period from their construction is a relevant safety issue for nuclear installations. Many nuclear facilities are approaching the end of their design life and programmes to extend operating licences have been undertaken by different countries to ensure adequate safety levels for extended operational periods and for the complete decommissioning phase that can last, depending on the strategy of decommissioning, from a few years up to some decades before dismantling. Ageing management programmes should encompass the full life cycle from conceptual design through to final decommissioning.

During decommissioning, dismantling and demolition it is possible to obtain samples from systems, structures and components that have experienced ageing mechanisms in situ and are therefore superior to artificially aged laboratory specimens in informing ageing management programmes and designers of new plant. Such programmes are of value not just to NPPs but also fuel cycle facilities and waste facilities.

In order that full benefit may be obtained from the collection of information, data and materials samples from concrete structures on nuclear facilities under decommissioning for ageing and degradation evaluation, these activities must be included in decommissioning plans.

In cases where systems, structures and components have been selected for removal, sampling or testing for such purposes during decommissioning, these should be clearly identified and described. This allows specific hazards during these operations to be identified and prevents unintentional loss or destruction of the relevant systems, structures and components during decommissioning.

This paper describes the information that may be collected during decommissioning in order to inform plant life management and ageing programmes and new plant designs and the activities required to collect this. Suggestions are also given for the expansion of similar information collection from non-civil engineering systems, structures and components.
RESULTS OF GERMAN INVESTIGATIONS ON DAMAGE DUE TO MATERIAL AGEING AND CORROSION

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Two main research programmes are on behalf of BfS to counteract material ageing. The results of these research programmes will be reported in the intended paper.

One fundamental prerequisite to counteract ageing is the continuous evaluation of German reportable events with regard to material degradation since 1972 in an extensive research project called „Central Investigations and Evaluations of Manufacturing and Operational Defects of Pressurized Components in Nuclear Power Plants“. In this sequence of projects with typical durations of 3 years actual events or important issues of material degradation are covered by MPA Universität Stuttgart (Materials Testing Institute, University of Stuttgart). In the ongoing project the fatigue of austenitic stainless steel dependent on time and strain in high temperature water and the improvement of ultrasonic testing of bi-metallic welds are investigated as main topics. The project is planned to be terminated in November 2007. First results show the strain dependency of fatigue life in air and simulated light water environment of austenitic stainless steels (figure 1), compared with international fatigue curves [1, 2, 3]. One more result is the improvement of ultrasonic testing of bi-metallic welds by using a testblock with a realistic transverse flaw in the circumferential pipe weld. This flaw is a crack of about 6 mm depth, generated by genuine IGSCC, similar to the wellknown SUMMER crack.

The other research project covers corrosion induced damages. In recent years, in non-domestic nuclear power plants corrosion induced damage has been observed which was safety significant. As examples the boric acid induced damage to the head of the reactor pressure vessel of the Davis-Besse NPP (USA) which was identified rather late and the damage in the Mihama 3 NPP (Japan) caused by erosion-corrosion shall be mentioned. These recent cases of damage are not due to intercrystalline stress corrosion cracking, corrosion of nickel-alloys, strain induced stress corrosion cracking or transcryalline stress corrosion cracking that have been observed and analysed in considerable depth. As a consequence of different maintenance strategies damages of this severity are not considered likely in German NPP but they can not be excluded. Precaution requires therefore that these damage mechanisms are analysed and evaluated. With the objective to continue to avoid corrosion damage in German NPP a special investigation programme „Corrosion Analysis“ has been launched which is carried out by TÜV NORD EnSys on behalf of BfS.

In addition to gaining insights on damage prevention, the programme is supposed to identify needs to adjust regulatory requirements and technical standards to the insights gained and issue recommendations how to achieve these adjustments.
FIG. 1: Strain dependency of fatigue life in austenitic steels in air and water

REFERENCES


MITIGATION OF DEGREDATION OF HIGH ENERGY SECONDARY CYCLE PIPING DUE TO FLOW ASSISTED CORROSION (FAC) AND LIFE MANAGEMENT OF HIGH ENERGY PIPING IN INDIAN NUCLEAR POWER PLANTS

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The phenomena of wall thinning in carbon steel piping due to flow assisted corrosion (FAC) are not new to nuclear industry and which had resulted in rupture of both single phase and two phase high energy systems piping of secondary cycle in Nuclear Power Plants.

There was a rupture of feed water pipe to Steam Generator, at Kakrapar Atomic Power Station (KAPS), Unit-2, 220 MWe pressurized heavy water reactor (PHWR) in India, releasing steam to boiler room (see figure 1). The ruptured pipe was of carbon steel material, 80 NB, 7.62 mm wall thickness and rupture occurred downstream of a flow measuring orifice during operation. This failure was assessed to be because of flow assisted corrosion (FAC). Subsequent to the above incident, lot of efforts are put to prevent similar incidence of piping failures of high energy systems in secondary cycle of Indian Nuclear Power Plants (NPP’s) on account of FAC.

![FIG. 1. Rupture of feed water pipe to Steam Generator](image1.jpg)

Various measures adopted to mitigate flow assisted corrosion (FAC) related degradation and life management of high energy system piping of secondary cycle systems in Indian NPPs is the focus of this paper. This paper also brings out an overview of studies conducted to understand and identify potential FAC affected areas, inspection methodologies adopted to assess the healthiness of pipes & fittings and equipments. Observations made on degradation/failures in pipes & fittings at various stations during inspection, brief summary of analysis of inspection data and guidelines for repair/replacement/future inspection etc for Indian NPP’s are enlisted. This includes study conducted to identify the most suitable material for replacement in place of existing carbon steel material for piping of secondary cycle systems.
A uniform approach is being followed in all Indian NPP’s by implementing periodic ultrasonic (UT) thickness measurement program, its evaluation and monitoring and further upgrading of the thickness monitoring program. Some of the characteristics of FAC management program adopted in Indian NPP’s for secondary cycle systems are given below:

- Identify systems which are susceptible to FAC and selection of sample of these systems for inspection based on Engineering Judgment and experience.
- Inspection of selected components.
- Analysis of data collected during inspection to determine FAC wear rates and balance life.
- Guidelines of future inspection times based on past inspection results.
- Repair or replacement of piping components determined or predicted to wear below the minimum thickness required.

Further, guidelines and procedure for FAC monitoring program in a very systematic and comprehensive manner are prepared and issued including grid sizing criteria, guidelines for repair / replacement and future inspection based on the balance life (see Figure-2 below), and procedures for repair. Format for reporting inspection parameters was also prepared and issued as part of the program so as to have a comprehensive data bank in future for all stations and projects. Outcome of inspection done at various stations have been analysed to identify the locations of degradation and possible causes. Accordingly actions are also taken for replacing the pipes and fittings at FAC prone locations in the systems with a better material. Steps are also initiated to review the chemistry parameters and to suggest changes, if any. FAC inspection plan for secondary cycle equipments is also evolved and being implemented at various stations.

![FIG. 2. Flow Chart](image_url)

Estimation of Balance life

- Balance life >2<4 years
  - Yes
    - Repair or replace before 50 % balance life is over
  - No
    - Inspect the degraded components with balance life > 4 years before 50 % balance life is over.
FAILURE ANALYSIS ON EROSION WEAR OF RCW HEAT EXCHANGER TITANIUM TUBES

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Erosion is defined as an abrasive wear process in which the repeated impact of small particles entrained in a moving fluid against a surface results in the removal of material from that surface [1]. In recirculation cooling water (RCW) heat exchanger tubes, the erosive wear due to foreign bodies such as oyster shell, sand, etc. usually induce the thinning damage and fracture of the tube materials.

RCW heat exchanger titanium tubes with material specification of ASME SB338 grade 2 failed after operation of only two and a half years in Qinshan nuclear power plant. The heat exchanger was of the shell-and-tube type with seawater inside the tubes and secondary desalted water at the shell side, which has allowable design life of 40 years. The service temperature of the media was about 30–42°C. The flow flux, velocity and the inlet pressure of the seawater at the tube side is 3.34m$^3$/s, 2.7m/s and 0.3MPa, respectively. Several foreign bodies have been found in the tubes during the visual examination. Some of the tubes were jammed by barnacles and some other tubes were plugged up with sand sediment.

The environmental medium including water, barnacles and sand sediment were primarily investigated using graphite furnace atomic absorption spectrometry, inductively coupled plasma atomic emission spectrometry, optical/scanning electron microscope equipped with energy dispersive spectrometer, and X-ray Diffraction.

The failed heat exchanger tubes were subjected to chemical composition analysis (C/S gas analyzer and N/O/H chromatography meter, etc.), metallography analysis, microstructure examination and a series of mechanical property test including tensile test, hardness test, flaring test and flattening test. The testing results show that the material of the heat exchanger tubes in service is conformity with the relevant materials standards. The cause of the failure was thoroughly investigated using stereo/optical/scanning electron microscope. The morphology of the titanium tube due to plugged sand sediment and barnacles were shown in Fig.1 and Fig. 2, respectively.

The study revealed that the crevasses have a typical character of abrasive grinding and plastic deformation. One of the tubes failed because of the abrasive wear and extruded plastic deformation of the shearing stress due to the plugged sand sediment. Another tube failed also due to the erosion effect. The jammed barnacles or sediment decreased cross sectional area of the tube and changed the direction of the flow, increasing the velocity of the seawater and accelerating the formation of turbulence. Since the erosion rate has an exponential relationship with the flow velocity [2], the wear effect of the tube wall will be enormous under this circumstance.
Based on those analysis, the following recommendations for the titanium tubes of RCW heat exchanger were proposed: the filter screen should be set at the seawater pump and at the inlet of the water chamber to prevent the oyster shell to enter the tube bundle; check out if there is sediment in the tube while stopping the heat exchanger for inspection, and flushing the tubes in opposite direction; the tapped nylon sleeving is suggested to install to prevent the erosion effect.

The study firstly displayed the tube failure due to the erosion effect of solid particles in the seawater induced by the jammed foreign bodies such as oyster shell, sand, etc, which provide a new suggestion and direction for analyzing the failure reason of titanium tubes. Also, a useful proposal for improving the structural integrity and safety reliability of titanium tube is presented.

![Morphology of the failure titanium tube due to plugged sand sediment.](image1)

**FIG.1.** Morphology of the failure titanium tube due to plugged sand sediment.

![Morphology of the failure titanium tube due to a jammed barnacle.](image2)

**FIG.2.** Morphology of the failure titanium tube due to a jammed barnacle.

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NEW APPROACHES FOR FLOW-ACCELERATED CORROSION ESTIMATION

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Main equipment and pipelines manufactured from carbon steels on NPPs are subjected to an intensive Flow-Accelerated Corrosion (FAC) wear. Malfunctions on NPPs of Russian design and construction as a result of equipments’ FAC wear occur 3 times per year on average. At present the problem of NPP equipments’ FAC wear remains very actual and until now it have not been resolved because of the following causes: no necessary regulatory documents exist, there is an imperfection of operational inspection programs, there are no certified codes for making FAC rates forecast, methods and devices which are currently used for a metal operational inspection on NPPs have many weak points.

In the report it is performed an approach for the FAC problem resolving as well as results of the primary approbation of this approach on four pilot WWER-1000 units of Balakovo NPP. The results include: NPP elements’ ranging on groups with different intensiveness of the FAC wear by means of complex analysis of damaging factors (metal properties, metal stressed state, elements’ geometry, flow hydraulics, water chemistry, etc); elaboration of computer codes for the calculation and forecast of FAC wear rates by use of a neuron networks technology; development and implementation of devices for the inspection of metal wall thickness.

Objectives of the new design-experimental approach application for the FAC wear monitoring on NPPs are as follows: development of optimal technical solutions and diagnostic methods, the forecast and control of FAC processes, an increase of the reliability and safe operation of NPPs equipment by means of the design-experimental FAC monitoring.
FATIGUE MONITORING FOR DEMONSTRATION FATIGUE DESIGN BASIS COMPLIANCE

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The design of nuclear power plant components includes a specific fatigue analysis based on a set of cyclic operating conditions considered to be conservative at the time of design. As part of the licensing requirements for most U.S. plants, plant owners agreed to limit the number of selected transients by including cyclic limitations in the plant Technical Specifications.

This paper explores the relationship of component design basis to actual cyclic operation, and helps define the consequences and actions associated with discovering previously unanalyzed events. Fatigue monitoring for a few fatigue sensitive components is a technically acceptable alternative to the cycle counting requirements contained in most Technical Specifications, and can be used to fulfill the related requirements associated with the plant licensing basis.

Implementation of fatigue monitoring programs at nuclear power plants has been very beneficial in supporting continued operation of plants, even when the original fatigue margins were challenged and also when considering the effects of environmentally assisted fatigue (EAF). In addition, the implementation of these types of programs has also become invaluable in demonstrating safe operation for extended plant operation (license renewal).
CORROSION MONITORING SYSTEM IN THE SLOVAK REPUBLIC NUCLEAR POWER PLANTS

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Corrosion stability monitoring of safety related components is an important task for the safety operation and lifetime extension of NPPs. From the operational experience and from the demands of the national regulatory bodies follows requirements for corrosion behavior monitoring of main structural materials. As the respond to these demands, several technical solutions have been prepared for the long-term monitoring of NPP structural materials. These monitoring systems are based on the principle of surveillance samples. All specimens made from original archive materials are placed into the original environment and loaded with stresses similar higher to the operational ones [1]. The main advantages of these systems are: simply construction, the same operational history and original environment as the monitored component and practically zero operational cost.

Following monitoring systems are keeping in the operation:

- Corrosion loops in primary circuit are the unique equipments that have been installed in primary circuits of Bohunice Unit-3 NPP and Mochovce Unit-1 NPP and provide the possibility to expose sets of samples in original conditions:
  - The corrosion loop in Jaslovske Bohunice is a single chamber type and was put into the operation in 1992. Up to now five sets of samples were exposed and evaluated.
  - The corrosion loop in Mochovce is a double chamber type and was put into the operation in 1999. Up to now two sets of samples were exposed and evaluated in the chamber No.1 and one set of sample was exposed and evaluated in the chamber No.2.

- Monitoring system on the primary flange of the steamgenerator collector represents a simple equipment enables a long-term exposition of various samples inside the steam generator above the primary collector flange in secondary circuit conditions. Several sets of samples were exposed and evaluated since 1995, when the system was first time installed.

- Monitoring system in RPVs water shielding tank makes possible to expose corrosion samples into the water tank surrounding reactor pressure vessel. Two different types of materials were exposed and evaluated.

- Corrosion monitoring systems placed in spent fuel interim storage pools. The spent fuel interim storage in the NPP Jaslovske Bohunice makes use a wet storing, i.e. the spent fuel assemblies, which are placed in casks, are stored in large water pools. Several sets of samples were placed into each water pools in the year 2000 and every year representative samples are periodically evaluated.
All described corrosion monitoring systems provides important information about corrosion situation of both materials and equipments [2]. From the obtained results follows that, used structural materials are stable in given standard condition and make possible to extend the lifetime of NPPs.

REFERENCES

Fatigue failure of small bore piping caused by vibration in Nuclear Power Plants is a universally concerned problem, and so the vibration assessment of small bore piping is becoming an important part of the daily work and aging management work of Nuclear Power Plants. Measurement method and corresponding assessment criterion for piping vibration levels are introduced based on ASME OM-3 and EDF experience respectively. Based on the effective velocity assessment method, the vibration measurement and assessment of 926 small bore pipes of GuangDong DaYa Bay Nuclear Power Plant were performed. The analysis of peak velocities and effective velocities indicates that effective velocity is a better parameter representing steady-state vibration levels for small bore piping. At the same time, the calculation for the allowable effective velocity of the pipes indicates that the screening value of 12mm/s (effective velocity) may be not conservative for some of the small bore pipes. Dynamic stress analysis and special monitoring and inspecting program are performed to prevent vibrational fatigue or failure occurring.
WWER PRESSURE VESSEL LIFE AND AGEING MANAGEMENT FOR NPP LONG TERM OPERATION IN RUSSIA

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Concern Rosenergoatom has organized complex measures for first generation WWER-440 power plant extension (45 years) including pressure vessel RPV items /1/. Guidelines, methodological and technological procedures supporting first generation WWER pressure vessel RPV life and ageing management for long term operation has been developed.

The next works has been fulfilled:

- new normative base;
- neutron flux reducing;
- cutting and tests of templets;
- supporting ageing managements programs for long term operation;
- PRV integrity assessment e.c.

New radiation embrittlement model will be developed

For long term operation (60 years) WWER-1000 RPV new approaches has been developed to improve:

- surveillance programs;
- RPV integrity assessment guidelines;
- ISI system e.c.

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123
IAEA COORDINATED RESEARCH PROJECTS ON IRRADIATED REACTOR PRESSURE VESSEL STRUCTURAL INTEGRITY

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The IAEA has conducted a series of Coordinated Research Projects (CRPs) that have focused on irradiated reactor pressure vessel (RPV) steel fracture toughness properties and approaches for assuring structural integrity of RPV vessels after extended service times. This paper presents an overview of the progress made since the inception of CRPs in the early 1970s.

The first two CRPs were devoted to the measurement and understanding of neutron radiation embrittlement of RPV steels. The first CRP, "Irradiation Embrittlement of Reactor Pressure Vessel Steels," focused on standardization of methods for measuring embrittlement in terms of mechanical properties and neutron irradiation environment. One outcome from CRP-1 was confirmation that specific residual elements, namely copper and phosphorus, enhance the irradiated embrittlement of RPV steels.

CRP-2, "Analysis of the Behaviour of Advanced Reactor Pressure Vessel Steels under Neutron Irradiation," involved testing and evaluation of RPV steels that had reduced residual elements of copper and phosphorus. Irradiations were conducted to fluence levels beyond expected end-of-life. In addition to Charpy transition temperature testing, some emphasis was placed on using tensile and early-design fracture toughness test specimens. Progress was achieved in reducing scatter in neutron dosimetry methods.

The third CRP included direct measurement of fracture toughness using irradiated surveillance specimens. The title for CRP-3 was "Optimising Reactor Pressure Vessel Surveillance Programmes and Their Analyses." One key achievement was the acquisition and testing of a series of RPV steels designed for radiation embrittlement research. One of these materials was given the code JRQ and has been shown to be an excellent international correlation monitor material.

The main emphasis during CRP-4, which began in 1995, was the experimental verification of the Master Curve approach for surveillance size specimens. CRP-4 was titled "Assuring Structural Integrity of Reactor Pressure Vessels," and was directed at confirmation of the measurement and interpretation of fracture toughness using the Master Curve method with structural integrity assessment of irradiated RPVs as the ultimate goal. The Master Curve approach using small size specimens was shown to be adequate for producing valid values of fracture toughness in the transition temperature region.

The fifth CRP was titled, "Surveillance Programme Results Application to Reactor Pressure Vessel Integrity Assessment." This CRP had two main objectives: (1) develop a large database of fracture toughness data using the Master Curve methodology for both precracked Charpy size and 25.4 mm compact tension specimens; and (2) develop international guidelines for measuring and applying Master Curve fracture toughness results for RPV integrity assessment. The results from CRP-5 show clear evidence that lower values of unirradiated $T_o$
are obtained using precracked Charpy specimens as compared to results from 1T-CT specimens. International guidelines also were published and provide a framework for using small surveillance fracture toughness specimens to assess the integrity of RPVs.

CRP-6 was focused on WWER RPV steels that had high nickel contents. It has been known that high levels of nickel can have a synergistic effect with copper and phosphorus increasing the radiation sensitivity of RPV steels. Some Russian WWER-1000 RPV welds had higher levels of nickel than used in typical western steels. The radiation sensitivity of these higher nickel steels was evaluated through a small round robin exercise and collection of data. Two major observations from this CRP were: (1) analyzed results clearly show significantly higher radiation sensitivity of high nickel weld metal (1.7 wt%), and (2) for a given high level of nickel in the steel and all other factors being equal, high manganese content leads to much greater irradiation-induced embrittlement than low manganese content for both WWER-1000 and PWR materials.

The seventh CRP was focused on WWER-440 steels and the need for a better predictive embrittlement correlation. In this study, a new correlation was developed that provides better predictive capabilities based upon chemical content and neutron exposure. This new correlation was developed in a framework simulating the embrittlement mechanisms for these steels.

CRP-8 is an ongoing extension of CRP-5 in that some of the outstanding issues associated with use of the Master Curve fracture toughness methodology are being studied in more detail. The three topic areas are: (1) constraint, geometry, and potential bias effects for different surveillance-type specimens and application to the RPV; (2) Master Curve shape limitations (or adjustments) for highly embrittled RPV steels; and (3) application of Master Curve to loading rates outside that for quasi-static testing. This latter topic area has required adoption of testing procedures and application for dynamic impact fracture toughness testing. CRP-9 also is underway for developing a critical review and benchmarking of calculation methods for structural integrity assessment of RPVs during pressurized thermal shock (PTS). The overall objective is to perform various deterministic PTS benchmark calculations in order to identify the effects of individual parameters on RPV integrity. The final product will be recommendations for best practices to be used in PTS evaluations.
ASPECTS OF OPERATIONAL LIFE MANAGEMENT OF NUCLEAR POWER PLANTS

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The highest priority key component of a nuclear power plant (NPP) is the reactor pressure vessel (RPV) that contains the reactor core. It has to be, by definition, fit for its operational purpose. That is, it has been designed, constructed and tested to meet the operational requirements of pressure, temperature and exposure but also it has to meet the appropriate standards for safe operation. The vessel is designed to appropriate Codes and built from good quality materials using established techniques and practices and operated to the design intent. Undue service degradation of the start of life mechanical properties, or operation outside the initial design assumptions, can lead to the requirement to re-evaluate the RPV operational life. In the last years, the internals components of the RPV have gained consideration. To ensure their essential functions, it is absolutely vital to maintain the integrity of the internal structures throughout the life of the nuclear power plant, so that this can be operated safely and with the highest level of availability. This paper comments on the scope of RPV and internals life management, and provides an overview of the current situation on ageing and structural integrity evaluation.
RADIATION EMBRITTLEMENT AND NEUTRON DOSIMETRY ASPECTS IN WWER-440 REACTOR PRESSURE VESSELS LIFE TIME EXTENSION

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The control of reactor pressure vessel (RPV) metal during all nominated service life is one of the basic conditions of the reliable and safe operation of a reactor and a NPP as a whole.

According to rules of safe operation of RPV the level of radiation embrittlement of RPV metal is supervised in frame of the surveillance specimens (SS) program. At the present one of the most actual tasks is monitoring of the current condition and prediction of the rest lifetime for WWER-440 pressure vessels of the first generation. RPVs of WWER-440 where the core welds were annealed for irradiation embrittlement mitigation have not SS programs according to their design in spite of the second ones which are equipped by SS programs.

For adequate assessment of the current condition and the expected rest lifetime for reactor pressure vessel it is necessary to have equations describing RPV materials behavior in actual operation conditions taking into account manufacturing technology, possible variations of basic chemical composition and neutron fluence value.

The methodology based on tests of templets cut out from operating WWER-440 first generation reactors was accepted for monitoring of the reactor pressure vessel condition under re-irradiation after annealing. The program of accelerated irradiation of templates in the surveillance channels of WWER-440/213 RPV with full core ("high" flux) and with dummies on the core periphery ("low" flux) has been proposed to gain embrittlement prediction assessments for the RPV welds. Furthermore it is supposed to use weld metal specimens irradiated in surveillance channels within the framework of research programs to extend experimental database.

To apply obtained results to assessment of real RPV material properties neutron field parameters on RPV wall and in SS positions were evaluated.

In the present work prediction of the radiation embrittlement behavior of the first generation WWER-440 RPVs materials based on research programs results as far as templets testing results and their transferring to the RPV wall for substantiation of lifetime extension are considered.
METHODOLOGY RESEARCH ON PREDICTION FOR OPERATING LIFETIME OF PWR RPV

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The residual lifetime of the operating PWR is always concerned by the owners and the RPV plays a critical role in the lifetime prediction of PWR due to irradiation induced embrittlement of the active core beltline and its non-replaceability. The PWR NPP in question have been operating over half of the initial design lifetime and overall survey and investigation for the RPV have been carried out focused on the Time Limited Aging Analysis (TLAA).

The transient monitoring and surveillance data had been collected to preliminarily estimate the cumulated fatigue damage for the high stress concentrated regions and irradiation embrittlement for the beltline of RPV. It has been revealed that the irradiation embrittlement is the most significant aging mechanism comparing with others, for example the fatigue damage etc.

Based on the above knowledge the efforts was moved on estimation of irradiated material, the enhance of transition temperature and upper shelf energy decline. Owing to the more than two specimen capsules have been withdrawn the specific transition temperature RTPTS for the end of lifetime (EOL) of NPP can be derived using the specimen tests data of the specific NPP and the comparison with the data from the NPP final safety analysis report (FSAR), which was predicted in design stage by the approach of R.G 1.99, had been done. Furthermore, the maximum allowable transition temperature had been estimated under the typical pressurized thermal shock (PTS) for the beltline with surface axial semi-elliptical flaw and the flaw depth is the ten percent of thickness depth according to the code requirements and the NDT measure error for such flaws. According to the balanced fuel cycles verified by dose measurement from the withdrawn capsule the fluence at the EOL and the predicted extention lifetime of the NPP in question was extrapolated. The transition temperatures for the different lifetimes are predicted and compared with the maximum allowable transition temperature obtained from PTS analysis. Finally, the residual lifetime of RPV has been predicted based on the above investigations. It is discovered from the diagram of $\Delta R_{\text{NDT}}$ versus fluence that the transition temperature increases with the fluence but the slope become lower with the increasing of fluence.

In other aspect, pressure - temperarture limits (P-T limits) for the primary reactor coolant system during heat up, cool down and normal operation of the NPP have been updated in accordance with the withdrawn specimen tests. Due to the NPP was design and constructed around 20 years ago, $K_{IR}$, the lower bound of arrest and static toughness, was used to determine the P-T limits. It is suggested that the $K_{IR}$ can be replaced by the static toughness $K_{IC}$ for P-T limit calculation to widen the NPP’s operating window if the national nuclear authority approve that changes.
CONFORMITY BETWEEN LR0 MOCK - UPS AND VVERS NPP PRV ATTENUATION

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The conformity between Reactor Pressure Vessel (RPV) of LR0 Mock-ups and NPP VVERs was studied in order to qualify the application of the mock-ups as benchmarks only, or/and as experimental tool for simulation of the NPP irradiation conditions. The LR0 Mock-ups were carried out at the Czech critical assembly LR0 in the Nuclear Research Institute in Rez near Prague [1, 2]. The attenuation factor AF, equal to the ratio of the neutron flux/fluence value at considered position through and behind the RPV with thickness T, to its value at the position onto RPV inner wall, was used for the conformity evaluation. The azimuth directions in VVERs which correspond to the LR0 Mock-up measurements are 30 degrees for VVER-440 and 0 degree for VVER-1000. The considered radial positions in VVERs correspond to the LR0 Mock-ups measurements - at the RPV inner wall, at 1/3T, at 2/3T and behind the RPV. A flux/fluence with energy above 0.5 MeV was used for the attenuation factor determination because this neutron fluence is applied in the Russian standard [3] for evaluation of the shift of radiation embrittlement temperature for VVER RPV’s steel.

The neutron flux results were obtained by three dimensional calculations with discrete ordinate code TORT and problem oriented multigroup neutron cross-section library BDL. The NPP RPV attenuation determination was based on the flux/fluence calculations carried out for Kozloduy NPP Unit 4 with VVER-440 standard core loading, Unit 3 with core loading with dummy cassettes, and Unit 5 with VVER-1000 standard core loading.

For VVER-440 the neutron flux attenuation through the LR0 Mock-ups is less than this one of NPP VVER. The difference is negative and decreases monotonically through the RPV thickness but does not exceed 10%. For VVER-1000 the difference between Mock-up and NPP results is negative too and does not exceed 9% through the RPV thickness except behind the RPV where the difference becomes positive and reaches 18%. It is demonstrated by appropriate calculations that this difference behind the RPV can be explained with the difference in the biological shielding construction of the Mock-up and NPP. The obtained results give a confidence that the LR0 Mock-ups of VVERs could be used as experiments for simulation of the NPP neutron flux/fluence attenuation.
Fluence distribution normalized to the value in the cladding

FIG 1. Fluence attenuation through RPV of VVER-1000 and LR0 Mock-up

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NEUTRON ACTIVATION OF REACTOR COMPONENTS DURING OPERATION LIFETIME OF A NPP

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Introduction:

The knowledge of the level of activation of materials, which have been exposed to high neutron irradiation during lifetime of a nuclear facility, is important for decommissioning and for lifetime extension as well if this is intended. Besides direct measurement of material probes, the calculation of material activation can provide useful and important information with respect to the long term irradiation behavior of the material of interest. The presentation gives an overview on state of art calculation methods for activation and shows examples of application with respect to decommissioning of NPP.

State of the art:

Since several years GRS uses own-developed calculation tools for material activation, where the well known ORNL code ORIGEN is applied as a main tool. The standard method is the GRSAKTIV-II code system, where ORIGEN runs in a loop over multiple material regions with different irradiation conditions of neutron flux strength and spectra, but with the same irradiation time history. Precalculated multigroup neutron fluxes and cross sections are used in 84 neutron groups. The ORIGEN libraries inside GRSAKTIV-II are based on ENDF/B-V with 6 nuclear reaction channels and 3 neutron energy groups up to 10 MeV.

Development of advanced methods and libraries for activation calculations:

Currently an extended version GRS-ORIGENX including new updated libraries based on modern nuclear point data files is being developed for practical application with 15 nuclear reaction channels and 6 neutron energy groups up to 20 MeV. In former versions of ORIGEN only 10 irradiation time steps could be used. Now a maximum of 999 time steps can be handled in double precision mode. Some problems existing in the former ORIGEN calculation method and in its data libraries concerning structural material activation calculations, the actinide build-up and fission product generation can now be solved, e.g. the Tritium, Na-22, Al-26, Fe-60 or Nb-93m generating problem. The decay data are taken from ENDF/B-VI data bases. Due to known contaminations of structure materials by uranium and thorium, the build-up and depletion chains of the heavy metal isotopes can also completely be recalculated in the same way as the build-up chains of induced fission products. More than 20 fission yield sets are taken from ENDF/B-VI data bases. The new generated ORIGEN libraries have also been successfully checked for reactor decay heat conditions.

The GRS AAA_Activation Sequence:

Executing activation calculations in the environment of a nuclear reactor one has to solve three parts of calculations, what is called a full AAA_sequence. The abbreviation AAA stands for the German words Abbrand as burn-up of fuel, Abschirmung as attenuation of
neutrons and gammas and Aktivierung as activation of the irradiated materials. Firstly 1d/2d/3d burn-up calculations KENO/OREST have to be started to find the neutron flux strength in the half-burned reactor core region. Secondly more dimensional multigroup deep-penetration transport calculations DORT from the core region to the chosen structure material region must be done to find the attenuation factors and the neutron spectra. The neutron spectra are necessary to achieve here the correct problem dependent neutron cross sections. And lastly the activation calculations have to be done by GRS-ORIGENX for the structural material regions.

For each part of the AAA_sequence a consistent set of burn-up, transport and activation libraries will be used: All neutron cross sections will be completely recalculated by point data files JEF2.2, ENDF/B-VI, JENDL3.2 and EAF97 for 500 structure material isotopes, heavy metal isotopes and fission product isotopes. Although in different data formats of the different codes, the same neutron cross sections will be used in the burn-up, the transport and the activation step of the AAA_sequence.

The two first parts of the AAA_sequence have now already been programmed in a closed GRS code system called DORTABLE for practical appliance, which consists of the two main code systems OREST and DORT and some interface tools for data transfer and cross section handling, running in a UNIX or LINUX environment. It is planned in the future to include additionally the extended GRS-ORIGENX code, which at moment is being started by the user as a stand-alone version.

Comparison of calculations with experiments:

At first we will show in this paper the application of the new AAA_sequence for activation calculations of the steel upper and lower parts of an UO2 assembly, which was irradiated four cycles in a PWR. The results are compared with measurements, published 1987. At second we will use the AAA_sequence for the irradiation of a test material of the biological shield, irradiated for one hour in a Japanese research reactor to analyze the impurities in concrete. The results will be compared with published measurements.

Applications with regard to the operation lifetime of a real NPP:

In this paper new results of the AAA_sequence for the operation lifetime of a real NPP will be presented. The sequence will start in a simplified whole core model generating realistic neutron flux data for a PWR type reactor. The average burn-up, the flux strength during burn-up, the inventories and the cross sections will be calculated by OREST in 83 energy groups. In the second part of the sequence the neutron flux data will be achieved by 2d deep penetration neutron transport calculations using the SN code DORT in 83 or in 175 energy groups. The up-scatter procedure will be done in 32 groups to achieve the correct flux shape in the thermal energy range of the neutrons. The anisotropy of the scattering will be handled by the PL order of 3 up to 5. The meshes of the 2d DORT core/vessel/shield model will be automatically calculated by the mesh generator, included in the DORTABLE system.

Activation of the NPP vessel:

The vessel has been divided into five layers which are activated separately by GRS-ORIGENX. The variations of the neutron flux strength, the spectra and most important...
build-up cross sections inside the vessel, calculated by DORT and interface tools, are shown at different geometric points. Evaluation of the results with respect to nuclides of interest, e.g. from the point of view of radiation protection and waste management, will be presented. A comparison with results of earlier calculations will also be presented. It will be shown that back-scattering effects of neutrons from the adjacent biological shield are important for the activation of the adjacent vessel layer.

Activation of the NPP biological shield:

The biological shield has been divided into five layers for which the irradiation has been calculated separately by GRS-ORIGENX. The variations of the neutron flux strength, the spectra and most important build-up cross sections inside the concrete different geometric points will be presented. Evaluation of the results with respect to nuclides of interest, e.g. from the point of view of radiation protection and waste management, will be presented. A comparison with results of earlier calculations will also be presented.

Analyses of important radioactive isotopes in the short, intermediate and long term:

Our analyses consist of three parts: the short term region up to 1 year after shut-down, the intermediate term region up to 100 years for intermediate storage of the components of the reactor, and the long term region up to 10,000 years responsible for final storage considerations. A maximum life time of 40 years of reactor operation has been assumed. Additionally, the influence of reactor life times reaching 50 years has been investigated.
RTPTS RE – EVALUATION OF KORI – 1 RPV BELTLINE WELD BY MASTER CURVE TESTS

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KORI unit 1, which was the first PWR in Korea, is approaching its initial licensing life of 30 years. In order to operate the reactor for another 10 years or more, it should be demonstrated that the irradiation embrittlement of the reactor will be adequately managed by ensuring that its fracture toughness properties have a certain level of safety margin.

Charpy V-notch impact data may not be adequate enough to estimate the fracture toughness of certain materials, such as Linde 80 welds. ASTM E1921, a so-called master curve test, represents a revolutionary advance in characterizing the fracture toughness of RPV steels, since it permits establishing the ductile to brittle transition portion of a fracture toughness curve with direct measurements from a relatively small number of relatively small specimens, such as pre-cracked Charpy specimens. Both the IAEA and USNRC provide technical guidance to apply the master curve test results to the integrity assessment of RPV embrittlement.

For the purpose of a long-term operation of Kori-1, the fracture toughness of RPV beltline weld has been evaluated by a direct measurement with the ASTM E1921-05 standard master curve procedure using precracked Charpy specimens. The beltline circumferential weld, Linde-80/WF-233, is the only material of concern for the irradiation embrittlement in Kori-1. Unirradiated specimens were sampled from an archive weld block, Linde-80/WF-233, in accordance with the surveillance specimen’s orientation. Irradiated samples were acquired from the broken halves of Charpy specimens reserved after conventional surveillance tests according to the ASTM E 1253 standard procedure. Three different sets of specimens had different neutron fluence levels.

The reference temperature $T_o$ was determined as -106.7°C from a total of 23 specimen data points for the un-irradiated material. Irradiated data showed a one-to-one relationship between the Charpy data and the master curve data for the transition temperature shifts of the Kori-1 weld as shown in Fig.1. Transition temperature shifts from both sets of test data were practically the same for the Kori-1 weld. It confirms that the current USNRC RG-1.99, Rev.2 procedure can be applied without a modification to the evaluation of an irradiation embrittlement of the Kori-1 weld by using master curve fracture toughness data.

Based on the guidance of USNRC and the IAEA, a PCVN bias factor of 10°C(=18°F) was added to a conservative safety margin of $2\sigma$(=65.8°F). Figure 2 shows an $\text{RT}_{\text{PTS}}$ evolution curve of the Kori-1 beltline weld estimated by using the rules of ASME Code Case N-629 and USNRC RG-1.99 Rev.2 by using the fracture toughness test results. The solid line is constructed from the best fit initial property of the Kori-1 weld, measured per the ASTM E1921-05 standard procedure. The dashed line represents a more conservative reference property, which was based on the lower bound initial data set only. In any case, the estimated maximum $\text{RT}_{\text{PTS}}$ value would not reach the current regulatory limit of 300°F.
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FIG. 1. Irradiation shifts measured by the master curve and Charpy tests.

FIG. 2. RT_{PTS} evolution curve of Kori-1 based on the fracture toughness test data.
STRATEGIES FOR SUSTAINING CURRENT NUCLEAR ASSETS

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Examination of today’s “big picture” electrical generation strategies, both economic and environmental, shows a common assumption – the current nuclear fleet will continue to operate successfully for at least 60 years. In accomplishing this current plants will provide significant quantities of low cost electricity to the public and are a fundamental, established contributor to limiting the quantity of CO₂ emitted to the environment in the production of electricity. Our industry should be honored by this confidence but should also recognize the several challenges associated with fulfilling this trust. From an operating nuclear power plant perspective there are major challenges associated with both the physical plants as well as the staff who ensure their successful operation. This paper largely addresses the several issues related to the physical plants – material degradation issues, sustaining equipment reliability and addressing obsolescence / supply chain issues. Of the multiple “people” aspects important to the future of nuclear power this paper focuses on the impact the changing workforce has on the ability to sustain high levels of equipment reliability and thus plant generation, economics. The importance of “Total Asset Optimization Strategies” to addressing these issues will be described as well as several key operative elements.

The importance of sustaining the operation of the existing nuclear fleet may be obvious to the informed. Scenario based studies performed by EPRI however provide further quantitative appreciation of this point. The key elements are:

- Significant economic contribution from high plant capacity factors and extended plant life
- Reduction of CO₂ compared to other proven generation capabilities of comparable size
- Provision of a bridge between the current nuclear fleet and the time when significant nuclear “new build” will come on line

As an example of this quantification the increase in value in today’s dollars that the electrical generation from the current U.S. fleet represents compared to its initial performance and 40 year life has been estimated to be of the order of 2 trillion dollars.

In the initial operation of the current plants much has been learned about the performance of materials installed in the plants’ primary system – BWR recirculation piping, PWR steam generator tubing, PWR vessel head penetrations, etc. The industry has a proven ability to react and deal with these issues but as the plants enter their fourth through sixth decades of operation new material aging issues may be expected and more rapid response is needed.

Proactive management of material degradation, including balance-of-plant systems is needed.
Finding solutions for physically and/or functionally obsolete plant equipment has been an ongoing challenge to the supply chain. Fortunately major plant outages have been avoided, however many feel that major improvements can be made. Successful, widespread use of digital electronic technology and expansion of coordinated relationships with suppliers providing quality components are solution examples.

Currently plants operate at high sustained levels of equipment reliability supported significantly by highly experienced and knowledgeable individuals. As the industry embraces a new workforce new paradigms in how we accomplish this work will emerge. Expanded use of and integration of electronic information, similar to other industries will be needed. Technology to integrate information is available; successful, visionary adaptation to our industry is still a challenge. Increased use of online equipment condition monitoring coupled with advanced diagnostics tools based upon our extensive current experience provides a promising strategy.

Discussions with chief nuclear officers often reveal their confidence that we can address the individual technical issues but also concerns over how to most cost effectively “get it all together”. As other industries have learned Total Asset Optimization Strategies will be needed to integrate and balance technical solutions using asset management principles and techniques.
The challenge in nuclear power lies in the very long time that has to be bridged over the life of a plant. From the first decision to start a nuclear power plant project it may easily take five years to get the first approval to make it possible to ask for tenders. The preparation of a tender may take a year even for an experienced vendor and the evaluations and negotiations perhaps another year. The regulatory review for a construction license may also take a year. Thus even if the construction time for the plant could be squeezed to five or six years, the total time of getting the plant on line can easily slip to 15 years. If the planned operational life of the plant is sixty years, a century may have passed before the plant has served its operational life and has been decommissioned. In practice this means that at least three generations have served the plant and that it has gone through several large technological changes during its life time.

The need to plan for such long time periods demands a good understanding of the nuclear technology and its requirements on people and organisations during the four major phases of plant life, i.e. design, construction, operation and decommissioning. Such requirements have to be based on recognition that nuclear is different from other technologies in at least three aspects. Firstly nuclear power plants have an unprecedented accident potential, which implies that safety cannot be compromised in any of the major phases of plant life. Secondly nuclear is a political technology, which in addition to a continuous regulatory oversight also stirs emotional reactions among media and the public. Finally it is a very complex technology in which both broad and deep know how from many disciplines has to be combined.

One conclusion from the long life time and the complexity of the nuclear power plants is that they require a national infrastructure that can grant them industrial peace for doing their job in a serious and prudent manner. The infrastructure should include a functioning legislative environment, a good educational system, support for research and development, and a businesslike regulatory regime. If a good operational environment is available, the nuclear utilities, can control the other preconditions for their success, such as high ambitions, good international contacts, plant autonomy, enough staff, a motivational climate, safety culture, etc.

In Finland these issues got a broad understanding from the very start of the nuclear power programme in the late 1960's and early 1970's. At that time research groups were established at VTT to establish a knowledge base in thermo hydraulics, materials science, instrumentation and control, and reliability engineering. The activities expanded over the years and they were instrumental in building and operating the first four reactor units.
During the 1980's the activities were organised as national research programmes. In addition to Finnish research organisations and universities, the nuclear utilities and the regulator have actively participated in the programmes. With hindsight, it is very clear that the programmes have had a profound influence on the success of nuclear power in Finland. At the end of 2006 one four year part of the Finnish national programme in nuclear research was brought to an end [1] and a new four year period was started [2].

The knowledge created in research projects has been utilised in many ways. Firstly and most importantly it has made it possible to support both the utilities and the regulator in important matters connected to the design and operation of the plants. Secondly the knowledge has been engaged in many in depth studies on various issues. Finally both the utilities and the regulator have been able to use the researchers as a resource pool for their own recruitment, and it has been possible to maintain a sound age profile in the research organisations. The researchers have been successful in networking both with other national research organisations and with international organisations. One example of this kind of networking has been in the projects funded by the European Commission within the EURATOM part of the FrameWork Programmes since 1994 when Finland joined the European Union.

The early research in instrumentation and control and reliability engineering provided a systems basis that was expanded and later divided into the areas of automation, risk analysis, control rooms and organisational factors. These areas also demonstrated their importance for the safety of nuclear power in the Three Mile Island and the Chernobyl accidents. VTT knowledge in the four areas has found practical applications in the licensing of the new plant in Finland. Present research work in automation, risk analysis, control rooms and organisational factors related to the plant life management deals with the following issues:

- I&C and control rooms modernisations,
- Implementing and licensing digital I&C,
- Risk informed decision making and licensing,
- Verification and validation of control room changes,
- Assessment and development of maintenance practices,
- Safety culture in organisational changes and the generation change,
- Management systems for conservative decision making.

More concretely the most recent projects have been involved in the following themes:

Software qualification is an essential part in ensuring functionality of digital instrumentation and control. Research has been carried out with a classification of error types and methods for error management in software life cycles. A new approach has been created that is based on the three aspects of computer semiotics: syntactic, semantic and pragmatic.

Risk-informed safety management becomes important especially when deterministic and probabilistic reasoning have to be combined. This is the case for digital instrumentation and control systems in high reliability applications. A process model for planning a risk informed and cost-effective maintenance programme has been constructed. A systematic method has been developed for the analysis of human failures from maintenance history with emphasis on identification of possible common cause failures. A quantitative reliability estimation method of computer based systems operating in safety critical applications has been developed.
Testing simulators have an important position in modernisation projects of instrumentation and control. They allow for testing of control algorithms in different stages of their development and they can be extended also to the verification and validation of control room functions. Presently a test simulator based on the VTT developed simulation tool APROS is used in the modernisation of the Loviisa nuclear power plant instrumentation and control systems.

The development of control rooms is an important task both for the four older reactor units in Finland and for the new unit being built. A scientifically based method for the evaluation of human system interfaces of complex industrial systems has been developed. The method emphasises the evaluation of the interface in the context of fulfilling operational targets and safety critical functions as given for the nuclear power plant.

Smart devices with embedded software are used at an increasing rate in modern instrumentation and control. The licensing of these devices implies an assessment of so called COTS (commercial-off-the-shelf) solutions, which can be very difficult if the device has to be treated as a black box. A goal based approach has been developed that defines claims, elaborates and apportions them to smart devices and components and then identifies the arguments required to prove the claims. The project was carried out in cooperation with organisations in the UK.

Organisational culture is the key in determining how an organisation manages safety. The research has focused on three themes: organisational changes, high reliability organisations, and methods for assessing organisational culture. Case studies have been carried out in plant maintenance and engineering organisations. Collaboration with Swedish researchers and power utilities has provided additional inputs to the work. A methodology for assessing organisational culture has been developed and tested in case studies. The methodology can be applied in both scientific and practical projects.

Management systems can be seen as the software of organisations. Their structure and content govern practices for safety management at the nuclear power plants. It is important also to reflect the new requirements on management systems that have been issued by IAEA. The work together with nuclear utilities in Sweden has surveyed the structure and content of present management systems requirements and to be placed on management systems. Specific solutions to make the managements systems easier to understand, access and apply have also been discussed.

Short descriptions of ongoing research projects at VTT in the areas of automation, risk analysis, control rooms and organisational factors will also be presented.

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Good quality Nuclear power plant life management (PLIM) program is necessary for successful Periodic safety review (PSR) and safety Long term operation (LTO). Nuclear Research Institute Rez plc (NRI) supports Dukovany NPP in enhancement of current program to fulfil modern requirements. On the base of these requirements NRI designed detail methodology for PLIM program preparation and in cooperation with NPP staff started work to prepare new enhanced program. The presentation describes proposed basic steps of the new program preparation and implementation together with examples and lessons learned from finished work.

The first part of the work was evaluation of current state of life management of SSCs important for plant life, called by the IAEA methodology Interim Ageing Management Study. Work done for mechanical components covered:

- gathering of necessary information,
- evaluation of current understanding of SSC ageing,
- classification of each SSC for PLIM taking into account safety importance, understanding of ageing and cost of exchange or refurbishment,
- evaluation of ageing monitoring and mitigation SSC ageing to maintain required safety, reliability and economic factors.

The second part of the work is design of general PLIM and particular ageing management procedures and their implementation.

To do that work effectively a database application called INFOZ was developed, which after incorporating of new management procedures can be used as a basic tool for Dukovany NPP PLIM.
CANADIAN APPROACH ON REGULATORY ISSUES REGARDING AGEING MANAGEMENT, LONG TERM OPERATION AND PLANT LIFE MANAGEMENT

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This paper discusses the Canadian Nuclear Safety Commission (CNCS) approach towards ensuring that licensees operate and maintain their plants in a safe operational condition. It briefly describes elements of the CNSC requirements and the overall regulatory process to achieve these goals.

To strengthen the role of proactive ageing management at the Canadian nuclear power plants, the CNSC is developing its regulatory document framework and regulatory oversight activities, as well as promoting further research on ageing degradation mechanisms important to safety.

The CNSC recognizes the importance of information exchange and cooperation, and is actively involved in a number of ageing management and structural integrity initiatives with industry and other regulatory agencies both within Canada and the international level.

This paper describes the current and planned initiatives to improve the Canadian regulatory requirements for ageing management programs, long-term operation, as well as the oversight for the surveillance of critical SSCs, including the use of probabilistic methods for condition monitoring and operational assessment, and risk informed in-service inspections.
REGULATORY FRAMEWORK FOR CONTINUED OPERATION IN KOREA

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As the first nuclear power plant in Korea, Kori unit 1, approaches its design life, the operation beyond design life becomes one of the important issues. To make sure that the appropriate safety level be maintained in old nuclear power plant the periodic safety review has been already in place since 2000. The legal framework of continued operation was, therefore, developed based upon periodic safety review rule, additionally taking into account aging management program, operating experiences, and recent safety research results. As results, the rule of continued operation in Korea has an outlook of periodic safety review with internal technical standards similar to license renewal. This paper will provide the rationale and safety review process, including field inspection, for continued operation in Korea.
RUSSIAN REGULATORY APPROACH TO EXTENSION OF NUCLEAR POWER PLANT SERVICE LIFE

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Due to the wide experience of NPP operation in Russia (more than 50 years) we have a large legislative base in the field of nuclear energy use.

The paper presents general system of regulatory lawful documents in area of nuclear and radiation safety and all lawful documents which concern on extension of the service life of NPP units in Russian Federation.

Since the service life of the first-stage NPP units came up to the end in 2000, the Federal regulatory documents were developed by SEC NRS on demand of Rostechnadzor [1],[2]. Long term operation has to realize the following main arrangements for extension of the service life of the NPP unit:

− to carry out a complex survey;
− to develop a program of preparation of the NPP unit for extension of its service life.
− to implement preparation of the NPP unit for operation within an extra time-period, including substantiation of safety and remaining life of components, replacement of equipment that has worked out its resources; modernization and reconstruction of the unit if needed;
− to conduct necessary tests.

These regulatory documents present methodology of the NPP unit complex survey, as well as requirements for preparation of the unit for the extra period of operation and required criteria. So, preparation of NPP unit to extension of its service life is realizing in two stages. There’s work package that is carrying out to evaluate technical possibility and economical expediency of extension of the service life on the first stage. The conclusion on safety level of NPP unit is realizing on basis of this work and suggestions on the programs of additional works for determination residual life of NPP unit systems are making. Moreover, volume and nomenclature of work for preparation to extension of service life are determining and investment project for a specific NPP unit is forming.

Investment project for extension of the service life and work package are realizing to assure safe operating of NPP unit during additional period on the second stage. The second stage is coming to the end by carrying out deep analysis of NPP unit safety and development of a report on deep safety analysis.

The paper presents the main provisions of the document NP-017-2000, gives examples of application of the regulatory documents for extension of the service life of operating NPP units in the Russian Federation.
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REGULATORY APPROACH TO THE LONG TERM OPERATION OF CZECH NUCLEAR POWER PLANTS

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Abstract. There are plans to operate Czech nuclear power plants (NPP) beyond the limits of the design lifetime. The Czech legal environment with respect to the licensing of nuclear installations is briefly described in the paper. The paper also shows the current situation and conditions of long term operation (LTO) in the Czech Republic and describes the regulatory authority’s approach to LTO. The results of the WENRA countries harmonization effort and the results of the IAEA “Safety Aspects of Long Term Operation “ EBP will be implemented into the Czech legal system either through regulations or safety guides. The main principles of the implementation are provided. The periodic safety review (PSR) relation to the licence for LTO is discussed.

1. Introduction

There are two nuclear power plants in the Czech Republic. The Dukovany NPP operates four units of WWER – 440/213 type. The Temelin NPP operates two units of WWER – 1000/320 type with Westinghouse fuel and I&C. Both plants are operated by Czech power utility ČEZ which is the major electricity supplier of the country. The utility intends to operate NPPs beyond their design lifetime.

2. Nuclear regulatory environment in the Czech Republic

So called “licensing” process for nuclear installations is regulated by the Construction Act (No.186/2006), the Atomic Act (No. 18/1997) and the Environmental Impact Assessment Act (No.100/2001) and their implementing regulations.

Issuance of three basic authorizations for all nuclear installations, i.e. site permit, construction permit and operation permit from the standpoint of the Construction Act, is within competence of the corresponding Construction Office. It is the local Construction Office for the site permit and the Ministry of Trade and Industry for construction and operation permits.

In the case that in the course of licensing proceedings arise issues protected by special regulations, the Construction Office decides by agreement or with consent of the State Administration Body which protects those particular interests. The Body concern may condition its consent on the fulfilment of the conditions established in its decision issued in compliance with authorization of relevant specific law.

Those bodies are:

- technical inspection authorities in respect of the conventional safety, including the safety of the electrical systems,
• District Municipality
  - in respect of fire safety,
  - in respect of waste management,
  - in respect of water consumption and waste water discharges,
• Czech Environmental Inspection in respect of air pollution,
• local public health authorities in respect of the occupational health protection in compliance with the Act No. 258/2000 Coll., on public health protection,
• State Office for Nuclear Safety (SUJB) in respect of nuclear safety, radiation protection, physical protection, emergency preparedness and industrial safety (pressure vessels).

The Construction Act is directly imposing on the Construction Office the duty to obtain from the applicant (constructor, operator) the permission issued by State Office for Nuclear Safety in compliance with the Atomic Act still before the issuance of the site permit, construction permit, and of any subsequent permit in respect of the nuclear installations containing project. In compliance with the provisions of the Act the decision of the Construction Office cannot be issued without this permission.

The Atomic Act establishes activities for which an authorization (license) issued by the SUJB is required. Besides siting, construction and operation (the permission is mostly issued for 10 years), a SUJB license is prerequisite also for a number of other activities, as for instance – for individual stages of nuclear installation commissioning, for reconstruction or other changes affecting nuclear safety, for discharge of radionuclides into the environment, etc.

Control activities of the SUJB are set forth in the Atomic Act as well as in the Act No. 552/1991 coll., on state inspection and monitoring.

The means of enforcement used to ensure the compliance with the legislative requirements are regulated by the Atomic Act, and include the SUJB’s power to require a remedial arrangement, to order the execution of the technical inspections, reviews, and tests into the operability, the power to withdraw the special professional capability authorization from the nuclear installation staff members upon breaching of their duties, and to levy the penalties for the violation of the law-imposed duties.

Where a delay is pending, SUJB can order the nuclear installation to reduce its power or shut down the operation at all.

The Act No. 100/2001 Coll. on the environmental impact assessment, orders to evaluate the construction projects in view of their environmental impacts within a specialized procedure in which even the public may take its part. The public can be represented either by the affected municipality which is the participant in the procedure by law, or in the form of the registered citizen initiatives. The state administration authority responsible for the issuance of the decision in respect of the constructed nuclear power plant environmental impact is the Ministry of Environment.

The inspection activities of SUJB are regulated in more details by the Atomic Act and also by the Act No. 552/1991 Coll., on state inspection, as amended by the Act No.166/1993 Coll.
Concerning the long-term operation of the nuclear installations, not only the above stated acts, but also the lower-level regulations are relevant, such as the Decree No. 214/1997 Coll., on quality assurance, the Decree No. 195/1999 Coll., on nuclear safety assurance in the nuclear installations, and the Decree No. 106/1998 Coll., on nuclear safety and radiation protection assurance in the nuclear installation upon the start-up and normal operation.

The additional documents, the NPP operator can use as a guideline while preparing the Supervisor-required proofs and arranging for the compliance with the conditions upon which any further operation can be permitted, are the Supervisor-issued instructions, terms and conditions of the supervisory decisions, requirements, and instructions contained in the documents published by the Association of the Czech Mechanical Engineers.

3. Regulatory authority’s approach to long term operation

Issuance of the nuclear power plant operating license is at the outcome of the approval procedure, within which all the nuclear safety and radiation protection aspects, including the issues of the power plant aging (ageing of components, systems, and buildings), have to be considered.

Service life of an NPP as a whole is given in its technical certificate, but it has no informational value in the contemporary economic and legislative conditions (the responsibility is borne by the holder of the NPP Operating License), as it was taken over from the Soviet philosophy in the field of the maintenance and scheduled replacement of the components and systems and of the manufacturers’ product warranties. In such a technical certificate, the so-called economic life is set out, i.e. the period of time, over which the plant is to be able to function safely and reliably with all the economic assessment related to this period of time. The technical certificate is not included in the Safety Analysis Report, on the basis of which the operating license was issued. But the design service life values of the individual components and systems are meaningful. They are subject of the further accounts below, being listed in the documentation to be considered and approved.

In respect of the nuclear safety, aging of a power plant finds its reflection mainly in the reduced “NPP safety margin” as a result of some worn out systems, components, and buildings. It must therefore be reliably proved, that this residual “NPP safety margin” is high enough and acceptable. Another nuclear safety influencing factor can rest in the development of the codes and standards to be applied while an NPP is being designed. But they are subject to changes that tend to be minor only. Henceforth, it can be deduced a priori, that an NPP cannot be operated safely.

In the advanced world, the programs are available, able to control the aging processes and keep them within the acceptable limits, if they are caused by degradation and wear and tear. They can be used as life-extending tools.

From the technical viewpoint, the main issues in connection with the further issuance of the NPP Operating Licenses (so-called “prolongation” of the life of the units beyond the limit rooted in the license and thus also beyond the limit given in the Technical Certificate) can be broken down into the following domains that should be solved:
consumption of the design service life of the components, systems, and buildings, controlled aging programs,
solution of the departures from the applicable international standards and application of the operational experience,
compliance with SUJB requirements,
innovation programs.

3.1 Consumption of the design service life

Life expectancy of the WWER units as a whole is limited by the component replacement possibilities and avoidance of the building degradation. From this point of view, the only irreplaceable part is the reactor pressure vessel (RPV) with its 40-year design service life. The design service life is to be construed here as an exactly determined (e.g. with use of the fraction mechanics algorithms) span of time the manufacture declares as the time over which the equipment must be able to function safely and reliably under the predefined conditions. The conditions are set out very conservatively, i.e. with a high safety margin, containing, beside others, the RPV material critical fragility temperature, number of the permissible fatigue cycles, and number of the permissible transience. The RPV embrittlement is monitored by the witness program, which meets the strict criteria of the ASTM (USA) standards and US NRC requirements, and there is moreover an additional method used to construct the embrittlement trend curves. If necessary, the RPV material embrittled by action of radiation can successfully be regenerated by the new heat treatment of the vessel material. The RPV regeneration is now a well-mastered process, applied to a variety of the RVPs throughout the world, also by Czech ŠKODA JS (NPPs Jaslovske Bohunice and Loviisa).

Consumption of the other components’ design service life is much more favourable in contrast to the design. In general, the degradation of the safety important buildings is insignificant.

Current condition of the main components is being detected during the service inspections, regular operability tests, and by the degradation phenomena monitoring, using the diagnosing tools. In compliance with the IAEA methodology and international practice, the controlled aging programs have been prepared for some important components in respect of their nuclear safety.

3.2 Deviations and their solution

They are mainly the safety problems that are associated with this model line of NPPs and that have been identified by IAEA. The problems relate to the deviations of the WWER model designs from the contemporary international standards and the NPP operational deviations from the current worldwide routine. The issues are categorized, depending on their importance and their solution is required by the SUJB decisions and checked by its inspectors. To be able to obtain an additional operating license, the NPP must document that mainly the problems in the higher level of importance categories have been sorted out.

3.3 Compliance with the SUJB Requirements

SUJB is checking the ways used to ensure the compliance with the conclusions drawn from the SUJB inspections, SUJB decisions, conclusions from the Safety Analysis Reports and
other documentation, which is subject to the approving procedure. The individual fields of the NPP operation are also analyzed, using a package of the SUJB safety indicators. These packages are used to establish the development trends that prevail in the various domains of the NPP operation.

3.4 NPP Modernization Programs

In addition to the attention paid prevalingly to the main primary circuit components, the preparation and solution of the actions that relate to component and system troubleshooting, the issue of the in-service inspections and their reliability in compliance with the requirements of the EU methodologies, innovation and reconstruction of the system and components that can affect safety, etc. have ever been discussed between the NPPs and SUJB. Some of these actions carried out in connection with the service life consumption and in conjunction with the pending safety problems, such as the innovation of the testing and control systems, selected emergency modes of operation, diesel generating sets, etc. are highly demanding in technical and investment terms. But this makes a basis for further successful operation in compliance with the license and for the prolongation of its term in the future.

Concerning the development of the “codes and standards”, the activities of the Czech Republic’s Association of Mechanical Engineers are significant. The Association has already prepared a series of the standards for analyzing the life expectancy of the WWER NPPs’ components and systems. The standards with their final goal to harmonize the WWER NPP evaluating processes with those applicable to the PWR reactors in EU and OECD have already been successfully evaluated by the analogous organizations abroad and registered in the appropriate EU committees.

In the field of recommendations and proposals, the SUJB staff members have prepared a series of the documents for evaluation of the life expectancy, integrity, reliability of the operational inspections. The goal is to set forth the base level of the nuclear installation safety requirements in compliance with the tenets described in the IAEA documents. This type of safety must never be compromised.

4. New legal requirements

NPP Dukovany operation licenses were issued based on Final Safety Analysis Reports revisions after ten and twenty years of operation.

The Czech regulations are revised in ten year period. There will be new revisions of regulations on design, QA and operation of NPP. The WENRA reference levels will be reflected in new regulations.

SUJB will prepare guide on LTO based on the US NRC practices and IAEA EBP SALTO results.
ACTIVITIES OF OECD/NEA IN THE REGULATORY ASPECTS OF PLANT LIFE MANAGEMENT

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The OECD Nuclear Energy Agency has 28 Member countries. In 2004 the NEA issued its Strategic Plan covering the period 2005-2009. The plan identifies six sectorial arenas of work for the NEA, including as the first arena “Nuclear Safety and Regulation”. In this arena the NEA operates through two senior standing technical committees. The Committee on Nuclear Regulatory Activities (CNRA) dealing with regulatory aspects, and the Committee on the Safety on Nuclear Installations (CSNI) dealing with technological aspects.

Recognizing that there are many common areas of interest, the close relationship between the work of the CNRA and the CSNI and the need for close co-ordination and co-operation between the two committees, it was decided to develop a joint CSNI/CNRA strategic plan. In developing the joint strategic plan it was important to recognise the current status of the nuclear power industry and, in particular, the main challenges that regulators and safety research will face in the next five years. These will likely determine the focus of CNRA and CSNI activities. One of these challenges is related to the aim of this International Symposium, which is the necessity to ensure safety over the plant life cycle.

CNRA and CSNI each prepare an operating plan which describes in more detail their committee’s organisation, planned activities, priorities and operating procedures to be used to fulfilling their mandates in accordance with this strategic plan. At the strategic level, the objective of CSNI is to perform work to address the above challenges. The strategy to be employed by CSNI is to work closely with CNRA in addressing these challenges and identify and perform work that will contribute to the resolution of related issues. To be useful, such work will need to be technically relevant and timely and clearly communicated to potential users. To help ensure that the work performed by CSNI is relevant to the challenges listed above, a list of safety issues and topics (SIT) has been developed to define the areas in which CSNI activities are to be focused.

Under the CSNI, the Working Group on the Integrity and Ageing of Components and Structures (IAGE) is responsible for conducting studies, research projects and sharing information and reaching consensus on issues related to the integrity and ageing of nuclear power plant.

The IAGE Working Group has three subgroups dealing with a) integrity and ageing of metal structures and components, b) integrity and ageing of concrete structures and c) seismic behaviour of components and structures. The main group provides overall integration and guidance, independently evaluates proposals to ensure that they are consistent with mandate
of the group, and reviews results and products prior to making recommendations to the CSNI.

The IAGE Working Group has been actively working mainly in two areas: Ageing management programmes, and External Hazards. In the area of the ageing management programmes, efforts have been devoted to address the environmental effects on the integrity of components, the reactor pressure vessel lifetime, passive components failure rates and structural behaviour, risk informed in service inspections and non destructive examinations, and the assessment of the containment integrity. In the area of external hazards, the work of the IAGE group has been focused on the seismic behaviour of SSC.

This paper provides background on the Agency itself, the results of the joint CNRA/CSNI strategic plan in terms of the main challenges that regulators and the safety research community will face in the near term and the strategy that both committees follow to address these challenges and to identify and perform the work that will contribute to the resolution of these issues. It presents the accomplishments of the IAGE working group in addressing issues related to the regulatory aspects of long term operation and plant life management, and also provides areas where efforts should be pursued.
LESSONS LEARNED FROM THE OECD NEA STUDY ON THE “IMPACTS OF NUCLEAR POWER PLANT LIFE MANAGEMENT ON LONG TERM OPERATION”

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Nuclear energy is an important component of electricity supply in many OECD countries and is gaining more and more attention of policy makers and the public in light of its real potential role in long term energy strategies aiming at minimising the risk of global climate change and sustainability.

With many existing nuclear power plants (NPP) having entered operation in the 1970s, often with an originally designed lifetime of 30 or 40 years, there has been increasing interest in the extent to which Longer Term Operation (LTO) will become a reality for plants of this generation. A significant proportion of existing nuclear capacity will become 40 years old between 2010 and 2020.

For many operating NPPs, it has been demonstrated that they are capable of safe and efficient operation for a significantly longer period than was envisaged when they were designed, with lifetimes of 50 to 60 years being likely in many cases. In most OECD countries with established nuclear programmes LTO of the NPPs has already been accepted as a strategic objective, to ensure adequate supplies of electricity over the coming decades, while in others it is being actively considered.

The main objective of the last OECD NEA study published in 2006 has been to review and analyse the impacts of lifetime extension on fuel cycle and waste management requirements, on the economics of nuclear energy, on knowledge management and preservation, and more broadly on the future of nuclear energy in OECD member countries. Its scope includes technical, economic, social and strategic issues raised by plant life management and longer term operation in countries planning an extended reliance on nuclear energy, in countries wishing to keep the nuclear option open, and in countries having decided a progressive phase-out of nuclear energy.

The report presents tendencies, the advantages and technical-economical challenges as well as environmental impacts of NPP lifetime management for LTO.

The Expert Group concluded, that the principal advantages are economic. Extending the life of a major generating asset avoids the need for immediate investment in new generating capacity. The capital costs of PLiM for LTO will be much smaller than investment in any type of replacement capacity. An important benefit of LTO will be a reduction in specific (per kWh) costs for waste management and decommissioning. With nuclear fuel costs being generally lower and more stable than fossil fuel costs, this means that LTO can be expected to provide electricity at a lower cost than any other available option.
During the operating lifetime of several decades, it will often be possible to enhance plant safety levels by upgrading systems, structures and components (SSCs). LTO helps to justify the investment in such upgrades, which means that it can also help to raise safety levels.

LTO of existing NPPs contributes to sustainability by maintaining security and stability of energy supply and the diversity of energy sources while safety is of paramount importance.

Furthermore, LTO can provide nuclear energy without the significant environmental impacts that would be created by alternative power generation options (notably CO₂ emissions). Most countries with operating NPPs consider nuclear energy contributes to the sustainability of their overall energy supply system, in that it minimises the long-term and irreversible impacts on the environment of meeting current energy demand.

When the current fleet of nuclear reactors was built, safety requirements of the existing plants were sufficiently stringent to ensure a considerable amount of conservatism in the design. Conservatism as such can facilitate LTO of the existing NPPs. While the operating experience, improved analytical techniques and training of personnel allows this conservatism for considering LTO proper regard must be given to the possibility of unknown ageing mechanisms. Properly managing the lifetime of different SSCs of NPPs LTO can potentially provide a bridge between the present generation of NPPs and future generations of power plants – either nuclear or non-nuclear.

There are a few major components (notably the reactor pressure vessel in most plants) which can be considered non-replaceable, either for technical or economic reasons. For such components it is necessary to implement ageing management programmes.

To achieve LTO it is important that there is a clear and predictable regulatory framework. Timely investments need to be made in upgrading the plant and replacing SSCs, and these will be influenced by the prospects for LTO. Once decided, the necessary licensing and approval processes need to be carried out in a timely manner. The process of consultation between regulators and plant operators need to be started well in advance.

The energy policy framework and political background are also important factors. A decision to allow LTO to go ahead may often be easier to take from a political perspective than the alternative decision to construct replacement generating capacity. However, in some cases NPP owners have continued to plan for possible LTO even where political support for it is unclear.

More broadly, it is vital to build public confidence in the LTO of NPPs. While the public in the immediate area around an existing NPP is usually supportive, LTO might raise concerns about safety. The public needs to be properly informed about plans for LTO and the basis for ensuring that safety will not be compromised. Furthermore, it is necessary to discuss the advantages and concerns about LTO.

One important aim of PLiM for LTO is to improve a plant’s operating performance. PLiM programmes have already resulted in significantly improved operational performance at many NPPs in OECD countries, which has often greatly increased the value of these nuclear generating assets.
Some human aspects of LTO were analysed as well. With LTO, NPPs may well operate for a total lifetime of 50 to 60 years. For this reason, management and preservation of knowledge are of critical importance. NPPs can be considered multi-generational projects, which will be the responsibility of several generations of engineers and other specialists over their lifetime. Steps should be taken by plant owners and by governments to support education programmes and provide suitable career opportunities for young scientists and engineers to guarantee a sufficiently large skilled workforce for the nuclear industry.

International co-operation and co-ordination are important in building confidence in LTO and international organisations have an important role to play here. At the industrial level, co-operation between plant operators, reactor vendors and technical support organisations from around the world in planning and in R&D will help ensure that the best practice is followed in implementing PLiM programmes for LTO at NPPs in all countries.

Overall, LTO can potentially provide a bridge between the present generation of NPPs and future generations of power plants – either nuclear or non-nuclear. The degree of success in achieving LTO with the current reactor fleet will have a significant impact on installed nuclear capacity over the coming decade.

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INSTRUMENTATION AND CONTROL (I&C) MODERNIZATION AT KANUPP

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Pakistan Atomic Energy Commission (PAEC) has Two Nuclear power Plants: Karachi Nuclear Power Plant (KANUPP), a 137 MW(e) PHWR that was supplied by Canada and Chashma Nuclear Power Plant (CHASNUPP), a 325 MW(e) PWR by China. A Third Nuclear power plant (CHASNUPP-2) is in construction phase and it is also being supplied by China.

KANUPP has completed its design life of 30 years and its operation life has been extended thru various modernization, refurbishing and upgradations. A program of self reliance was initiated by PAEC to ensure safe operation of KANUPP. Under this program various projects were initiated and one of these projects is Technological Up-gradation (TUP) project that was initiated for combating technological obsolescence in areas of Process Computers, Control and Instrumentation.

This paper describes methods and tools which have been used for managing the modernization of Instrumentation and control at Karachi Nuclear Power Plant (KANUPP) resulting in successful achievement of the goals set by the organization in this important area. Identification of problematic areas, requirement specifications for the modernization, resource planning, documentation control, change management, implementation plans, questioning attitudes, feedback collection and implementation, and consultation with the relevant working groups provided effective methods and tools so as to properly manage the project.

This paper also provides an overview of the facilities developed to provide technological base for the effective implementation of I&C modernization such as test and calibration facility and dynamic test facilities to ensure the reliability / functionality of the equipment being replaced and for the purpose of manpower training and development.
The physical and moral lifetime of the I&C installed in electricity generating plants is generally shorter than the envisaged service time of the power generating unit itself. Due to this situation the appropriate handling of ageing and obsolescence of I&C equipment is a continuous task in nuclear as well as in conventional power plants. Ageing and obsolescence gradually increases I&C operation and maintenance costs, problems may arise with spare parts and maintenance know-how, modification of I&C functions gradually becomes difficult or even impossible and finally plant availability is endangered.

In spite of the fact that all nuclear power plant I&C components belong to the “replaceable” category, replacement of some I&C equipment (e.g. cables or actuators) is usually handled with extra care. In the international practice the following methods are applied to handle I&C ageing and obsolescence:

- scheduled and timely repair or upgrading of I&C equipment and components,
- scheduled and timely replacement of I&C equipment and components,
- a full modernization of selected I&C systems in given time intervals.

At Paks NPP all above listed methods were applied – although to various extent – in the last 10-15 years „to fight” I&C ageing and obsolescence. Equipment replacement is carried out on a regular and scheduled basis, this systematic activity was started in the early nineties. The gradual and small-scale I&C replacement was not sufficient to cope with certain design and architectural deficiencies of the original I&C, therefore modernization projects were initiated in the mid nineties. Until now the largest project in this series was the Reactor Protection System (RPS) reconstruction, when the RPS of all units was modernized by using digital Teleperm XS equipment (see [1] and [2]). In order to provide a modern human-machine interface to the control room personnel and to replace obsolete and unmanageable hardware, plant computers were also modernized on all units. Intel-Windows based servers and SCADA software items were used for the realization of the new process computers ([3]). Then other computerized systems were reconstructed in a scheduled manner: plant information center, reactivity measuring system, VERONA core monitoring system, reactor noise diagnostic system, environmental and radiation monitoring system (see [2], [4] and [5]).

As a result of the above mentioned modernization projects and due to considerable operation and maintenance efforts, presently it can be stated that the I&C systems of all Paks units are in sufficiently „good shape”, they do not endanger safe and economic operation of the units. However, it can also be stated that the conventional (i.e. not computerized) I&C systems and associated components have approached the end of their envisaged service time. In case they
are to be used for an extended period the plant may face more frequent electricity production (plant availability) problems, and thus inevitable economic losses. As far as the planned Paks service time extension is considered it is now quite clear for the plant’s I&C experts that the modernization of these conventional I&C systems is one of the conditions for the extension. In order to define forthcoming tasks and requirements in the field of process instrumentation and control, a guide was compiled at the NPP in late 2006 (see [6]). The document proposes a large-scale I&C refurbishment project in order to meet the requirements created by the plant service time extension. I&C modernization tasks are assessed according to the IEC TR 62096 document [7] and detailed project schedules with work packages are proposed, as well.

The paper first briefly outlines the main characteristics and results of the I&C refurbishment projects completed at Paks so far, then provides an assessment of the present I&C situation. Detailed information is given on the planned large-scale I&C modernization, discussing e.g. project time-scale and scheduling, the scope of the required preparatory engineering work, the contents and technological details of the different project phases. Expected operation, maintenance and economic advantages are discussed, as well.

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MANAGE I&C OBsolescence FOR GE BWR Nuclear POWER PLANT LIFE EXTENSION

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Most nuclear power plants in operation were built in the 1970’s and 1980’s. Parts obsolescence for instrumentation and controls (I&C) has been a continued issue for these plants. Although modern I&C could benefit a nuclear plant in terms of improved reliability, reduced maintenance, and reduced plant vulnerability, I&C upgrades are not common for nuclear plants. This is because the current economic model is such that the assessed benefits for I&C upgrades, especially for the safety related applications, seldom exceeds the modifications cost. Generally, there is no capacity loss associated with I&C problems. The plant staff is usually talented and resourceful enough to maintain the existing system. Any upgrade involving safety related instruments require extensive reviews by the regulatory body, e.g., the Nuclear Regulatory Commission. Cost/benefit analysis that focuses on the immediate pay back and discounts potential long term impact could not justify the cost of I&C upgrade. Consequently, I&C upgrade has been conducted more from a as-needed basis as opposed to being based on a strategic plan.

However, plant life extension would change the economic model for the I&C obsolescence issue. The estimated revenue for all of the GE BWR for a 20-year life extension is estimated to be $356 billion in today’s dollar. The economic benefits would be even higher again if a nuclear plant seeks to increase it power output by programs such as Extended Power Uprate (EPU). Programs such as Performance 20™ are designed for the plants to achieve maximum benefits for 20 or more years of operation with 20% more power level output. Given the significance of these potential benefits, utilities must evaluate obsolescence issues as a threat to plant life, as the obsolescence issue will only become more difficult in future years. Parts that are difficult to obtain today will become impossible to obtain in coming years. The skill set that currently serves to keep the plant in operation will also diminish in time and which will affect the ability to maintain the older equipment and systems. Therefore, I&C upgrades should be planned to be in concert with the plant life extension and/or power uprate efforts.

The planning of the I&C upgrades to support plant life extension should be executed in a systematic approach. Six Sigma tools, such as Quality Function Deployment (QFD), can be used to help prioritize I&C upgrades for coming years. An example of employing such a tool would be to first list the criteria or goals for upgrades. This is also termed “Critical to Quality (CTQ)”. The CTQ should be developed based on inputs from plant staff including engineering, maintenance, and operations. Where applicable, quality assurance, outage management, and plant strategic planning staff should be included. Examples of CTQs are: (a) reliability, (b) ease of maintenance, (c) ease of operation, (d) outage impact, (e) long term viability, (f) ALARA, and (g) ability to support plant operation (e.g. EPU). The CTQs are then weighted based on the consensus of the plant staff. The performance of each I&C system can then be evaluated against the CTQ. The result of this evaluation represents the priority of the systems that would require upgrade or other actions. Possible solutions for each of the system can then be developed based on a more specific set of CTQs for a
specific system. This evaluation will identify the optimal solution for a given system. It should be noted that upgrade might not be the optimal solution for all obsolescence issues.

For example, if the issue is with a part, e.g. the scram contactor in the reactor protection system, it may be cost effective for the utility to team up with other utilities to purchase sufficient parts for the rest of the plant life before the part becomes obsolete. An example of such a QFD is shown in Figure 1. Potential I&C upgrades that would meet the listed CTQs for a GE BWR are Power Range Neutron Monitor (PRNM), Wide Range Neutron Monitor (WRNM), Rod Control and Management System (RCMS), Turbine-Generator Electro-hydraulic Controls (EHC), Recirc Flow Control (RFC), Feedwater Level Control (FWLC).

FIG 1. Application of QFD tool for I&C upgrade

The prioritization of the system upgrades is only a part of the solution to support plant life extension. In order to support plant operation, a systematic approach is required for the managing of the obsolescence for I&C systems. Therefore, it is necessary to develop a strategic plan for continuous managing of the obsolescence issue. Over decades, GENE has gradually developed a system as shown in Figure 2 to assist power plants in dealing with these problems. The objective is to achieve minimal outage, power output uprate with 20 more years of operation, and safer and more secured and reliable operation.
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CASE HISTORIES AND LESSONS LEARNED FROM THE DESIGN, DEVELOPMENT, PLANNING AND IMPLEMENTATION OF NEW I&C SYSTEMS, INCLUDING EFFECTIVE INTEGRATION WITH EXISTING SYSTEMS AND PROCESSES

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**Background:** Extending power plant lifetime, enhancing plant operations and performance will result in many components and systems being modernized, upgraded or replaced. New approaches for monitoring the plant structures, systems and components will be required to allow safe ageing and risk management. This implies new sensor technologies and monitoring concepts will be integrated into or replace existing plant I&C systems and subsystems. It will be essential to ensure and demonstrate these changes will never compromise the safety of plant operators and the public.

This paper will go beyond new technology approaches and implementations. It will consider the best practices, procedures and approaches needed to handle the seemingly mundane human factors that make or break the introduction of new equipment and technologies? It will consider effectively involving people to reach the full potential of these enhanced capabilities, and further address what is really involved when retrofitting systems?

This paper is based on practical, wide ranging and in-depth experience of many aspects of power plant design and operation. The authors have decades of experience working in the nuclear industry, as well as high tech fields. This includes leading teams on development, design and implementation of plant diagnostic and monitoring equipment; fundamental technology and R&D investigations; and integration of advanced technologies and processes into nuclear energy, space reactors, and other industries. The paper will draw from this experience and describe the lessons learned when implementing systems ‘from-womb-to-tomb’, and sometimes beyond! We have been called upon to revive legacy projects. We will use practical experiences to develop and illustrate lessons learned, and suggest resulting best practices.

**Passive Acoustic Tomography Condition Monitoring System:** The authors will use a plant condition monitoring system as a foundation for suggesting best practices. It was developed and tested for use in nuclear plant monitoring; specifically for protecting steam generators. This system uses passive acoustic tomography monitoring techniques to identify and locate anomalous sounds within plant components. When used to protect process plant it uses externally mounted sensors to monitor the internal volume of a large vessel (i.e. a steam generator), and indicates when an internal defect or fault condition exceeds a threshold intensity. This application provides both the absolute sound intensity and its exact location conditions within the internals of the plant component. It detects and precisely locates the fault. The experiential database will be followed from idea conception and proof of principle testing through system development, design, installation, operator acceptance of I&C subsystem, and performance experience on operating steam generators.
Defining an I&C Subsystem: Technologies, instruments and sensors do not exist in isolation; they must be safely and effectively integrated into the plant I&C system. They must also fit into the monitoring and control culture of the staff and plant operators. To progress from a good idea to an instrumentation subsystem suitable for monitoring operation and protection of a critical component power plant such as a steam generator is not a straightforward process. It has many stages, and each must be addressed totally in order to enhance the power plant life or efficiency. The very first stage is to decide upon the subsystem requirements. This is not just listing expected sensitivity and performance parameters.

a) What does the monitoring subsystem have to do? Most plant component failure modes have a precursor condition that precedes serious or catastrophic failure of a component. The major design objective of a monitor is to detect this incipient or initial condition and allow a timely and controlled movement of the plant to a safe mode of operation. Note, the detection criteria is not a sensitivity requirement, it is a required protection action. Defining an action is potentially a more powerful requirement than defining a list of expected I&C system operating parameters. For the acoustic monitor the action requirement became

- The monitor must detect an incipient fault level within a predefined time for all operating modes.

A required action requirement is typically independent; it is not linked directly to another action requirement. Further definition sets limits or ranges of monitoring parameters.

- Ability to extract water/steam leak acoustic noise when totally masked by background noise S/N -5dB detection time within 0.5 seconds at full power operation and S/N of -25 dB within 5 seconds during standby operation of the component.

b) What does the Utility operator need? The cost of a power plant ‘SCRAM’ is not only the loss of revenue from selling power. This loss can become almost negligible compared to the cost of persuading regulatory bodies the cause of the shutdown is known with certainty and returning the plant to full power operation. Every SCRAM also impacts the remaining lifetime of the plant due to the thermal shocking etc of components. Proving an alarm is false can be quite difficult, and may require extensive analyses and diagnostic investigations. The second required action requirement is responsive to these factors.

- Reliable detection of fault condition with very low false alarm rate.

Again it is an action not a number that is important. The numbers will define an intensity or a range over which actions are expected. Further definition to set limits for the acoustic monitor:

- The system will not reduce the plant lifetime by operating with less than one false alarm in thirty years.
c) What are the engineering staff concerns? Most Instrument and Control systems are not self contained. They have to be installed into an existing physical structures, and often have to be physically attached, pass through, or be fused to the walls and piping of critical structures. Meeting all of the safety, installing and operating requirements are usually a mandatory condition of installation. Every I&C component or part, features, every expected operation (both planned and unexpected) have to be considered. Regulatory design limits (e.g. NRC regulation, ASME code, etc) have to be met. Supplementary investigations were often needed to confirm there would not be problems generated by incorporating the acoustic monitor into the plant I&C system.

- *Installation and operation of acoustic monitoring system will not degrade the physical plant or it’s I&C systems.*

Again it is required actions that govern the monitoring system detailed design. There are still the issues of installing the monitor into the plant I&C, and then integrating its operation into overall operating procedures.

The next stages include installing a monitoring system, and in the author’s opinion the most difficult, is to gain plant management and plant operator acceptance of the system. This is not a trivial problem. It requires very deliberate efforts to build operator confidence in the system from understanding the concept to daily use expectations.

d) *What does the power plant operations management need?* Their main objective is to keep the plant online. The operators want a monitoring system that will have an insignificant impact on day-to-day operations. They have only peripheral interest in non-critical monitoring instruments. This is especially true if the instrument is looking for a low possibility event. One does not expect a steam generator to have regular fault events; it is most likely to be a very unusual condition. A critical integration factor is to educate and train both management and operators on all aspects of the monitor, including its operating concept/technology, design, features and operation.

The goal is to reach for 100% operator confidence in the system. If they have any doubts about its efficacy they may have a tendency to doubt its efficacy, and thereby limit the functionality of the monitor. The immediate instrument/monitor operational management problems include:

- Getting operator “eyeball space”, especially if looking at very long term data trending. Maintaining operators understanding of the monitors use, operation, and expected responses. (action requirement: Operator understanding and knowledge reinforcement)

The acoustic monitor described in this paper has integrated healthy operation and fault operation diagnostics. Regular reports are provided to operators and management.

How can the instrument designer maintain operator interest in the monitoring system from womb-to-tomb? Relatively sophisticated features are incorporated into the GRDI Acoustic Monitoring Subsystem. The design requirements actions include:

- *Make the monitoring system interactive, integrate and maintain operator training and knowledge.*
The software has “built-in” capacity to provide off-line plant operator training in reacting to alarms. It can be used by management to check operator performance by simulating different levels of fault condition.

e) What is system’s added value to operating staff? The instrumentation subsystem includes integral thermal/hydraulic modeling of the system being monitored (e.g. steam generator and secondary loop). This allows "what if" scenarios and system design analyses by engineering staff when considering different age management possibilities. This interactive capability was used to confirm the use of neural networks and fuzzy logic system enhancements improved operator interfacing and increased monitor performance.

And then there are the issues of planning against obsolescence of all aspects of the monitoring system. From its inception, the monitor was designed with the assumption that hardware, software, requirements and users of the system would change. The acoustic monitor has successfully accommodated orders of magnitude changes in hardware systems, and several computer operating systems and languages. The same basic passive acoustic tomography monitoring system has been used to successfully monitor the condition of rotating machinery (ship propulsion system) and to investigate three dimensional jet engine noise generation characteristics. The experience gained from the passive acoustic tomography plant condition monitoring system showed it is possible to design, document, and “keep young” systems throughout decades of technology change.
Monel 400 steam generator tube R41C52 was removed from service from Pickering Unit 4 Steam Generator 12 in 2005. Laboratory UT examination showed an outside surface crack-like flaw approximately 36-mm long, and 81% through-wall. Both the axial flaw model of Ontario Power Generation Steam Generator Fitness-for-Service Guidelines (FFSG) and finite element analysis were employed to perform the Condition Monitoring Assessment required by the FFSG.

Metallurgical examination showed that the flaw was 71% through-wall deep and initiated at a manufacturing defect. The crack-tip was blunt and characteristic of a crack that had propagated slowly by corrosion. Orientation Imaging Microscopy (OIM) showed that there is a lack of plastic deformation along the crack path except for a small amount of plastic deformation concentrating only at the crack-tip, see the following figure. The non-mandatory FFSG axial flaw model predicts the failure pressure for this flaw to be 11.5 MPa. This is only slightly higher than the maximum pressure differential 8.7 MPa that occurs during a small temporal window at startup and shutdown, and 9.5 MPa for accident or faulted conditions (main steam line break). The margins on the failure load are insufficient to meet the “prohibiting leakage” acceptance criteria of the FFSG. Therefore, the flaw in tube R41C52 would not pass the Condition Monitoring Assessment of the FFSG, if the FFSG axial flaw model were used.

Alternatively, a heterogeneous finite element model (HFEM) and a failure model were used to predict the failure pressure for this flaw. The HFEM considers the inherent variation in mechanical properties due to the spatially heterogeneous microstructure, such as texture. The FE model (mesh size, element type and time step) and the failure model have been extensively calibrated and validated with respect to a database of burst-pressure test results of Monel 400 tubes with a variety of defects [1]. When applied to the flaw in R41C52, the calculated plastic strain at the crack-tip for the maximum normal operating pressure differential (8.7 MPa) is limited at the crack-tip and the maximum value is 0.005 as shown in the figure. This agrees well with the measured plastic strain distribution from the OIM. The predicted failure pressure for this flaw is close to 26 MPa, resulting in sufficient margin on load to meet the “prohibiting leakage” acceptance criteria of the FFSG. Therefore the tube R41C52 passes the Condition Monitoring Assessment of the FFSG, when this alternative HFEM simulation is used.
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The measured plastic strain distribution using the Orientation Imaging Microscopy (OIM) technique. It is clear that plastic strain only occurs at the end of crack-tip along the thickness direction.
NUCLEAR POWER PLANT LIFE MANAGEMENT: MATERIALS AND COMPONENTS, RESEARCH, HUMAN RESOURCE, RADWASTE AND REGULATORY ASPECTS

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Abstract: Aspects concerning nuclear power plant (NPP) life management (PLiM) programmes are examined. The objectives of PLiM and their influence on safety and operation are presented in terms of potential gains in safety, reliability and performance of systems, structures and components (SSCs) by giving due emphasis to social and economic considerations. A nuclear plant implementing a PLiM programme needs state of science and technology information and approaches as well as availability of well trained personnel in sufficient numbers to achieve safe and optimized operating life of the NPP. Accordingly, the importance of succession planning for assuring trained human resources, knowledge management and the need for continued research in all fields of nuclear power generation are highlighted in the paper.

Commercial nuclear power has been a feature of the industrialized world since the middle of the last century. Nuclear power is thus a significant non-fossil based energy source that does not add to the already substantial “greenhouse gas” inventory of the planet. The world’s fleet of NPPs, with nearly 30 currently under construction is given in Table 1. Figure 1 shows the age of operating NPPs world-wide. Accordingly, much experience has been gained with respect to materials development, design and manufacturing of SSCs, operational practices and NPP management. The fleet’s current cumulative operating experience exceeds 11’500 years, which is a significant pointer to the maturity of the industry. Diverse designs of NPPs have been successfully commissioned and put into commercial operation, the main types being light water pressurized and boiling water reactors of American, European and Russian design, and the heavy water moderated types of Canadian design. Other NPP types feature gas (carbon dioxide or helium) and liquid metal (sodium) cooling. Designs capable of generating higher power with more passive safety features are going to be put into operation in the future. It is thus clear that a wide selection of materials and operational characteristics are features of the current and projected NPP fleet. Today’s situation shows that the average effective full power operation years (EFPY) with respect to all NPPs is around 24 years, and thus, a number of older NPPs (e.g. >30 years EFPY) in the fleet’s age distribution are approaching their design life (usually 40 EFPY), or end of current licence to operate. However, this does not necessarily translate into the end of safe operational life due to the fact that utilities, owners and operators have continually refurbished and improved SSCs, back-fitted safety and other features, optimized operational conditions and have implemented technologies gained from experience as well as having effectively used results from research concerning the performance behaviour of materials in the nuclear environment. Regulatory requirements have also enforced many aspects of NPPs operation and monitoring for enhanced safety. In fact, many NPPs may now be technically in a superior condition compared to when they were first commissioned. Given that safety is
maintained, and continually improved, it is economically attractive for utilities and owners to continue operation of older NPPs, especially when plant amortization has been achieved. Continued operation, always with the pre-condition of satisfactorily maintained licensing conditions (with either re-licensing or periodic safety review (PSR) approaches), thus requires that the NPPs are always managed primarily to fulfil safety goals. However, even when a safe plant cannot make a profit, the commercial aspects will dominate, leading potentially, to shut-down and decommissioning. Market competitiveness of nuclear generated power, compared to other sources, factoring in the safety, reliability of supplies and increasingly important environmental considerations, will be an important aspect for the future of nuclear power. The overall life of a NPP is influenced by various factors such as, technical and regulatory constraints, life-time of non-replaceable components, a major accident, wherever it happens, lack of cost-effectiveness, absence of industrial support, lack of skilled personnel and loss of public acceptance.

Many SSCs are directly or indirectly crucial to safe operation of NPPs, so appropriate PLiM programmes can be perceived as not only favouring reliability and economic aspects, but also safety goals and improvements. PLiM is a methodology whereby all expenses are optimized for profitability and competitiveness, whilst providing safe and reliable supplies of electricity. Furthermore, many NPPs are aiming for long-term operation, aided by PLiM strategies. Long-term operation (LTO) may be combined with the issue of power uprates, which is a potential core issue for the nuclear industry in future. It will necessitate on-going and comprehensive updating of ageing management and associated PLiM programmes to facilitate monitoring and mitigation/elimination of any negative impacts that power uprates could have on the SSCs.
LIFE ASSESSMENT OF PRIMARY HEAT TRANSPORT SYSTEM FEEDERS AT EMBALSE NGS

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In this report of Life Assessment the main features of Primary Heat Transport System Feeders, the changes introduced in these components along the operation time of the plant, as well as their materials and the environment details are summarized. The operational history is analyzed by taking into account the deficiency reports, engineering reports, event reports, and external reports. Also, the programs of Preventive and Predictive Maintenance, and In-Service Inspection are analyzed compared to the recommended practices. All the gathered information is used to determine the possible degradation mechanisms according to every component, material and environment. Then, the risk that represents every possible degradation mechanism is determined, by analyzing the way of controlling them by means of the current programs at CNE. Finally, there is a conclusion about the condition and a life prognosis is also made. From this, some recommendations - which are given - arise. It is concluded that the most important aging related degradation mechanism, which could be a life-limiting factor for feeders is the flow-assisted corrosion. It is probably that feeders could reach safely the end of life design. Their replacement will be necessary to deal with a future life extension.

Another important subject for this Life Assessment are feeder supports, which are also analysed.
REFURBISHMENT OF MAIN CONDENSER CIRCILATING WATER PIPING OF EMBALSE NPP

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During periodic inspections realized at different Annual Outages of Plant, it was observed an increasing presence of corrosion products in the inner surface of piping, like sediments and microbiologic sludge covering this surface. Under some bubbles was liquid like product of sulphato-reducer bacteria.

Besides, quantity and sizing of pitting signals were increasing along service years and it was augmented the microorganisms colony which produce bubbles on the surface covered by additional bivalves colony. When these bivalves colony falls down it was observed a ferric oxide layer.

It was decided to apply an epoxy bituminous paint layer. The surface was prepared previously trough an abrasive sand jet and crushed steel in order to obtain previously the rigidity specified before to apply the zinc silicate ethyl. Finally a layer of epoxy bituminous paint layer (400/600 micron) was applied.

Some surface porosity special tests are described and tests of non visual contamination products, pores detection and painted surface final approval. At the next Annual Outage the painted surface must be verified trough visual inspection, thickness and porosity measurements for surveillance of surface condition.
INVERTER SYSTEM REPLACEMENT AT EMBALSE NPP

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The failure historical data relative to single phase inverter system of bar 220VAC 5542 BUC has registered some minor failures but one significant event. On this significant event the bar 5542 BUC was remain without power supply because the input filter fuses of thyrstorn bridge blow out. Immediately a failure of static transfer switch 5542 VR-C failed in transfer operation. Then, transfer process to the network supply was not successfully due to control and power electronic failures.

The consequence was an impairment of one channel of Reactor shut-down system and one channel of Plant process control measurements. Besides, the Turbine electro-hydraulic control panels remain without power supply and then turbine control and Main Condenser by-pass valves were tripped.

The event analysis was done and one of the corrective actions was the replacement of original equipment by a new automatic inverter/switch system.

A new technical specification was developed taking into account the international nuclear guidelines, external operating experience and corresponding standards.

A new supplier was selected and a reference of successfully performance at nuclear plants was required and the limits values for over-voltage protection was verify regarding the Foshmark I at 2006.
REPLACEMENT OF FLUX DETECTORS AT EMBALSE NPP

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The work began replacing, in first place, the units with flux mapping detectors only, taking care not to replace detectors of the reactor regulation system and safety systems in order do not have an impairment on the safety systems in case of replacement program’s interruption during the first steps of replacements.

During the phase of connectors installing in the seventh vertical flux unit (7th VFD) the original Twinax cable was found with a lack of its original properties. The internal conductor insulation was stiffened. For that reason, the internal conductor insulation became very difficult to handle without possible cable damage.

Immediately, it was decided to stop the detectors replacement because the general schedule of the outage must delayed by the additional unexpected task of cable replacement. Besides, properly replacement cable was not available at de plant.

A series of insulation measurements were made on the detectors cables, including those which were not replaced. A periodical control of insulation resistance measures of detectors and cables was arranged to be carried out during normal operation, until the next outage. In the next Annual Outage the detectors replacement was completed and the PVC cables was replaced by KKK cables with EQ requirements

At the present a decision making process must be realized taking into account the Long Term Operation of Plant
CONDITION ASSESSMENT OF MAIN TRANSFORMERS AND REPLACEMENT CRITERIA

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Embalse NPP, a 24 years plant, is connected to the 500Kv national interconnected system through a bank of three single-phase transformers of 255 MVA each, with a strain relation of 22KV / 500KV

Considering the importance these equipments have in the plant’s safety and availability, as well as the economic loss a failure in one of the transformers would cause, a follow up of the status of the transformers has been carried out for several years now, in order to assess the remaining life.

The following routine electrical assessments were performed following the procedures recommended in the IEC, in SOER 2003 (WANO) and by the manufacturer: ohmic resistance, power factor (Tg delta), insulation resistance, , and FRA.(Frequency Resolution Analysis) Chemical analysis were also performed on a six months –basis, and on a yearly basis to determine the oil condition, possible combustible gas generation, and the polymerization degree of insulators. to determine furans presence..

The first Condition Assessment of the transformers was performed by an international transformers manufacturer. This manufacturer based his assessments in the documents on the oil and electrical assessments provided by Embalse NPP.

In 2003, while the Condition Assessment was being performed, an increase in the value of the combustible gases was observed in the Unit, during the T phase. Upon observing this situation Embalse NPP management decided the replacement of the transformer, and located the back up transformer in its place. This transformer had been manufactured in 1979 and had never been energized.

In May 2004, routine assessments were performed in the back up transformer, in order to ensure its safe connection to the grid. These assessments comprised: ohmic resistance, insulation resistance, power factor, capacity factor, as well as other assessments as applied strain and induced strain (Vn x 1.12), measuring partial discharges for 24 hours and oil tests. As the outcome of these assessments was satisfactory, during the 2004 planned outage, the T phase transformer which had a failure, was replaced by the back up transformer. This transformer has been connected to the 500 Kv grid without inconveniences so far. The chemical tests performed in the insulation oil have normal values.

Some routine assessments were also performed on the replaced transformer in order to evaluate its condition, together with an inspection. The active part was detanked for a better observation. Zone heating with important carbonization was observed, In the AT output connection in vacuum switch, partial internal discharges, especially in the winder regulation, and no rigidity in the winding axial supports.
During this inspection, an insulating paper sample was taken in order to produce more accurate assessments on the paper polymerization degree.

Deficiencies in the transformer were repaired, the connection in the vacuum switch was modified, and it was finally tanked. The transformer was then operative. Oil was treated with hot oil spray until the humidity values in oil had acceptable values.

Last, but not least, the electrical routine assessments were performed, especially with induced strains, measuring partial discharges (strain=1.05 Vn) for 24 hours. Results were satisfactory, and the transformer is currently kept as back-up.

As a result of the assessments performed in the transformers, and considering long term operations analysis, two alternatives were studied in order to have at least one transformer in stand-by situation:

1. In-site refurbishment of the transformer
2. Purchase of new transformer

The first choice has a visible disadvantage: Embalse NPP doesn’t have a back up transformer for a period longer than a year. This means an important risk, due to the high chance of a failure in a transformer. Should this happen, Embalse would have to be taken out of service for a long time, thus producing economic and social damage.

The second choice, although more expensive, means having a back up transformer permanently, and it allows in-site repair of the transformers already installed.

This is Embalse NPP choice, so the technical specifications for an international bid are currently being prepared since, unfortunately the original designer no longer exists and there are no technical documents on calculus and design at the plant.
MODERNIZATION OF INSTRUMENTATION IN WATER DEMINERALIZATION PLANT AT EMBALSE NPP

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The demineralization water plant of Embalse N.P.P. has a Main Control Room which has installed different panels: mimic panels, controllers panels, and operation panels for operators.

The plant process water lake in order to obtain demineralized and potable water for the Plant. The different process are automatized by means relays logic and analogic controllers which technology was designed at 70’s years.

Because obsolescence of electric and electronic components and failure frequency of different components specially relays and step programmers it was decided to initiate the modernization of the operation panels related to demineralization process. Many advisory notes were exchanged with the original supplier to define the general guidelines of new components and corresponding associated software.

In order to minimize the outage time of the water plant a special schedule was followed. Final test for approval were done with simulator equipment at the supplier laboratory in order to verify the software customization. Participation of different specialists from Engineering and Maintenance departments in different steps of project was done to obtain the proper level of training before start-up.

The plant is not designed under nuclear standards but the availability must be assured with conventional standards because it supplies demineralization water to main feed-water systems.

After the period of time at normal operation new cost-benefit analysis is done to complete the modernization process of main control room.
HELICOIDAL SPACERS INSTALLING IN THE MODERATOR HEAT EXCHANGER AT EMBALSE NPP

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The installing of helicoidal spacers in both moderator heat exchangers 3311 HX1 /HX2 is described.

The operating experience data of both heat exchangers indicated a high level of vibrations due to the induced impacts by refrigeration water flux at secondary side. This vibrations level was reduced in order to avoid or to minimize the possibility of tubes damage resulting in a heavy water (with high isotopic level) leak to the environment. Vibrations analysis was done with flux distribution determination with the vein analysis of the HTRI (Heat Transfer Research Institute). This method finds the fluid vein fraction corresponding to the fractional velocities of different zones of heat exchangers.

The analysis was focused to the surface zones close to the tube U-bend and the by-pass partition by-pass because they are under the conditions with highest velocities of cross-flow. The vibration induced phenomena is normally due to vortex, turbulence, instability, elastic flux. From results it concluded that the tubes of the by-pass lane, of U-bend and in the surface of the tube mallet are the most sensitive to the flux induced vibration.

The tubes were structurally modified, trough additional supports installing with helicoidal spacers inserted in the tubes mallet. Practically there are not changes in the flux and the natural frequency of tube is augmented that is the mallet resistance to the flux induced vibration.

The heat exchangers structurally modified are more resistant to tube vibration by at full rate flux and to 125% of full rate flux. After modifications vibrations tests were done in the heat exchangers without evidence of pulse signals or frequency variations due to impacts flux in the equipment.

Conclusion is that the modification gives a proper supporting and avoids a damage of tubes mallet.

An analysis about proper surveillance program for the heat exchangers is opened.
MANAGEMENT ACTIVITIES FOR LONG TERM OPERATION OF ATUCHA II NUCLEAR POWER PLANT

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The argentinean atomic agency – Comisión Nacional de Energía Atómica (CNEA) - is a state owned research and development institution that has the following functions among its responsibilities:

- To permanently update the argentinean NPPs technological information in all operation, commissioning and decommissioning stages, in order to lead to the most efficient use of the plants.
- To implement basic and applied research programs in nuclear related science and technology disciplines.
- To encourage the generation of highly qualified human resources; as well as the development of nuclear science and technology, including the promotion of technology innovation programs.
- To establish cooperation agreements with foreign countries in order to develop the previously mentioned technology innovation programs, as well as to contribute to the development of the atomic fusion technology, throughout the Argentinean International Relationships Ministry (Ministerio de Relaciones Exteriores, Comercio Internacional y Culto).
- To establish fluid and fruitful relationships with foreign institutions aimed to similar objectives.
- To make agreements with argentinean utilities in order to develop research programs
- To provide services required by nuclear generation utilities, or by any other nuclear facility.

Some of these activities are carried out by the Plant Life Management (PLIM) / Plant Life Extension (PLEX) group by means of contracts celebrated with the plants operators and by the participation in regional and international working groups. Besides, the incorporation and capacitacion of personnel in the area is permanently encouraged.

In spite of the narrow and fruitful relationship existing between CNEA and the utilities; it is important to consider that CNEA aims to develop its own criteria regarding its responsibility about “The update of the argentinean NPPs technological information in all operation, commissioning and decommissioning stages, in order to lead to the most efficient use of the plants”. This independent criteria stands for a wider view of the applied strategies, and for the long term activities that sometimes cannot be attended by the constructor or the operator; which are urged by the day by day issues, as well as by the economical expectancies. For these reasons, a Plant Life Management and Long Term Operation (LTO) program is proposed for Atucha II Nuclear Power Plant.

Atucha II is a 700 MWE heavy water pressure vessel reactor. The commissioning of the plant has been delayed for a long period of time, and the activities were retaken in 2006.
Despite the plant design is 30 years old, it would be very important to achieve a long term operation of 50 – 60 years. This requires a comprehensive management plan according to the new situation. The plan must be able to take into account the possible aging effects that could have on the equipment the very long storage time.

Having said that; it is important to carry out a thorough analysis of the critical components, as well as a base line data gathering. The collected information will be essential for assessing the current state of the equipment, as well as for the confection of future degradation studies. Doubtlessly, the most important component is the pressure vessel; for which an especial surveillance program has to be developed, in order to allow the operation of the plant for the expected period of 50-60 years.

The proposed management plan is in agreement with international practices, and aims to the safe and economy operation of the plant for a long period of time. The first phase of the plan includes:

- Selection of critical components (Component Screening)
- Determination of the existing relationships between equipment materials and possible degradation mechanism (i.e. Aging Related Degradation Mechanism Matrix – ARDM) for 3 critical components.
- Determination of the necessary information for the future monitoring of the critical components.
- Database design and implementation.
- Participation in the design of the pressure vessel surveillance program.
PIPE WHIP RESTRAINTS-PROTECTION FOR SAFETY RELATED EQUIPMENT OF WWER NUCLEAR POWER PLANTS

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Keywords: Nuclear power plant, High energy piping, Pipe rupture, Pipe whip restraint

The nuclear power plants of the WWER type operate for about three decades in several countries. In the Czech Republic, the Dukovany NPP of the VVER-400213 type has been put into operation in 1985. A steady very high level of plant’s operational safety, availability and reliability has been unambiguously proved. An effective way of managing updated safety requirements ensures a high safety level for the whole plant’s design life. In accordance with general trends, operation of the plant in excess of its design life is assumed. An effective ageing management of systems, structures and components is decisive for maintaining operational safety during the extended plant’s operating lifetime. Moreover, due to steadily increasing additional requirements, the plant’s operational safety has to be enhanced. Besides many other important problems, those related to the safety of NPP system piping have to be considered.

The contributed paper deals with problems of enhancing the protection of WWER NPP against the effects of pipe break. The selected problem is formulated as follows: A circumferential break in high energy piping results in a pipe whip. The protection of nearby safety related systems and components are ensured by a pipe whip restraint provided with energy absorbing visco-plastic structural elements. The number, arrangement, dimensions and properties of elements are optimized using results of iterative non-linear dynamic analyses of the piping system response to fluid forces.

The results of the presented analyses have been applied in the design of whip restraints for the NPP Temelin (WWER-1000 Model 320) and NPP Dukovany (WWER-440213), among others.

The majority of the series of dynamic response analyses has been devoted to whip restraints for NPP DUKOVANY. Particularly, the whip restraints located at the main hermetic penetrations have been analyzed. For one production block involving 6 steam generators, 6 whip restraints of feedwater piping and 4 restraints of steam piping have been analyzed in detail. The 18 restraints of remaining production blocks have been analyzed in a simplified way. Versions of whip restraints for tee-pipes have been analyzed for 10 postulated breaks. For NPP Temelin 8 whip restraints have been analyzed in detail. The analyses have respected relevant clauses of ANSI/ANS-58.2-1988 Standard and IAEA 1996 Safety Issues for WWER NNP. Two analyses out of those, prepared for the presentation are described below.

For dynamic response analyses of the NPP Dukovany steam piping system with whip constraints, a spatial computation model with up to 343000 DOF has been developed using the FEM based COSMOS program package. The model includes the complex of all interacting steam piping and steel structures of the selected main production block of the
plant. Large diameter pipes have been modeled using shell elements with material density values selected so as to attain overall mass distribution corresponding to operating conditions. Branch pipes, piping supports, valves etc. have been modeled using spring, beam and mass elements. The RC structures and steel structures (incl. restraint at the hermetic zone wall penetration) have been modeled using combination of shell, beam, spring and solid elements.

The whip response analysis starts by increasing gravity, pressure and temperature loads from zero up to steady state values. During the break opening time, using a set of governing time curves, stress resultants in the postulated pipe section are decreased to zero. The effect of the steam streaming out of the broken pipe is introduced by both varying the pressure and adding a system of variable external forces at the ruptured pipe ends. These loads depend on the gap width between the broken pipe end sections and on the time as well. The forces developed by the restraint in function depend on relative displacement and relative velocity of the restraint ends. For an actual whip restraint construction, the respective characteristic functions are determined experimentally. The restraint is modeled using a set of special finite elements, the properties of which fit the given characteristics of the restraint.

The initial response analysis of the piping model without restraint is used to estimate the extent of potential damages due to the pipe whip. The next analysis with preliminary designed restraint in function shows the attained limitation of the pipe whip and yields the data required for an appropriate design modification. A series of repeated analyses leads to optimization of characteristic functions of the whip restraint.

The performed analyses have shown, that the modeling of the high energy piping complex as a whole - sometimes even with supporting RC structure included - represents the fundamental condition to get a reliable dynamic response to a pipe break event. Response analysis using separated piping system parts should be avoided.

In the second selected case a special pipe whip response analysis is described. It has been performed in order to assess the combined effect of earthquake event and pipe rupture. The analyzed system includes reactor coolant pump, steam generator and attached pipe branches (WWER-1000 NPP). A pipe whip restraint using axial viscous elements is located at the steam pipe - steam collector weld. The restraint is provided with a protective tube limiting the lateral whip. Seismic motion of the containment base is described by acceleration time histories. The break in the weld is postulated to occur at the instant when the vibration of the system due to seismic excitation attains the highest level. The pipe break causing the whip is assumed to be initiated by a random increase of the pipe operating pressure. The FEM-based computer program SYSTUS is used for the response analysis, taking into account large deformations and non-linear material properties of the model. The program library offers a special element, the stiffness and damping coefficients of which may be defined as dependent on the relative displacement and velocity of its end nodes, respectively. Such elements, inserted between shell elements modeling the steam pipe weld zone, are used to simulate the process of circumferential crack development leading to pipe break opening.

The same finite elements are used to model the viscous elements of the whip restraint. The performed response analysis has proved that properly designed whip restraint with a protective tube ensures the seismic resistance of a broken pipe, too.
DESIGN REVIEW OF THE VVER – 440 – NPP MAIN COMPONENTS ON THE BASE OF ASME BRVC

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The presentation deals with the ongoing structural analysis of VVER-440 reactors in Paks, Hungary in the framework of the project for the life time and capacity extension of the nuclear power plant. The main purpose of the calculations is a comprehensive design review of the main components on the base of the internationally accepted ASME BPVC Division III; in order to support the introduction of the ISI methodology required by ASME XI. As source, quite scant design documentation was available. The new complex analysis creates a gateway between the Russian PNAE G-7-002-86 and the ASME Code, and reconsiders the scant design calculations. All the input data – geometric data, material properties, loads – for the RPV, the vessel head, and the components of the core support structure have been collected, and FEM models have been created, which follow the original geometry with high accuracy. The structural calculations are performed for all the loads and operational cycles of the units taking into account the planned 50 years of operation. The results of the new fatigue calculations will be used as TLAA for supporting the technical possibility of the life extension. The calculations will facilitate to decide if the critical sections of the components are those, that were known originally, or there are further ones. The newly constructed computation models and the detailed uniform documentation of all the main components will be reliable and comprehensive tools to solve any possible structural analysis related issue in the period of extended operational life of the NPP Paks.
PRELIMINARY COMPREHENSIONS ABOUT RPV AND RVI AGEING MANAGEMENT AND ASSESSMENT TECHNOLOGY

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The ageing mechanisms of PWR RPV and RVI include irradiation embrittlement, thermal Ageing, fatigue, corrosion, mechanical fretting/wear, etc. Aging Management of PWR RPV and RVI is a complicated process, which includes understanding of ageing mechanisms, aging related information, methods of aging data collection, methods of analysis and assessment. Aging assessment technology includes irradiation embrittlement assessment, fatigue assessment, corrosion assessment methods, etc. This paper summarizes aging susceptibility analysis methods and Time-limited analysis methods. This paper separates ageing analysis and assessments into two aspects: one is susceptibility analysis, the other is Time-limited analysis. The later is deeper analysis than the former. This paper summarizes aging management related information, and describes some experiences from related aging susceptibility analysis for Daya bay NPP performed by the author and his colleagues.
RAPID DETERMINATION OF HISTORIES OF SIF DISTRIBUTIONS ALONG 3-D CRACK FRONTS OF RPV SUBJECTED TO PTS BY UNIVERSAL WEIGHT FUNCTION AND FINITE VARIATION METHOD

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The 3-D thermal weight function (TWF) method\(^{1-4}\) is extended to a more general form, universal weight function (UWF) method, which can be used for efficiently numerically determining the histories of SIF distributions along 3-D crack fronts of a RPV subjected to both mechanical and thermal loading in pressurized thermal shock (PTS). The UWF is also a universal function, dependent only on the crack configuration and geometry, and independent of “loading” and time. Once the UWF for a specific crack and body geometry is determined, the SIFs for that cracked body subjected to any loading (distribution of temperature and mechanical loads) can simply be calculated through integration of the products of the UWF and “loads”. The repeated determinations of the distributions of stress (or displacements) fields for individual loads and time instants are avoided in the UWF scheme. The amount of computation can thus be greatly reduced.

A variational integral expression of the basic equation for the 3-D UWF method for Mode I in an isotropic elastic body is proposed.

\[
\int_{\Gamma} \frac{2K_i^{(1)}K_j^{(2)}}{H} \delta_{ij} \alpha(s) ds = \int_{\Sigma} u^{(1)} \cdot \delta_{ij} \alpha^*(\tau) dS - \int_{\Sigma} u^{(2)} \cdot \delta_{ij} \alpha^*(\tau) dS + \int_{\Sigma} f^{(1)} \cdot \delta_{ij} \alpha^*(\tau) dV + \int_{\Sigma} f^{(2)} \cdot \delta_{ij} \alpha^*(\tau) dV
\]

\[+ \frac{E}{1-2\nu} \left[ \int_{\Sigma} \alpha \delta_{ij} \alpha^*(\tau) \delta_{ij} u^{(1)} \cdot ndS - \int \nabla \alpha^*(\tau) \cdot \delta_{ij} u^{(1)} dV \right] \]

It can be understood from this equation that the so-called TWF is in fact coincident with the mechanical WF (MWF) except for some constants of elasticity.

The multiple virtual crack extension (MVCE) method, which was proposed by authors\(^{1-4}\) to numerically solve the 3-D TWF equations, is extended to a more general form for solving the basic equations in the 3-D UWF scheme, which will be renamed as finite variation method (FVM) from its mathematical meaning. Finite variation (virtual crack extension, VCE) modes are introduced for solving the UWF equations. In the mean while, finite linearly independent special “shape functions” are introduced for the discretization of unknown SIFs. Finally, a system of linear equations is obtained.

\[
\sum_{i=1}^{N} A_i \int_{\Gamma} \frac{2}{H} \omega_i(s) \left[ K_i^{(1)} \right] \omega_j(s) g(s) ds =
\]

\[
\left\{ \int_{\Sigma} \frac{\delta u^{(1)}}{\delta \alpha} dS - \int_{\Sigma} \frac{u^{(2)}}{\delta \alpha} dS + \int_{\Sigma} f^{(1)} \cdot \frac{\delta u^{(1)}}{\delta \alpha} dV + \right. \]

\[
\left. + \int_{\Sigma} f^{(2)} \cdot \frac{\delta u^{(1)}}{\delta \alpha} dV \right\}_{\delta_{ij} a_j}
\]
Solving this system of equations and from interpolation of unknowns, the SIF distribution along crack fronts can be determined.

Iso-parametric technique was introduced by selecting the “shape functions” directly from the VCE modes. When the VCE modes were selected as linear, the coefficient matrix of the final system of equations would be a triple-diagonal matrix with good numerical properties. The number of VCE modes, which can be introduced in a problem, is unlimited. Complex situations during PTS, in which the SIFs vary dramatically along the crack fronts and the variation history of SIF distributions is complex, can be well simulated by the FVM.

A new procedure based on the self-consistency condition of the FVM was also introduced for conveniently and accurately extracting the reference SIF distributions along 3-D crack fronts. From the viewpoint of energy balance along crack fronts, the SIF distribution extracted by this procedure could be regarded as the “most acceptable and accurate” numerical estimation corresponding to the current finite element discretization of the body. Several examples including a reactor pressure vessel (RPV) subjected to a typical PTS (Rancho Seco PTS) was analyzed, and the results were compared with available literatures. Examples show very high efficiency and good accuracy.
FIG. 1. Histories of SIF distributions along crack fronts

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ROLE OF RESEARCH IN MATERIALS DEVELOPMENT, MITIGATION STRATEGIES AND NON–DESTRUCTIVE EVALUATION FOR PLANT LIFE MANAGEMENT (PLiM) IN THE INDIAN NUCLEAR POWER PROGRAMME

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Steady gains in reliability and safety of systems, structures and components (SSC) in Indian nuclear power plants over the last 50 years are mainly due to the use of improved materials, adopting stringent quality assurance measures, better understanding of degradation mechanisms through systematic research and incorporating advanced techniques for in-service inspection. Focussed research resulted in development of appropriate strategies to mitigate or eliminate service induced degradation in SSC enabling implementation of effective plant life management (PLiM).

The successful design, construction and operation of the fast breeder test reactor (FBTR) at Indira Gandhi Center for Atomic Research (IGCAR), demonstrating the technological viability of fast breeder reactors (FBRs) has paved the way for stepping into the commercial phase of the second stage of the nuclear power programme [1]. This success is as a result of comprehensive multidisciplinary R&D programmes addressing every aspect of design, material development, manufacturing technology, corrosion, welding, equipment development, chemistry, reprocessing, quality assurance, in-service inspection, economy and efficiency and safety. IGCAR has established collaborative research with academic and research institutions and adopted a variety of approaches to develop human resources and also to enhance the quality of human resources.

In the Indian nuclear power programme, great importance is given to development of new materials and improvement of existing materials, material properties and fabrication methodologies, e.g. zirconium alloys for PHWR applications, D9 alloy for FBR applications and Ti-alloys for fast reactor fuel reprocessing applications. In order to identify the optimum microstructure with the best combination of corrosion resistance and mechanical properties, several investigations have been carried out on Ti-alloys and it has been established that heat treatment at 1000° C followed by water quenching results in best corrosion resistance. R&D has been focussed on grain boundary engineering to develop the methods to increase the amount of special or crack resistant boundaries in 9Cr-1Mo ferritic steel by suitable thermo-mechanical treatments, to minimize embrittlement.

In the context of innovative reactors with 100 years design life, development of advanced fuels with higher burn-up up to 200 GWd/t are attractive and more economical and thus, fuel development is an important aspect in PLiM. A vast irradiation experience is available in the case of oxide fuels and considerable experience has been gained in the development, post-irradiation and reprocessing of carbide fuel. Helium bonded carbide fuel pins have been successfully irradiated in FBTR to a burn-up of more than 150 GWd/t. When high fissile
density, thermal conductivity, high breeding ratio and high burn-up are targeted, metallic fuels are a better choice for fast breeder reactors. Accordingly advanced austenitic stainless steels, ferritic steels and oxide dispersion strengthened (ODS) alloys are attractive for cladding and wrapper materials. Research has been focused on development and fabrication of sodium bonded metallic fuels and on development of advanced materials.

One of the most important issues in PLiM, i.e. ageing management and economic operation, is the condition assessment of systems, structures and components in NPPs. This is facilitated by in-service inspection (ISI) using NDE techniques. The necessary NDE expertise, techniques and procedures have been developed, in many cases to meet the challenging demands of the ISI programmes of PHWRs and BWRs [2]. Further, a number of mitigation and repair technologies have been developed. Some of these include feed water nozzles of BWR, en-masse coolant channel replacement of PHWR, repair of the end shield of a PHWR and rehabilitation programme for overcoming the problems faced in the manifold system of two PHWRs. For SSCs of FBRs, systematic procedures for ISI of main vessel, safety vessel and steam generators have also been developed For example, for the ISI of the core support structure shell weld (Figure. 1) of the 500 MWe fast breeder reactor (PFBR), an innovative ultrasonic inspection procedure has been developed.

This paper highlights as to how focussed research has provided understanding of material behaviour under operating conditions, thus leading to the development of strategies to mitigate or eliminate service induced degradation mechanisms associated with neutron irradiation, thermal ageing, fatigue, creep, corrosion and fretting in SCs in Indian NPPs. It is explained how this research provided vital inputs to develop robust life prediction methodology upon integrating the inputs from field experience, laboratory experiments, modelling and simulation studies. A few examples drawn from the first and second stages of the Indian nuclear power programme highlighting the materials development, fuels, mitigation strategies and ISI technologies are presented with a clear emphasis to the future technologies relevant to the Indian nuclear programme.
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THE USE OF ROTOR DIAGNOSIS FOR THE ANALYSIS OF HIGH VIBRATION EXPERIENCE AT TURBINE GENERATOR SYSTEM IN NUCLEAR POWER PLANTS

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As the life extension of operating nuclear power plants becomes bigger issue, structural integrity of NSSS component is getting more concern. Along with the NSSS component, turbine-generator (as one of the biggest rotating component in nuclear power plants) can be considered seriously, due to long period of operation. Many cases of high vibration have been reported. Some experiences of turbine component failure in US and Korean nuclear power plants are shown. For example, Figure 1 shows a case of turbine failure; low pressure turbine rotor blades with one blade missing due to poor design and lack of the consideration of installing environment. The reason for the failure, mechanism of these turbine failures can be listed as;

- Poor design

- Lack of the consideration of turbine installing environment

- Improper maintenance

- Improper procedure and operation

FIG. 1. Low pressure turbine rotor blades with one blade missing
The vibration data from main turbine component in operating nuclear power plants are analysed in view point of rotor diagnosis. *TABLE. 1* shows examples of high vibration of main turbine at some Korean nuclear power plants and the reason of each high vibration case. From the analysis result and the experience data, the optimum operation and maintenance method is proposed by using the vibration data and analysis result.

**Table 1. Example cases of high vibration and damage of main turbine at some Korean nuclear power plants.**

<table>
<thead>
<tr>
<th>NPP unit</th>
<th>Date</th>
<th>Symptom</th>
<th>Reason of high vibration</th>
</tr>
</thead>
<tbody>
<tr>
<td>NPP-A</td>
<td>1989.11</td>
<td>LP TBN High vibration</td>
<td>Damage at LP TBN rear diaphragm stage-2.</td>
</tr>
<tr>
<td></td>
<td>2005.06</td>
<td>LP TBN BRG High vibration.</td>
<td>Damage and loss of blade by aging due to long term operation.</td>
</tr>
<tr>
<td>NPP-B</td>
<td>1985.11</td>
<td>TBN exciter BRG High vibration.</td>
<td>Electric erosion at exciter BRG.</td>
</tr>
<tr>
<td>NPP-C</td>
<td>1984.03</td>
<td>LP TBN High vibration</td>
<td>Blade damage at stage-5 of LP TBN.</td>
</tr>
<tr>
<td></td>
<td>1994.11</td>
<td>LP TBN BRG High vibration.</td>
<td>13 cracks found at root of LP TBN blade.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Crack found at LP TBN stage-4 disc rim.</td>
</tr>
<tr>
<td>NPP-D</td>
<td>1986.07</td>
<td>Step change in vibration amplitude and phase.</td>
<td>Loss of a LP TBN blade at rear stage-5.</td>
</tr>
</tbody>
</table>
THE SPECIFIC SURVEILLANCE PROGRAM IN THE PLANT LIFE MANAGEMENT OF LAGUNA VERDE NPP

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The reactor pressure vessel (RPV) is the most important component for the operation in the nuclear power plants (NPP). The RPV embitterment, produced by the neutron irradiation during operation of the NPP, must be to monitor along the plant life operation, this embitterment is measured by the ductile-brittle transition temperature increase. The DBTT shifts regulations, require very low RPV failure probabilities both for normal operation and primary defect postulated accident events. The surveillance programs are the best method for monitoring the changes in mechanical properties of RPV materials. The surveillance programs are conducted in accordance with mandatory documents like appendix H to 10CFR part 50.

The original surveillance program is established in the beginning of operation and is applied along of lifetime of the NPP, for Laguna Verde NPP the surveillance program was performed by ASTM E-185 82. However changes in the operation conditions or operation problems like cold hydro and leak test make necessary incorporate modifications in the original surveillance program for assurance a more precise RPV integrity evaluation.

Laguna Verde NPP increased the power 5% in 1999. At this moment the utility is studying the possibility to make extended power up rate (EPU). The EPU will modify the neutron spectrum and will increase the flux and, as consequence, the damage in the RPV material will change. Therefore, with the implementation of the EPU will make necessary to performance a specific monitoring of the RPV materials proprieties in LVNPP unit 1 and 2. The performance of the specific surveillance program has to include both the original program and supplementary monitoring that can support the safety operation during design life and the possibility of license extension. The supplementary monitoring involves additional capsules and testing. The standard guide E 636 covers the test methods and procedures that can be used in conjunction with, but not as alternatives to, those required by Practice E 185 for the surveillance of nuclear reactor vessel. The supplemental test methods outlined permit the acquisition of additional information on radiation induced changes in fracture toughness, notch ductility, and tensile strength properties of reactor vessel steel.

By the other hand, the operational procedure to make the hydro and leak test needs to use the P-T curves to restart the reactor operation; in the case of Laguna Verde NPP the P-T start up field to achieve this testing has been narrowed with the time of operation. The problem to operate in a narrow field is, in some cases, the long time to set up testing or operations conditions.

The specific surveillance program has to consider irradiate reconstituted specimens from previous capsules, supplementary or similar material, dosimeters, temperature monitors, precracking Charpy specimens for fracture toughness testing. In addition to surveillance
The utility has to consider too the possibility to apply alternative requirements for developing pressure-temperature operating limits. Appendix G of 10 CFR Part 50 specifies that the requirements for these limits are based on the application of evaluation procedures given in Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code.

Since ASME Section XI, Appendix G, was first developed, significant advances in fracture mechanics and the analysis of reactor vessel integrity have been achieved. In general, advancements in knowledge are promulgated as ASME Code Cases like N-588, N-640, and N-641, which provide alternatives to existing Code requirements for developing pressure-temperature operating limits or the Code Case N-629 where to indicate that fracture toughness test data may be used as an alternative to the methods specified in Appendix A, A-4200, and Appendix G, G-2110 to establish fracture-toughness based reference temperature, \( RT_{T_0} \), for pressure retaining materials, in that document requires that the fracture toughness testing for specific base metal or weld materials shall be performed in accordance with ASTM E 1921-97, “Standard Test Method for the Determination of Reference Temperature, \( T_0 \), for Ferritic Steels in the Transition Range,”. This standard uses the master curve methodology to determine \( T_0 \) temperature and the fracture toughness test specimens could be integrate in the surveillance program. The application of the code cases to develop the P-T operating limits curves and structural integrity assessment for RPV will be less conservative but with good safety margin.

The original Laguna Verde NPP RPV surveillance program has the follow schedule

1st outage capsule 30º azimuth withdrawal dosimeters

6 EFPY. Withdrawal capsule 30º azimuth 36. Charpy specimens (12 base metal, 12 welding metal, 12 HAZ), 10 tension specimen, (3 base metal, 4 welding metal, 3 HAZ), neutron flux measurement, developing of P-T curves.

15 EFPY. Withdrawal capsule 300º azimuth. 24 Charpy specimens (8 base metal, 8 welding metal, 8 HAZ), 6 tension specimen, (2 base metal, 2 welding metal, 2 HAZ), neutron flux measurement, developing of P-T curves.

32 EFPY. Withdrawal capsule 120º azimuth. 24 Charpy specimens (8 base metal, 8 welding metal, 8 HAZ), 6 tension specimen, (3 base metal, 3 welding metal, 2 HAZ), neutron flux measurement, developing of P-T curves.

Specific Laguna Verde NPP surveillance program

Schedule until 2011
Year 2007. Supporting documentation to modify the surveillance program.
Year 2008. Capsule 300º azimuth withdrawal
  Dosimeter.
  Mechanical Testing.
Year 2009. Fabrication of new capsule with reconstituted specimens, dosimeters and temperature monitors.
Year 2010. Installation in the reactor unit 1 the new capsule.
Year 2010. EPU
Year 2011. Dosimeters withdrawal.
This paper will present the current status of Laguna Verde RPV surveillance program and the design of specific plant supplemental actions to monitor the embitterment of RPV materials to be used in the plant life management of Laguna Verde NPP

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CONTINUING SYSTEMS/COMPONENTS RELIABILITY PROCESS

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The purpose of Paper is to presents the new developed PLiM Program which now is to be implemented into NPP Cernavoda in order to develop and revise the approaches required for anticipating, identifying, preventing and resolving performance problems with Structures, Systems and Components (SSC) on the basis of risk, support safe and reliable plant operation at optimum costs.

Cernavoda NPP is now developing a strong program able to ensure the proper station performance during its operation life which is expected to be extended beyond its design limit. This program is developed considering INPO recommended best practices (AP-913).

Requirements for inspection, maintenance, certification, testing and evaluating components and equipment are to be presented into Paper as follows.
- how monitoring results are tracked and trended;
- how the results will be utilized by the station System and Component Engineers to improve the availability, reliability and efficiency of components and equipment;
- how monitoring results are provided to the system engineers to assist in the monitoring and assessment of system performance.

Also, the Paper will presents the following topics:
- scoping and identification of critical components including the identification of important functions of SSCs and functional failure modes and effects analysis;
- continuing equipment reliability improvement;
- developing Preventive Maintenance templates for SSCs;
- developing program for Life Management;
- major equipment Life Cycle Plan;
- periodic in-service inspection for SSC programs:
  - pressure vessels requirements;
  - instrument calibration requirements;
  - pipe wall thinning (erosion – corrosion);
  - supports and snubbers;
- preventive maintenance for critical SSCs;
- system health and components condition monitoring;
- corrective actions and performance indicators.

A special emphasis will be addressed to PLiM process, how a major problem is identified and included into plant Long Range Plan and integrated with Plant Business Strategy. Also, in frame work of monitoring SSCs performance, the Paper will presents the System Health Monitoring and Reporting Process, specially the plant field specific collection data and analysis process mode in order to mitigate systems and components degradation.
EVOLUTION OF WWER-1000 RPV MATERIALS NANO-STRUCTURE UNDER IRRADIATION AND POST IRRADIATION ANNEALING

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Safe operation of the water-cooled VVER pressure vessel (VVER RPV) is determined by metallurgical, engineering, technological and many other factors, including validity of accepted reference dependences for prediction of physical and mechanical properties of RPV materials. RPV is one of the most important component in nuclear power plant. Failure or partial damage of RPV is not allowed with operation of nuclear power unit. Radiation safe life of RPV materials limits mainly operating safe life of the whole plant with VVER reactor.

At present moment the most actual task in the area of life time management VVER-1000 RPVs is the validation of possibility long term of operation. The validation of possibility VVER-1000 RPVs operation in terms of radiation embrittlement is being perfomed with use of normative dependence (transition temperature shift = function of irradiation dose (\(\Delta T_k = f(F)\))).

The normative dependence for VVER-100 RPV materials was developed in the seventies on the base of accelerated irradiation of VVER-1000 RPV materials with nickel content not higher then 1.5%. It does not take into account effect of nickel. At the same time the nickel content of the most part of the operating WWER-1000 RPV welds exceeds 1.5%, and in some of them it reaches 1.9%. The accelerated embrittlement of the welds with high nickel content is one of the basic problems, which limits radiation safe life of VVER-1000 reactor pressure vessels.

The testing results obtained withing the frame of surveilance and research programmes shown that normativ dependence is not coservativ for VVER-1000 RPV materials with nickel content more, then 1.5% and it is not possible to use it for the validation of RPV safe operation. Thus it is necessary to develop of the model which describes the behavior of the VVER-1000 RPV materials under irradiation adequate.

The main advantage of any adequate model is good forcast. Validation of the model essentially improves its prediction if it involves micro-processes that occur under irradiation and are the reason of mechanical properties behavior: hardening and embrittlement.

Progress in understanding of nano-structure evolution process and materials degradation mechanism under irradiation is due to improvement of "fine" methods of research, such as: focused ionic beam dispersion; transmission electron microscopy; positrons annihilation and, atom probe tomography (APT).
Atom probe tomography is one of the most demonstrative research methods of nano-structure evolution in steels under irradiation. The most part of studies were performed for western steels. There are only two research, made with VVER-440 RPV materials. Direct use of all these research results for VVER-1000 reactor pressure vessel materials is incorrect, as difference in chemical compositions of western and Russian steels are significant. Manganese content in Russian steels is lower, nickel content is much higher, etc. There are some difference in VVER-440 and VVER-100 RPV materials, in particular in nickel content. In this work for the first time research of nano-structure and mechanical properties behavior of base and weld metals has been carried out.

One of the most effective way to recove the mechanical properties of irradiated materials is annealing. This technology has been successfully applied for the old generation VVER-440 lifetime extension and it can be considered as one of the options for VVER-1000 RPV lifetime extention in perspective.

This work demonstrates the possibility of RPV materials monitoring degradation and mitigation using the Atom Probe Tomography. Atom Probe Tomography characterization of WWER-1000 base and weld metals was performed in cooperation between Oak Ridge National Laboratory (ORNL) and Russian Research Center, Kurchatov Institute (RRC KI). The WWER-1000 base and weld metals are low-alloy steels with low Cu levels (~0.06%), and nickel levels of 1.34% Ni and 1.77% Ni, respectively, produced by a standard technology procedure were selected for investigation.

Charpy impact specimens of both materials were irradiated at various fast fluences (>0.5 MeV). ORNL irradiated base and weld metal specimens at University of Michigan Ford Reactor. RRC «Kurchatov institute» made irradiation in commercial VVER-1000. the specimens were irradiated at temperature ~ 290°C. Characterizations of both materials in unirradiated state were performed by Charpy tests in RRC KI and in ORNL. Transition temperatures were assessed with usage of joint ORNL and RRC KI data.

The atom probe tomography results were analyzed with transition temperatures and yield strengths changing with increasing irradiation fluence. The evolution in size and number density of clusters under irradiation were assessed. Atom probe tomography investigations revealed a higher number density of Mn-Ni-Si clusters in the weld metal, which correlates with a higher transition temperature shift for the WWER-1000 weld metal as compared to the WWER-1000 base metal.

Atom probe tomography investigations of these steels were also performed for the post irradiation annealing condition. Parameters of the heat treatment were following: temperature of annealing was 450°C: the durations were 2 and 24 hours. The idea of this study was to demind the duration of annealing at 450°C which result in dissolution of the Mn-Ni-Si clusters in the base and weld metals.

A portion of this research was conducted at the SHaRE User Facility, which is sponsored by the Division of Scientific User Facilities, Office of Basic Energy Sciences, U. S. Department of Energy and by the Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission, under inter-agency agreement DOE 1886-N695-3W with the U. S. Department of Energy.
The portion of research, performed in RRC “Kurchatov Institute” was sponsored by ISTC within the project 3420 «Material science work package to extend safe life of VVER-1000». 
APPLICATION AND ISO PRINCIPLES FOR REGULATORY PRACTICES AND SAFETY CULTURE OF NUCLEAR INSTALLATIONS BEING BUILT TO RUSSIAN DESIGNS

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This Report covers conceptual approaches of technical regulation for processes of maintenance and management of service life of NPPs being built to Russian designs abroad. The Report presents basic principle of ISO 2000:9000, which we use in regulatory practice for management of maintenance, risk of operation and service life of NPPs being built to Russian designs. The Report offers the way to accomplish important missions of technical regulation based on ISO 2000:9000 international standards and recommendations on functional safety of IEC.

A package of new federal laws of the Russian Federation in the field safety culture and licensing of hazardous production facilities defines a main principle of requirements to a vender of nuclear equipment and to NPP on the whole. This principle is as follows: at any moment of operation the qualitative indications of risk should not exceed an acceptable social value of established safety indicators. On the other hand, to derive the highest benefit from equipment operation or from NPP on the whole, the second principle should be applied: “Operation as long as possible and full commercial utilization of resource and service properties of equipment, systems and NPP on the whole”. Implementation of the principle calls for transfer from an aging concept of planned regulation of diagnostic and maintenance service, repairs and/or replacement of equipment, pipelines or systems to a modern resource-saving concept of maintenance and/or termination of NPP equipment or pipelines operation with account of their real technical condition. In doing so, one shall take into account all available justifications of guarantees of NPP or its separate components further safe operation. Performing practical tasks of technical regulation, we should select new strategies and programs of maintenance and repairs considering engineering and financial analyses of safety, reliability, efficiency and profit of NPP operation. Hence, a new task arises: to establish a system for technical regulation quality management. The main objective of the task performance is to optimise expenses for diagnosis and maintenance service based on main operation strategies: before service life expiration, before failure, before pre-failure condition, and compliance with international and national NPP safety regulations.

Considering the world tendencies of providing products and services, FSUE VO "Safety" performs this task in the environment of continuously improved QMS for technical regulation of the whole service life cycle of nuclear installations being built to Russian designs abroad. ISO 2000:9000 international standards quality management basic principles lay the foundation for the QMS, namely, customer tailoring; system approach to management, process approach to normative and technical regulation, decision taking based on accumulated experience; knowledge base and facts, manager leadership, continuous improvement of QMS; involvement of employees; mutually beneficial relations with
venders and customers; etc. The gist and composition of the system is illustrated on Figure 1 below. One has to note, that ISO principles are applied jointly with a certain hierarchy of priorities by the whole team and participants of technical regulation of design, erection and operation of NPPs being built to Russian designs abroad.

**FIG. 1. Gist and composition of the system**
IDENTIFICATION OF OPTIMUM PARAMETERS OF ANNEALING FOR THE VVER-1000 RPV MATERIALS WITH HIGH LEVEL OF NI.

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This work shows an opportunity carrying out annealing welds № 3 and 4 cases of reactors VVER-1000 for decrease of critical transition temperature shift these elements. Annealing is one of effective ways for elimination or essential easing radiation embrittlement a reactor pressure vessels (RPV) materials. For carrying out of experiment the material of weld with the contents nickel 1,74 %, irradiated in free from regular assembly channels shield reactor NVNPP unit 5.

In this work present to be carried out development of mode thermal processing, (that is definition of parameters the temperature and duration of endurance at the given temperature), allowing to receive effective return of properties irradiated RPV materials VVER-1000. Annealing is carried out at temperatures from 400 to 490\textdegree{}C. Duration annealing varies in limits from 2 till 100 hour. Isothermal curves of restoration properties irradiated RPV materials VVER-1000 in process annealing are constructed various temperatures.
PREVENTION OF RESONANCES BETWEEN FLOW PARAMETERS OSCILLATIONS AND STRUCTURE VIBRATIONS IS THE PRINCIPAL RESERVE OF NUCLEAR POWER PLANT LIFE MANAGEMENT

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Resonant destruction of constructions takes place in cases when Eigen-Frequencies of Oscillations of the Coolant Pressure (EFOCP) begin to be equal to the eigen-frequencies of structural oscillations. The most dangerous dynamic interaction of the equipments and the fluid flow are supposed to be in the resonance region of mechanical oscillations of the elements and the parameters of the flow. The worked out R&D provides the understanding of the nature of the concealed dynamical processes in thermal hydraulic circuits of NPP, which are not foreseen in design and normative documents and not predicted by the thermal hydraulic computer codes. The basis research shows, that these processes appear in the form of self oscillations, caused by the equipments and coolant resonant interaction and other system effects. In many cases, due to the existence of these physical phenomena and processes, sudden failures of the equipments and accidents occur.

To prevent the appearance of the conditions for resonance interaction between the fluid flow and the equipments, it is necessary to provide the different frequencies for the self oscillations in the separated elements of the circulating system and also in the parts of the system formed by the comprising of these elements. To show this situation, we use experimental data that have been obtained from the measurements of noise signals in the frame of the cold -ops, hot -ops and nominal operation program done at WWER – 1000 NPP. The experimental data that have been obtained from the measurements of noise signals in primary interaction between Eigen-Frequencies of Oscillations of the Coolant Pressure (EFOCP) and structure vibrations are discussed.

Considering the research results of vibro-acoustic measurements which have been carried out in the commissioning stage of WWER-1000 NPP, it is possible to understand the reason of underestimation of the role of high-cycle loadings in occurrence of cracks in metal of the primary equipments of the first circuit. Really, in all commissioning stages in the steady-state modes, the level of vibrations did not reach maximum permissible values. The mode in which resonant interactions with the coolant takes place can be designed and predicted on the basis of application of the methods had been developed. It concerns to reactors of PWR types, like Davis Besse, and to any type of reactors with water or steam-and-water coolants.

It is necessary to emphasize, that combinations of parameters for the reactor operating mode in the worst case for reactor lead units, for each reactor will be unique. Generally, they are determined by considering the geometrical sizes of the reactor core, reactor, the equipment, pipelines, and thermo-physical parameters of the coolant and the scheme of its movement.
Thus, as shown above, for revealing this mode, the type and frequency of rotation of MCP is also important. The researches carried out, allow determining procedure for control of residual resource of reactor lid units and the timely warning to the personnel about achievement of a maximum permissible level of dynamic weariness of metal of reactor lid unit. This procedure is based on:

a) application of non-destructive testing for obtaining information on dynamic processes in the coolant and lid unit,
b) substantiation of the types of detectors, their quantities and their placements,
c) substantiation of periodicity of measurements of the specified signals,
d) determination of ranges of reactor power level change and thermal-hydraulic parameters of the coolant in transient modes in which it is necessary to carry out continuous monitoring of signals,
e) the statistical analysis of signal information in system of not-destructive testing,
f) determination of the residual resource of metal of reactor lid unit in the zone of prospective destruction,
g) documentation of the results of measurements, the analysis and an estimation of a residual resource.

To prevent this resonant interaction and realize Nuclear power plant life management, it is proposed to develop Vibro-Acoustics Specifications (VAS). Based on NPP VAS, it would be possible to reveal the dynamical loadings on metal that are dangerous for the initiation of cracking process in the early stage of negative condition appearance. An application-oriented circuit of NPP with reactor of WWER –1000 are presented. The causes of resonant approach to the problem of identification of abnormal phenomena of thermal-hydraulic parameters is proposed. Thus the objective is to simultaneously monitoring the deviation between EFOCP that corresponds to normal or initial operating conditions and those to abnormal phenomena. These deviations can be characterized by the set of error criteria. Logarithmic Decrement $\delta$ has two uses. First, the successive peak values of free oscillations can be found if one peak value plus damping ratio are known. Second, measurement of the ratio (or average ratio) of successive peak allows for estimation of the (equivalent) damping ratio $\zeta$ for a real system. The bigger $\zeta$ provide bigger $\delta$ and correspondingly smaller values of $\varphi$-factor and amplitude $x(t)_w$. All experimental units intended for NPP severe accident investigation must satisfy to the NPP $\varphi$-factor criterion of similarity.

The worked out results are intend to share technical updates on ageing management issues, including identification of maintenance, inspection issues and mitigation to material degradation.
EXPERIENCE OF NPP I&C MANAGEMENT OF AGEING

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The report devoted to Ukrainian experience, shared to I&C of 4 NPP’s with 11 units WWER-1000 and 2 units WWER-440. The peculiarities of Ukraine are the follow:

- instrumentation and control systems (I&C) which began to operate with units start were obsolesce after definite time, didn’t satisfy modern safety requirements in full volume, weren’t state of art (didn’t use digital technique);
- resources for I&C replacement were limited (especially 10-15 years before);
- deficiency of spare parts for many types of instruments;
- age of big part of instruments– from 17 to 27 years;

The main actions of I&C management of ageing are the follow:

1. Elaboration of two regulations of Ukrainian Nuclear Authority:
   - safety requirements to new and modernized safety important I&C, which were harmonized with requirements in IAEA and IEC standards; scientific base of this document (see [1] ) was consideration of tendencies of I&C evolution and analysis of international and national standards.
   - requirements to order and content of works of life extension of I&C systems important to NPP safety, 1-st revision - 1996, 2-nd revision - 2003 (additionally - elaboration of the standard of Operator Company devoted to methodic of statistical analysis of information for life extension decision); scientific base of these documents were comparison of an efficiency and possibility to use methods of checking of statistical hypothesis; we chose a nonparametric method of inverses.

2. Determination of the types of the systems which have supreme importance from safety and didn’t correspond to new safety regulations. Operator Company elaborated program of step by step modernization of these systems – by replacement on modernized types. This process actively began after 2000. The most part of this program was fulfilled till 2006. SSTC NRS prepared expert reviews of all modernized by this program safety important I&C. Now modernization process is continuing for the next set of the systems.

3. Elaboration of the programs for periodic analysis of technical state of instruments which didn’t included in the list for immediate replacements. The programs contained external inspections; components testing; instrumentation total testing; analysis of spare parts presence; environment conditions analysis. SSTC NRS prepared expert reviews of these programs. The 1-st step was individual programs for every NPP’s ,the 2-nd step – common programs for all Ukrainian NPP’s.

4. Ukrainian safety regulations include the following requirement to the safety important components: each component at its service life end has to be replaced or a possibility of its life extension has to be proved. This requirement relates to different types of equipment, including I&C.
The task of the life extension was decided by the following three ways:
- replacement of equipment with ended service life by modernized new one;
- replacement of equipment with ended service life by similar one;
- life extension, that is possible when the equipment can continue to fulfill its functions with required characteristics after the service life end.

The following directions are included in the evaluation of the possibility of the equipment life extension:
- analysis of the equipment technical state;
- analysis of the equipment operating reliability measure (especially an analysis of a failure rate trend);
- analysis of the I&C failure impact upon the safety violations.

The 1-st step was individual decision of pilot instruments by every NPP, the 2-nd step – common decision for all Ukrainian NPP’s.
We can note that all NPP can a number of the single-type instruments. Consequently, it is possible to deal with the decision on the life extension for group of the single-type instrumentation being under the same conditions.

5. Adequacy of this technical actions in definite degree could be confirmed by analysis of violations of NPP safety because I&C. Statistical information, which have been processed from 10 years, shown that there aren’t increasing of violations rate because I&C as to WWER-1000, as WWER-440 [2].

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REPLACEMENT OF HEAVY WATER SUPPLY CONTROLLERS IN FUEL HANDLING SYSTEM

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Both automatic analog controllers and both analog manual controllers (63526-PC#11 A and 63526-PC11#1 C) (63526-PC11#2 A and 63526-PC11#2 C) were replaced by three Digital Controllers as follow: 63526-PC11A, 63526-PC11C and 63526-PC11 because of the aging of old analog controller.

In the new System, there is a separate controllers for each one in both side, (Side A and Side C) and other to control the common bleed valve.

The reasons more important for this replace.

The analog pressure control system

- Is a design from the early 1980s
- Analog control principle used
- Custom-built instrument control system
- System control by electronic hardware. No programming involved.
- There is not more commercial support for hardware. (electronic printed circuit board)

The digital pressure control system

- Programmable
- Commercial full support
- Easy calibration and maintenance
- Easy operation

Because of the above keys, Embalse NPP decided to change the control system together with AECL, without shutdown the Plant.

The last one was the challenge, to install and commissioning the new controllers without affect the electrical production.
INDIAN REGULATORY REQUIREMENTS WITH RESPECT TO PLANT LIFE MANAGEMENT FOR RENEWAL OF OPERATION AUTHORISATION

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Atomic Energy Regulatory Board (AERB) is the regulatory authority in India, and Nuclear Power Plants (NPPs) are under its regulatory purview. License is issued to an NPP for design life, which is generally 30 to 40 years. During the process of licensing, aspects important to safety are assessed at stages, such as siting, design, construction, commissioning and operation. The regulatory body issues the license for operation of NPP after review of commissioning test results. Within the operating license, AERB issues initial authorization for operation for a specified period. Renewal of authorization for operation for further period is issued after assessment of safety performance of the NPP. Utility has to apply for license renewal for further period of operation (beyond design life) well before (atleast 5 years) the end of design life.

AERB safety guide on ‘Renewal of Authorisation for Operation of NPP’ provide methodology and guidelines on the periodic renewal of authorization of operational NPP by conducting PSRs and submitting the same to the Regulatory Body. AERB safety guide on ‘Life Management of NPPs’ details the essential factors that are required for a comprehensive assessment of the ability of the IIS for performing their intended functions reliably as per design specifications and addresses the planning and implementing an effective life management programme for IIS.

Consideration by AERB for License Renewal are (i) Safety margin in the design (ii) safety level (iii) residual life of SSC (iv) Adequacy and effectiveness of the technical management of the plant, safety culture, (v) Site characteristic (vi) Reassessment of Environmental burden (vii) Probabilistic Safety Assessment (PSA) complemented with deterministic judgment for both individual and collective issues for assuring compliance with safety requirements.

Methods of Assessment for License Renewal are (i) Periodic Review for Life Management and (ii) In-Depth Review

For life extension beyond initial license period (30 to 40 years), it is imperative that a comprehensive in-depth review of design and operational aspects of the plant should be performed with reference to current safety standards to ascertain non-conformance to safety requirements. Safety issues are identified based on acceptance criteria from the Regulatory Body. Deterministic Safety Analysis Methods and Probabilistic Safety Analysis Methods together can help in the exercise of judgment. The in-depth review of the results of the assessment will indicate the weakness of systems/subsystems and urgency of corrective measures.
Tarapur Atomic Power Station Unit 1 & 2 (TAPS 1&2) were commissioned in the year 1969. The units remained shutdown between October, 2005 and February 2006 for implementation of safety upgradation and ageing management related actions. These actions had been identified earlier during 2000-2004 through detailed ageing studies and comprehensive safety review of these units. AERB granted permission in February 2006 for restart and renewed the authorization for these units upto March 2011. This paper deals with the regulatory issues with respect to plant life management for renewal of plant operation authorization.
FAILURE PROBABILITY ASSESSMENT OF A PWR PRIMARY SYSTEM PIPING SUBCOMPONENTS UNDER DIFFERENT LOADING CONDITIONS

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The integrity of primary piping system of pressurized water reactor must be maintained during the lifetime operation period. In doing so, credible structural integrity analyses methodologies including fracture mechanics approach, etc., are required. Conventional deterministic approaches are being widely used for this purpose though these methods have many inherent uncertainties to encumber a coherent assessment. In this respect, probabilistic methodologies are considered as an alternative but appropriate technique for the structural integrity analysis of reactor piping system. In the nuclear reactor piping system, the elbows were reported to be among the most highly stressed piping components. This requires an extensive flaw evaluation of these components. The objectives of this paper are to evaluate the fatigue failure probabilities of various subcomponents in a nuclear reactor primary system and to evaluate the relative risk ranking among them. For this purpose, a probabilistic fracture mechanics code using Monte Carlo simulation techniques was developed by using both circumferential and longitudinal crack module and then utilized it to calculate fatigue failure probabilities due to small leak, big leak and LOCA situations subjected to all possible loadings. The crown of the shut down cooling elbow shows the highest leak failure probability about 1.00E-03 per base metal section and charging nozzle safe end shows the smallest leak failure probability about 1.00E-08 per circumferential weld for the sixty years of plant life.

The purposes of this paper are to find out the life time fatigue failure probabilities of primary piping subcomponents of pressurized water reactors (PWR) made of either 304/316 SS steel or SA508 Cl.1a low alloy steel (LAS) and to propose the relative ranking of severity of failure probability of these components. In doing so, a probabilistic fracture mechanics (PFM) code has been developed by using the Monte Carlo simulation technique. This code can handle both circumferential and/or longitudinal weld/base metal postulated crack problems. Five components e.g. RPV Inlet Nozzle, Surge Line Elbow, Charging Nozzle Safe End, Safety Injection Nozzle Safe End, Shutdown Cooling Line Elbow were analyzed for circumferential weld cases. Axial base metal crown sections were also studied for elbows considering stress indices. The probability of detection curve taken from recently published NUREG document was used for pre-service inspection [1]. All the calculations were done for both stainless and low alloy steels to show relative failure probabilities and finally attentions were given only to low alloy steels for ranking the small leak failure probabilities of primary piping subcomponents for sixty years of plant life.

The piping integrity evaluation program based on PFM was developed using Microsoft Visual Basic 6.0. The program consists of input material property part, reactor operating condition part, constants part, the controlling part, i.e. cell size, simulation number and the output result part. This code can be used for austenitic stainless steels and carbon or low alloy steels materials which are commonly used for today’s light water reactor.
FIG. 1. Cumulative probabilities of small leak of SA508 Cl.1a LAS for all components of interest.

Fig. 1 shows the relative ranking of all the selected subcomponents of primary system of PWR. The crown sections of elbow show the highest small leak failure probability. Among two base metal section of shut down cooling line elbow and surge line elbow, shut down cooling line elbow poses a little bit higher risk of small leak failure probability on the order about 1.00E-03 per base metal section. On the other hand, the circumferential weld section of shut down cooling line elbow has the ten times more leak failure probability than that of surge line elbow. So, taking both circumferential weld and longitudinal crown base metal section in account, shut down cooling line elbow has the highest risk of small leak failure probability. Charging nozzle safe end and safety injection nozzle safe end demand the least circumferential weld small leak failure probability although the safety injection nozzle safe end shows a very little higher failure probability. RPV inlet nozzle shows a moderate small leak failure probability which was also verified in earlier work [2].

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METHODS OF EVALUATION OF OPERATIONAL EXPERIENCE FOR REGULATING DECISIONS SUPPORT

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In the given paper the methods of operational experience evaluation which are used in the regulation of nuclear and radiation safety of Russian NPPs are stated

To follow the current tendencies in NPPs operational safety SEC NRS – organization of technical support to the Regulatory Body – realizes analyses of annual reports submitted by operating organization. These reports contain information on different aspects of operational activity, including:

- State of physical barriers (fuel cladding, primary circuit, confinement);
- State of physical barriers protection systems (i.e. safety system);
- Information on stability of NPP operation;
- Number of operational (including accidental) loading cycles which limit the lifetime of main NPP equipment;
- Water chemistry indices;
- Changes in the technical support and maintenance practice;
- Results of safety important non-destructive equipment check;
- Detailed data on NPP operation violations;
- State of radiation safety etc.

In the article we characterize evaluation methods for the information on the stated indices submitted by the operational organization as well as trend detection methods. Besides some examples on connections between analyses results and regulating decisions are shown (decisions can imply assignment the inspections on specific safety aspect(s), decisions on operational licenses conditions changing etc.).

Comprehensive evaluation of operational experience is realized by SEC NRS also within the framework of scientific and engineering review carried out upon regulatory body request in the cases when operator applies for getting renovated operational license.

Now Russian Regulatory Body studies the possibility of risk-informed approaches implementation into the process of NPPs operational states and operational experience trends for all Russian NPPs (though PSAs diff in scope, quality and readiness for use in applications), as well as many specialists from industry, design institutions and regulatory body become familiar with probabilistic methods.

The first step in the direction was development of Safety Guideline (1), which expounds risk-oriented approach to the decision making the situations when discrepancy with the present normative demands was discovered.

Besides we developed special SEC NRS Methodological Guide (2). It expounds general principles of PSA technique use in course of regulatory decision making for NPPs. The Guide describes use of risk-oriented technologies in the regulatory activity for such fields as
operational experience (operational events) assessment, operational limits and conditions evaluation and others.

Discussion on risk-oriented methods incorporation into regulatory practice forms the second half of the paper.

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COMPARATIVE ANALYSIS OF IRRADIATION CONDITIONS OF SURVEILLANCE SPECIMENS AND RPV FOR LIFETIME EXTENSION OF ROVNO-1 AND ROVNO-2 NPP'S

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The key component of WWER is the Reactor Pressure Vessel (RPV). The evaluation and prognosis of RPV material embrittlement and the allowable period of its safe operation are performed on the basis of impact test results on irradiated surveillance specimens.

Surveillance specimens are irradiated in special channels located on the outer surface of the shaffle of VVER-440 reactor. Periodically chains with surveillance specimens are unloaded and then tested. Results of irradiated surveillance specimens tests are compared with test results of unirradiated spesimens.

The neutron fields parameters for surveillance specimens (neutron flux and spectrum) are different from the RPV irradiation parameters.

In the present work analysis and comparison of irradiation conditions (neutron fluxes and neutron spectrum) of containers with surveillance specimens and RPV inner and outer wall has been done. It was shown that netron fluxes on SS are much higher (10-15 times) then neutron fluxes on RPV wall. Also doze rates in DPA terms from neutron spectrum on SS and RPV were analyzed.
APPLICATION OF THE “LEAK BEFORE BREAK” CONCEPTION ON NPP UNITIS OF THE FIRST GENERATION WITH WWER – 440 REACTORS. IMPROVEMENTS OF LEAK DETECTION SYSTEMS CONSIDERING NEW ELABORATED APPROACHES

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In the report there are presented main results of the work of substantiation of “Leak Before Break” (LBB) concept applicability and its implementation on WWER-440 NPPs: Kola NPP, units 1, 2; Novovoronezh NPP, Units 3, 4; Armenian NPP, unit 2.

The essence of the LBB concept consists in a fact that the design of NPP component and materials used for this component manufacturing assure that a total destruction of the component having a through wall crack is impossible without a preliminary existence of the stable leak of the working medium. Such a leak can be detected in advance before the moment when crack length reaches its critical size and the crack becomes unstable. LBB concept should be applied to the pipelines of the main circulation circuit in order to assure a safe and reliable operation of NPP.

The main stages and results of the work of substantiation of the LBB concept applicability for Main Circulation Lines and Surge Lines of the unit 2 of Armenian NPP with WWER-440 are presented in the report. The work was conducted together with specialists of Ansaldo Energia (Nuclear Division) and Empresarios Agrupados Internacional. It is shown, that Surge Lines are the most critical components concerned with necessity of LBB requirements fulfilment, also possible technical ways intended to resolve this problem have been proposed. There was carried out a comprehensive analysis of technical performances of regular Leak Detection Systems which exist on WWER-440 units and are suitable for LBB concept implementation. There are performed recommendations on upgrading of that LDS in order to meet all LBB requirements. Also there are proposed fundamentally new approaches as regards to a design and mounting on NPP of new modern and extra-sensitive leak monitoring systems.
PROBLEMS OF DEVELOPMENT OF A PREDICTION TECHNIQUE OF A RESIDUAL LIFETIME OF HEAT EXCHANGING TUBES OF WWER NPPS

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A forecast of a residual lifetime of steam generators’ (SG) heat exchanging tubes (HET) is a complex task. Estimation of the residual lifetime includes an analysis of the following factors: influence of water chemistry parameters of the working medium in which SG tubes operate, determination of mechanisms of corrosion damaging in deposits surroundings, possible damaging due to chemical rinse, information regarding the revealed defects and plugged tubes.

Results of the analysis allow to define complex of input factors, which determine a metal condition of SG tubes through the effect on following parameters of SG tubes: metal condition, bearing capacity, defectiveness, working ability, reliability and lifetime.

Researches of HET samples being cut from a dismantled SG allowed to determine (classify) main types of defects which could be observed on HET of horizontal SGs.

For each defects’ type there were determined special features of eddy-current signals depending on defect’s type. The atlas of defects which includes the results of eddy-current inspection and metallographic researches of the main defects’ types was developed.

For the calculation analysis of the residual lifetime there was selected a neuron network theory, as the most suitable tool for calculation using so indefinite and fast changed initial data. The neuron network approach is useful not only for linear dependencies but for complicate nonlinear dependencies as well, because the neuron networks are nonlinear by there nature. Also the neuron networks have ability to learning and generalization of collected data.

Application of the neuron networks technology for making forecast of the residual lifetime allows to reveal a mutual influence of the above listed input factors on the metal condition of SG tubes. Also it can be estimated an effect of each separate factor on the damaging and its contribution to the total damaging, as well as mutual influence of different factors one to another.

As a result we obtain a technique, which allows to analyse not only a current condition of SG tubes metal, but also to make forecasts regarding the defects’ number and size changing, determination of rates of defects’ growth and appearance.

Furthermore, identified dependences of influence of the each damaging factor on HET metal condition allow to define factors with maximum negative influence. On the following stage of the work definite actions intended to minimize influence of negative factors should be fulfilled. That is the essence of SG tubes’ lifetime management. Presented results use the data from the Research Project on Strategy for Assessment of Steam Generation Tube Integrity Reproducing by the IAEA Vienna.
EFFECT OF THE RPV CLADDING PROPERTIES ON THE WWER – 440 REACTORS LIFETIME

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The RPV cladding generally not involved into the stress and strain analysis, even it is considered in structural integrity assessment only as a layer in the heat transfer models. The IAEA PTS guide of WWER-440 RPV-s allows to consider only underclad hypothetic cracks if the RPV cladding is free of defects and ductile.

The cladding is a welded structure. According to the chemical composition it is austenitic, but due to the welding it has 5-10% delta ferrite. The delta ferrite changes the material behavior, it shown transition properties of ferrite and austenitic material.

Irradiation increases the strength of it, and decreases the toughness. To evaluate the ductility and mechanical properties of the WWER reactor’s cladding test blocks have been cut from the Zarnowiec and Greifswald 8 units. Both reactors were manufactured at Skoda Works (Czech Republic) but they never operated. Cladding specimens have been irradiated, annealed and re-irradiated in the Budapest Research Reactor and tested.

Several mechanical tests (tensile and fracture properties) and metallographic samples have been studied to evaluate the properties of the irradiated cladding.

Database of irradiated cladding properties have been collected to allow elastic plastic analysis of the reactor pressure vessels during transient thermal stresses.

The data have been used for PTS evaluation of WWER-440 V-231 type blocks. Calculation have been performed by traditional elastic stress analyses using surface and sub-cladding hypothetical cracks, and by elastic-plastic finite element code using sub-clad hypothetical defects according to the IAEA PTS guide and the VERLIFE guide. The calculated safe lifetime is doubled in the case of the critical transients.
DEVELOPMENT OF AGING MONITOR FOR OPERATING NUCLEAR POWER PLANTS

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As NPPs' operating times increase, the integrity of nuclear components is continually degraded due to aging effects of systems, structures and components. In addition, for the case of continued operation beyond design life, additional aging effects occurred during the extended operating period lead to more degradation of the integrity of nuclear components. Therefore, it is very important to monitor and evaluate the aging to secure the safety of NPPs.

In this paper, web-based Aging Monitor (AM) to give real-time Coloring and Alarming on aging degradation were introduced to secure the safety of nuclear components. The AM was constructed to monitor, manage and evaluate the aging systematically and effectively. Basic concept for AM was established. The AM is composed of 6 modules: Aging Alarm/Coloring Monitor, aging database, aging document, real-time integrity monitor, surveillance and inspection management system, and continued operation and Periodic Safety Review (PSR) safety evaluation. Aging Alarm/Coloring Monitor is the key module which shows the state of aging with 4 different levels such as integrity, monitor, watch and alarm for aging unit and aging mechanism. The other modules are supporting and application ones. Aging Alarm/Coloring Monitor for primary system components and supporting data bases was also constructed.

The AM can be used for regulators, operators, and public information. The regulators can use the results for regulatory management: operating NPP aging-related licensing, utilization of database, and solving crack and wall thinning problems etc. in the site. The operators can utilize AM results for evaluation of aging factor for PSR, evaluation of TLAA and establishment of AMP for life extension.
DACAAM – NET – AN INTRANET APPLICATION FOR MANAGEMENT OF AGEING RELATED DATA

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The new network version of the Data Acquisition and Analyzing System for Ageing Management (DACAAM-NET)) system is based on an earlier application for stand-alone computers and was developed as a tool to integrate the ageing management related information into the general IT support system of the nuclear power plant. It provides systematic way of collecting, assessing and presenting information on ageing of main components. The DACAAM-NET is designed as an open system with an interface exposing the technical data on ageing to other technical databases and applications and with capabilities to incorporate relevant ageing related data of other applications. The link is implemented through alphanumeric identifiers of equipment commonly used across the whole plant. The network version includes more components and it is closely linked to the web application on defining the scope of license renewal. On the other hand, selected ISI results of the reactor vessels and the steam generators are linked to DACAAM-NET and presented through a special graphical interface with 3D modeling.
APPLICATION OF GENERAL METHODOLOGY PLiM IN THE LIFE ASSESSMENT OF EMERGENCY CORE COOLING SYSTEM HEAT EXCHANGER AT EMBALSE ENGS

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The Argentinean Atomic Agency – Comisión Nacional de Energía Atómica (CNEA) – provide assistance to Embalse Nuclear Generating Station (ENGS), in its PLEX Project. ENGS is a CANDU 6 type with 648 MWe output. The end of design life is foreseen for 2011.

ENGS and CNEA personnel were trained in PLiM methodology in Canada and also at the plant. In order to validate this methodology, some pilot studies have been performed. With this experience, several Aging Assessment (AA) studies are being performed of Systems, Structures and Components (SSCs). This report pretends to show through a specific AA example how the PLiM methodology could be applied.

A Life Assessment (LA) Report of the Emergency Core Cooling System Heat Exchanger (ECC HX) will be analyzed. This HX remains in stand-by, due to the system design, ready to work if an extremely unlikely event takes place.

The LA process consists of the following basic steps:

1) Information Gathering
   • The common gathered information are Operation, Maintenance and Design Manuals, Technical Specifications, etc.

2) Defining SSC Scope & SCC/Subcomponent List Preparation
   • The scope of the LA report was determined by using the physical limits that were established to primary and secondary side connections and external supports. The interfaces of the ECC HX are with the ECC and the Service Water System.

3) Screening of Subcomponents (as appropriate; in the absence of subcomponent screening, all subcomponents are evaluated further)
   • No subcomponent screening was needed because a significant reduction in the work scope was not expected for this type of subcomponents. So, almost every subcomponent was evaluated, except those that were screened out based on experience.

4) Information Review (including historical data, walkdowns, interviews, etc. specific to individual SCCs)
   • The operational and historical data is analyzed by taking into account the:
     o Deficiency reports: seven reports were found. Three of them were related to inspections, one mention the change in the operative conditions in the secondary side circulating a small flow in order to
o maintain it full with service water, and the others with no big significance.

o Engineering and event reports: a spurious trip of the ECC System occurred. Therefore, all equipments involved actuated, including the heat exchanger. That was the only time in which this heat exchanger was in service. During this event, the ECC worked at high pressure, while the mid-pressure ECC was in stand-by condition.

o Interviews with specialist: These interviews allowed to know the state and operative history of the heat exchanger with data not reported in the official documents.

o Walkdown trough the ECC HX: The walkdown performed for this LA was done with the heat exchanger not open. So, only was seen its external state.

o Review of the Preventive/Predictive Maintenance Programs and In-Service Inspections (ISI) related to this component: The recommended frequency of inspection was followed and this heat exchanger was inspected.

o Obsolescence: the manufacturer is still in the market and every subcomponent needed can be easily acquired.

o Environmental Qualification: up to now there is not any environmental issue identified for this component.

5) Evaluation of aging related degradation mechanisms
All the gathered information in the previous stages allow to determine the possible degradation mechanism of the ECC HX

o Pitting and Crevice Corrosion had been found in the channel plates in the service water side proximally to the outlet zone and gasket. It can not been mitigated because this water comes from the lake, in order to avoid the progress of this degradation mechanism, every channel plate found with marks of pitting and/or crevice is change in every inspection.

o MIC and Fouling in the service water side is an active degradation mechanism in ENGS. These degradation mechanism enhances localized corrosion processes (e.g.: pitting and crevice corrosion) and, if fouling deposits are big and hard enough, it could reduce thermal performance.

6) Establishing conclusions, health/life prognosis, and recommendations.

o A preliminary assessment of the data gather for this component allow us the following conclusion: the condition of the ECC HX and its health/life prognosis is GOOD (it means that it will finish its design life and is likely to operate satisfactorily over the extended target life with no significant aging degradation evident). Recommendations had been done and are mainly related to improvements in the maintainance and inspections practices. It is also concluded there is not a degradation mechanism that could be a life-limiting. Also if, these degradation mechanism are easily handled. Therefore, taking into account all above mentioned, is easily decide that this heat exchange will reach safely the end of life design and it will be able to operate in a future life extension without major changes.
THE LESSONS AND FINDINGS FROM THE FUEL CHANNEL LIFETIME MANAGEMENT STUDY

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Wolsong-1 NPP in Korea has been operating for 24 years since 1983. Although the life of the fuel channels was designed as 210,000 effective full power hours (about 30 years), many aging phenomena were observed in the primary heat transport system. KEPRI has performed the plant lifetime management study for Wolsong-1 by two stages.

The first stage of the plant lifetime management study began in 2000 and its target was to find the ongoing aging phenomena and evaluate the aging status and remaining lifetime of the major systems and components. Major components and structures such as fuel channels, feeder pipes, reactor assembly, steam generators, heat transport system piping and pumps were selected for assessment by the selection criteria. For assessing the aging status, all the records of design, construction, operation, inspection and maintenance history were collected and reviewed. Lots of technical standards such as ASME, CSA, IAEA and technical documents of COG, AECL, etc. were referenced as the technical basis to assess the aging status of the components. The fuel channel elongation due to the irradiation creep and growth was found as the most life-limiting aging phenomena. An economic assessment was done to see the cost and benefits of the continued operation. The continued operation of Wolsong-1 with the refurbishment was compared with the construction of new 900 MWe type plant. The sensitivity analysis was done to find the optimum investment point. According to the assessment, the continued operation had more economic advantage than the construction of new plant and the cost-benefit ratio was about two, in case 20yrs extended operation. The feasibility of long-term operation was also reviewed as the viewpoint of regulation. The international regulation status about the continued operation was reviewed and compared with domestic regulation. As the results of the first stage study, Wolsong-1 established the plan to replace the fuel channels and feeder pipes.

Two years after completing the first stage of the PLIM study, KEPRI also started the second stage of the plant lifetime management study for the purpose of establishing the aging management program for the long-time operation of the plant and finding the components or systems necessary to replace together with the refurbishment of the fuel channels and the feeder pipes. The overall aging assessment was performed for all the components, structures and systems in the plant. To conclude the exact refurbishment time, the detailed lifetime assessment of fuel channel was reperformed by incorporating the latest inspection data. Based upon the results of the study, Wolsong-1 decided to perform the channel shifting work before the large scale refurbishment by the recommendation of KEPRI. By performing the channel shifting, W-1 can afford the time to manufacture new fuel channels.

This paper will show the lessons and findings through the plant lifetime management study, particularly focusing on fuel channels and strategy for the long-time operation of the Wolsong-1 which includes the large scale refurbishment.
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AGEING ASSESSMENT OF RBMK – 1500 FUEL CHANNEL IN CASE OF DELAYED HYDRIDE CRACKING

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Ignalina NPP contains RBMK-1500 type reactors. RBMK reactors are graphite-moderated with a water-cooled reactor core. The fuel cell assembly is located in the centre of the moderator column and consists of a fuel channel into which the fuel element assembly is inserted and through which the coolant flows. Zirconium alloys are used as a constructional material for manufacturing both claddings of fuel assemblies and fuel channels (FC) [1]. Zirconium alloys can pick up hydrogen during operation as a consequence of corrosion reaction with water.

When the terminal solid solubility of hydrogen in zirconium alloy [2] is exceeded in a component such as pressure tube that is highly stressed for long periods of time, delayed hydride cracking (DHC) failures may occur. DHC is a phenomenon where a crack can propagate in stepwise fashion as a result of hydrogen redistribution ahead of the crack tip under stress level below the yield stress. The high mobility of hydrogen enables hydride to redistribute. If stress levels are sufficiently high the hydride platelets precipitate in the primary cracking direction. The formation of hydrides under certain conditions can reduce resistance to brittle fracture and cause the initiation and development of hydride cracks (delayed hydride cracking). Therefore the evaluation of the influence of hydrides to fracture of FC is important.

The paper presents results of the analysis of aging process of the Zr-2.5%Nb alloy fuel channel (FC) tubes used in the second unit of the Ignalina NPP. The investigation of the hydrides influence to life time of the fuel channels was carried out. In these experimental investigations the sections of the FC were hydrided to produce required hydrogen concentration using an electrolytic method and diffusion annealing treatment [3].

Predetermined amounts of hydrogen ranging from 27 to 76 ppm were added to the unirradiated sections of the FC tubes by electrolytically depositing a layer of hydride on the surface of the FC material followed by dissolving hydride layer by diffusion annealing at elevated temperature. The compact toughness specimens were machined from hydrided FC tube according to ASTM E-399 requirements. The hydrogen content was determined using hot vacuum extraction method. Hydride crack formation conditions were investigated and delayed hydride cracking (DHC) velocity was measured at different temperatures [4]. It has been determined that the DHC velocity is approximately $2 \times 10^{-9}$ m/s in RBMK TMO-2 tubes at temperature 250 °C. Obtained DHC velocity results for the TMO-2 tube were compared with the DHC results for the tubes of different manufacturing technology. DHC velocity for the TMO-2 tube are about 15 times lower than in RBMK TMO-1 and about 42 time lower than in CANDU pressure tubes material at same temperature.

The evaluation of the influence of the hydrogen to mechanical properties and fracture parameters was carried out. J-integral and mechanical properties of the FC tubes were
determined under different temperatures and hydrogen concentrations. Deterministic analysis of the fuel channel employing “leak before break” concept was carried out. The critical length of through-channel cracks, the function of crack opening, the leak rate through this cracks was calculated. The stability analysis of crack growth was carried out too. The influence of the hydrogen concentration was evaluated using tested material properties and fracture parameter of the zirconium alloys with different hydrogen concentrations. The influence of the irradiation to yield strength and DHC velocity was evaluated in this analysis also.

The prognosis of DHC crack groving was calculated at 70 ppm hydrogen concentration. The critical cracks length in zirconium alloy without hydrogen and with 70 ppm hydrogen concentration, the decreasing of temperature and the prognosis results of the DHC crack growing in time shutdown of reactor are presented in this Figure 1.

![FIG. 1 LBB assessment results for emergency reactor shutdown procedure](image)

Analysis confirmed, that the length of postulated crack, at which necessary leak rate is reached, is less than half-length of critical through-wall crack. The possible intersection of the growing crack curve with a critical crack size curve could not be reached.

REFERENCES


PIPING EVALUATION UNDER THE DEGRADATION MECHANISM CAUSED BY CONTROL VALVE FREQUENT ACTION INDUCING VIBRATION

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Vibratory fatigue is listed as one of the piping degradation mechanisms in the IAEA technique documents, because it has caused failures of small-diameter piping worldwide, especially at socket welds in NPP. It is indicated that failures have resulted from high-cycle mechanical fatigue accompanied by low-amplitude cyclic stress. The excitation mechanisms for vibratory fatigue are mainly explained as pump-induced pressure pulsation, cavitation and flashing. Here an unique excitation mechanism relating to control valve frequent action is discussed.

When a branch pipe line with two normally closed valves suffers the first valve leak, the inner pressure of the pipe between the first valve and the second control valve becomes higher. And the control valve will partially open to release the pressure when it reaches a designated value of the valve. Then the control valve will close so as the pressure goes down enough. The control valve will repeat this kind of opening and closing regularly since the first valve continues leaking. Owing to the design characteristic of the control valve, its action of opening and closing sometimes initiates repeatedly pressure pulsation acting on the branch pipe. It will cause distinct vibration of the pipe when the branch line is inadequately supported and the pressure pulsations coincide with a structural frequency of the piping. This excitation mechanism sometimes will cause leakage of the connection between the valve and the pipe, or the failure of the elbow in the downstream line.

On the other hand, the discharging line of a container is connected with a control valve. The function of the valve is to control the water level of the container. The valve is normally closed until the water level of the container reaches a designated position. Then the fluid passes through the discharging line from the container, and the control valve is opened until the water level reaches a minimum position. If the volume of the container is badly proportioned to the diameter of the container’s inlet and outlet pipe line, the control valve will repeat opening and closing frequently. This will also becomes an excitation mechanism of vibratory fatigue when the piping line is not sufficiently supported. It is not a solution to change the leaking valve, because the replacing valve will leak again after a period of operation time. It is not easy to change the container and its connecting piping lines, furthermore it may be unnecessary.

The first thing should be done to solve the above problem is to make sure how severe this excitation is and how much its effect on piping. We find way to determine the time-dependent excitation load, and analyze the reaction of the piping system under the load to make sure whether or not the piping system is sensitive to vibration fatigue caused by the excitation load.
If the service life of the piping system is adversely affected by the above excitation mechanism, the second thing is to decide a way of solution. To propose modifications of piping support and position is an effective and economical method to solve the problem. From the above analysis result, we can choose one or several correct positions and the right directions to support the piping line, and re-analyze it. Repeat these actions until the results show an acceptable conclusion.

For other excitation mechanism of vibratory fatigue, such as cavitation and flash, as they cannot be predicted accurately, we take the following steps to account for their effect and adopt practical resolutions:

1) To measure the magnitude of the vibration.
2) To calculation the allowable vibration amplitude, and analysis the excitation mechanism.
3) To propose effective and practical modifications of piping support and position.
4) To accomplish the modifications.
5) To measure the magnitude of the vibration again after modification.

The above procedure has been successfully implemented to eliminate or mitigate the vibratory fatigue of piping lines for some native NPPs.
FROM THE FIRE ALARM SYSTEM TO IMPROVE FIRE SAFETY OF NUCLEAR POWER PLANT

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Fire safety is an important component of the safe operation of nuclear power plant, is an important aspect of the security of nuclear safety. In the frequency of fires, or in the extent of the losses caused damage to the nuclear power plant accident both ranked in the forefront. Fire Detect and Alarm System is the front components of the Fire Protection System, while play a role in the rapid detection and alarm. So the reliability of Fire Detect and Alarm System plays an important role in the safety of nuclear plants. In this article, high reliability and sensitivity of the Distributed Intelligent Fire Detect and Alarm System in the nuclear power plant to do some research into the specific application. From the interaction between the Detect System and other Fire Protection Systems, analysis of how to improve the overall system reliability, reduce fire damage. Reduce the impact of fire on the nuclear power plant. Now a variety of advanced technology in the use of Fire Detect and Alarm System, improve the reliability and sensitivity of the system. If the Distributed Intelligent Fire Detect and Alarm System, fire detectors will implantation CPU. Some algorithms and model will be used to analysis the situation of smoke particles quantitative, judge fire from the analysis.

Through automatic environmental compensation to adapt the changes in the situation, so reduce nuisance alarm

The digital bus communication system, a warning signal is transmitted to the fire alarm controller. Bus use transmission loop, if any point break, the detector also can communicate controller, thus ensuring transmission reliability.

In the fire alarm controller, analyze the transmission signal from detector by intelligent control technology. If the detector signal is divided into pre-alarm and fire alarm, the detector send out the pre-alarm, the controller will enhance real-time communication with the detectors, and receive nearby detector signals. If other detectors found the concentration of smoke particles also increase, so the fire is determined.

Meanwhile, there are a number of new fire detection techniques. If Air Sampling Fire Detection, Fiber Temperature Detection System, Prism Flame Detector, Laser Fire Detectors and so on. If using the advance technology to detect the fire as early as possible, to increase the sensitivity of various characteristics of fire. And environmental factors minimize the impact on the detector. Thus, improve the reliability and sensitivity to fire detectors. The operation of nuclear power plants will provide adequate time for intervene the fire. In the early stages of the fire, it will be controlled using artificial means, and to reduce the fire hazard.
Conducting extinguish the fire, we must consider what effect, to avoid unnecessary losses. Such as: cable layer, important equipment room. Fire Detect and Alarm System is must be reliable linkage action with Fire Protection System. System considers "Pre-action", logical is triggered, but fire system is no act. If the fire can’t be controlled by manual, the Fire Protection System automatically moves. Therefore, control fire in the early, this will reduce the fire hazard, but also the secondary effect of the fire.
AGEING MANAGEMENT REVIEW FOR REACTOR INTERNALS OF PWR NUCLEAR POWER PLANT

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To maintain a high level of safety operation for NPP in its whole life, except routine review (review for procedure modification, review for primary events, review for operation experience) and special review (for example review for appearing important event relative with safety) a periodic safety review (PSR) should be carried out every 10 years that involve ageing progressive effect, operation experience and technical developments. Therefore “Guide on Periodic Safety Reviews of Operational Nuclear Plants” Safety Series No. 50-SG-012 was published by IAEA in 1994. “HAF 0312, Periodic Safety Reviews of Operational Nuclear Plants” was also published by NNSA of China in 1999.

Up to now, 300MW nuclear power plant of Qinshan has been operated continuously beyond 10 years. A periodic safety review (PSR) must be done according to the requirements of code above. Ageing management is one of the safety factors for PSR. Reactor internals (RI) is one of the most important objects for ageing management review because reactor internals has high security requirement and its maintenance and replacement cost enormously.

Ageing management review done for reactor internals of Qinshan NPP is based on investigation of practical operation condition, information collection relevant ageing management including design, manufacture, installation, debug, operation, inspection, maintenance etc. the different aspects, summary for 10 years operation experience, analysis important failure events of the equipment in 10 years operation occurrence, profound understanding potential ageing mechanism of reactor internals and attention, in particular, ageing mechanism of reactor internals for Qinshan NPP. Meanwhile, ageing management review done for reactor internals of Qinshan NPP is also based on performance investigation at site and interview and communion with relative personnel of NPP.

Through review ageing mechanisms, ageing effects, ageing managements and ageing conditions of each parts of reactor internals the objective is to examine if reactor internals (RI) is performed the effective management and if there is a reasonable ageing management program for detecting and mitigating of ageing processes or ageing effects to ensure reactor internals (RI) safety operation in aftertime life.

Ageing management review report of reactor internals includes structure description of reactor internals, design requirements and main analysis results in design stage, general operation situation and important failure events of reactor internals since starting service of NPP. Ageing management review report also includes ageing mechanisms summary of reactor internals for Qinshan NPP shown in table 1, ageing management plan elements of reactor internals for Qinshan NPP shown in figure 1 and detailed ageing management review tables. At last review conclusions and recommended correct measures are given.
### Table 1 Ageing mechanisms summary of reactor internals for Qinshan NPP

<table>
<thead>
<tr>
<th>Components</th>
<th>Corrosion</th>
<th>Fatigue</th>
<th>Irradiation embrittlement</th>
<th>Stress relaxation</th>
<th>Wear</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core barrel</td>
<td>Thermal ageing, FIV fatigue</td>
<td>Neutron irradiation embrittlement</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Baffle/former assembly</td>
<td>Thermal ageing</td>
<td>Neutron irradiation embrittlement</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Alignment bolts and pins of baffle/former</td>
<td>SCC, IASCC</td>
<td>FIV fatigue</td>
<td>Worm, relaxation and swelling</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Lower secondary support assembly</td>
<td>FIV fatigue</td>
<td></td>
<td></td>
<td>Stress relaxation</td>
<td></td>
</tr>
<tr>
<td>Hold-down spring</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Support columns / guide columns</td>
<td>FIV fatigue</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Support column pins/fuel pins/alignment bolts</td>
<td>SCC, IASCC</td>
<td>FIV fatigue</td>
<td></td>
<td>Wear</td>
<td></td>
</tr>
<tr>
<td>Irradiation surveillance capsule</td>
<td>FIV fatigue</td>
<td>Neutron irradiation embrittlement</td>
<td></td>
<td>Wear</td>
<td></td>
</tr>
</tbody>
</table>

**FIG. 2 Ageing management plan elements of reactor internals for Qinshan NPP**
AGEING RESEARCH FOR UPGRADES USING DIGITAL I&C SYSTEMS OF NUCLEAR POWER PLANT

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This paper describes philosophy of methodology for management of ageing of I&C systems. These studies were aimed at developing a methodology from an experience based ageing analysis, and applying it to identify the most critical components from ageing and safety points of view. After selection of NPP components important to safety for ageing management studies, we suggest to use the digital technology to completely modernize the reactor protection system of Qinshan nuclear power plant (QNPP), which will improve the reliability and availability of the whole I&C systems of QNPP.

Because Qinshan nuclear power plant is built before the foundation of NNSA, it did not have the PSAR review stage. After the foundation of NNSA, QNPC submit the FSAR to NNSA in 3 delivery package on Sep. 1989. In the FSAR evaluation report (NNSA-0031), it points out some problem for FSAR chapter 7. During the period from Oct. 2001 to Jun. 2004, it has happened 30 equipment failure events. According to the analysis of statistic data of failure cause, we can get the following results: 53% is caused by electronics ageing, 38% is caused by bad connection problem.

Based on the about 10 years operating experience, it shows that the reactor protection system can basically meet the safety objectives requirements of QNPP operation. It is largely benefit from continuous modification of reactor protection system for the good operation. We have completed the partial modification of reactor protection system during commissioning stage. Up to now, we have modified all the modules of reactor protection system except bistable modules and isolation modules.

We will use digital technology to replace the old analog reactor protection systems. The process engineering task description for the digital reactor protection system is divided into four staged detail levels.

The reactor protection system is designed as with fourfold redundant configuration with physical separation. Each of the four redundancy groups comprises equipment for signal acquisition and conditioning, signal processing and component control.

The signal acquisition and conditioning level comprises the sensor supply, fusing and conditioning such as signal distribution. Analog or binary signals from the following systems are processed in the reactor protection system. The TELEPERM XS computers read and digitize the measurement signals for data acquisition and processing, generate the initiation criteria and gate these with actuation signals in accordance with the specifications of the process engineering task description. The actuation signals for the reactor trip and ESFAS functions are generated on various computers.
The four redundant data RPS processing computers in a diverse mode group are connected to each other by means of optical TXS Profibus such that each computer has a direct communication link to each of the other three redundant computers. Further optical TXS Profibus links are implemented between all the automation computers (RPS processing computers and ESF actuation computers) in a single redundancy group and the redundant Message/Service Interface (MSI) computers. All the computers in a safety I&C redundancy group are connected to the dedicated MSI by means of TXS Profibus in point-to-point topology. The Plant Computer System (PCS) is connected to the redundant MSI computers by means of two redundant Gateway computers and separate TXS Ethernet connections.

TELEPERM XS safety I&C systems require virtually no maintenance and allow a major reduction of service effort compared with predecessor hardwired systems. TELEPERM XS hardware components are robust. Figure 1 illustrates the failure rates of TELEPERM XS systems hardware.

We will carry through the ageing studies for the new digital I&C system.

![FIG 1 : Failure rates of TELEPERM XS systems hardware](image)
VIBRATION FATIGUE ANALYSIS FOR MAIN STEAM PIPELINES ON A NPP

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The vibration and noise of pipings adjacent to isolation valves of main steam pipelines in a NPP were excessive during operation. To assess the safety in operation of main steam pipelines, both stress and fatigue analysis were performed using ANSYS program, and evaluation was performed according to ASME code.

The main steam pipeline considered in this report, includes the main steam piping, safety valves, main steam isolation valve and other branch pipings between the containment penetration and the first transverse direction limit stop downstream of the isolation valve. The portion of main steam pipeline is classified as safety-related piping, Class 2 and Seismic Category I. The operating temperature of main steam pipeline is 290°C. Based on the requirements of ASME OM-S/G-2000 Part3, the alternating stress intensity $S_{alt}$ caused by vibration of the pipeline should meet: $S_{alt} = \frac{C_5 K_z^2}{Z} M \leq S_{alt}$.

The 3D pipe element PIPE16 of ANSYS was used to build the calculation model of FEM for main steam pipeline. The model was shown in Figure 1. The model is constrained at the containment penetration and the first transverse direction limit stop downstream of the isolation valve. The vertical supports at elbow bends of atmospheric relief valves were regarded as vertical restraining points. The limit supports of exhaust pipings were regarded as guide. Both weight of insulation and steam in pipings were considered in calculation.

The calculation model was verified by measured the natural frequency of pipeline. The first frequency of FEM obtained by dynamic modal analysis was 13.6Hz, and the measured value was 15.0Hz. Two values were closed enough. It was shown that the dynamic FEM of pipeline was appropriate and could be used in corresponding stress calculation.

The stress of pipeline was calculated in static analysis method. The measured maximum vibration displacement values of main steam piping, main isolation valve and safety valves were input in analysis.

The tee joints between the safety valves, isolation valve and the main steam piping, and transverse direction limit stop downstream of the isolation valve were selected to evaluate (see Figure 2). The loads were applied to these components in different direction and the maximum values of results were combined in RMS method. The alternating stress intensity $S_{alt}$ listed in Table 1 was obtained by multiplying the combined values with stressintensification factor.

It was shown through the results of fatigue evaluation that the alternating stress intensity caused by vibration of main steam pipeline were less than the allowable limitation of fatigue intensity and it could meet the safety requirement in its lifetime. However the in-service inspection to welds of main steam piping should be performed in operation to assure its...
integrity. Also the wall thickness of main steam piping should be detected to meet the design requirement.

Table 1. The results of stress calculation and evaluation

<table>
<thead>
<tr>
<th>Analytic component</th>
<th>Maximum value (MPa)</th>
<th>Combination of stress intensity (MPa)</th>
<th>$C_2K_2$</th>
<th>Alternating stress intensity $S_{alt}$ (MPa)</th>
<th>$S_{alt}$/limited value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Safety valve 119VV</td>
<td>1.2678 2.1209 0.5345</td>
<td>2.528</td>
<td>3.21</td>
<td>8.11</td>
<td>0.17</td>
</tr>
<tr>
<td>Safety valve 116VV</td>
<td>1.1960 2.9772 0.3861</td>
<td>3.232</td>
<td>3.21</td>
<td>10.37</td>
<td>0.22</td>
</tr>
<tr>
<td>Safety valve 113VV</td>
<td>1.1419 2.9383 0.5121</td>
<td>3.194</td>
<td>3.21</td>
<td>10.25</td>
<td>0.21</td>
</tr>
<tr>
<td>Safety valve 110VV</td>
<td>1.3449 2.1369 1.6469</td>
<td>3.015</td>
<td>3.21</td>
<td>9.68</td>
<td>0.20</td>
</tr>
<tr>
<td>Safety valve 117VV</td>
<td>2.1270 2.4972 2.6917</td>
<td>4.243</td>
<td>3.21</td>
<td>13.62</td>
<td>0.28</td>
</tr>
<tr>
<td>Safety valve 104VV</td>
<td>5.7761 4.7434 4.8946</td>
<td>8.934</td>
<td>3.21</td>
<td>28.68</td>
<td>0.60</td>
</tr>
<tr>
<td>Safety valve 101VV</td>
<td>4.0175 4.4251 4.4918</td>
<td>7.477</td>
<td>3.21</td>
<td>24.00</td>
<td>0.50</td>
</tr>
<tr>
<td>Isolation valve</td>
<td>2.9736 4.0927 6.1014</td>
<td>7.926</td>
<td>3.21</td>
<td>25.44</td>
<td>0.53</td>
</tr>
<tr>
<td>Limit stop</td>
<td>4.2038 4.5697 6.1014</td>
<td>8.705</td>
<td>3.85</td>
<td>33.51</td>
<td>0.58</td>
</tr>
</tbody>
</table>
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AGEING MANAGEMENT OF CARBON STEEL PIPINGS IN 300Mwe PWR SECONDARY SYSTEMS

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1. Introduction

300 MWe Pressurized Water Reactor (PWR) nuclear power plant, which was designed by Shanghai Nuclear Engineering Research and Design Institute, is the first nuclear power plant in mainland of China. With the operation of nuclear power plant, ageing management of some components and pipings are being performed. For pipings in the secondary systems, wall thickness of carbon steel pipes, which is susceptible to flow-accelerated corrosion, will be monitored through the lifetime of nuclear power plant. The wall thickness monitoring is one part of ageing management of pipings. In this paper, the selection of pipings in the 300 MWe PWR nuclear power plant secondary systems for ageing management according to flow-accelerated corrosion mechanism is described.

2. Ageing Mechanism of Flow-Accelerated Corrosion

Carbon steel are used extensively in nuclear power plant secondary systems, such as turbine extraction steam system, condensate system, heater drain system etc. In high temperature water, a layer film of Fe_3O_4 is formed to protect carbon steel against further corrosion. If the Fe_3O_4 layer dissolves in turbulent water or wet steam, accelerated corrosion occurs in the unprotected carbon steel. Flow-accelerated corrosion is one of the most frequently experienced causes of component failures for power plant, and many researches have been done in laboratories all over the world. Flow-accelerated corrosion is a chemical corrosion process, and some factors such as thermal-hydraulic properties, water chemistry and composition of material can affect it. In nuclear power plant secondary systems, temperature, velocity, two-phase flow and pipe geometry are the most important factors.

3. Information on 300 MWe PWR Nuclear Power Plant Secondary Systems

The main function of nuclear power plant secondary systems is to supply saturated steam from steam generators to steam turboset for electric power generation. Secondary systems mainly consist of turbine, electric generator, condenser, condensate pump, heater, deaerator, feedwater pump, moisture separator reheater and instruments, valves, pipes etc. In addition, auxiliary systems such as main steam bypass system, recirculation cooling water system, control protection system, and lubricant system etc are also included in secondary systems.

According to experiences of flow-accelerated corrosion in other nuclear power plants secondary systems and data of wall thickness measurement, the pipings in turbine extraction steam system, main steam system, 1st and 2nd stage reheat steam system, high pressure feedwater system, high pressure heaters drain systems and MSR drain system need to be focused on for ageing management. The information on these systems are listed in Table 1.
Table 1  Information on 300 MWe PWR Nuclear Power Plant Secondary Systems

<table>
<thead>
<tr>
<th>System</th>
<th>Temperature</th>
<th>Fluid</th>
<th>Velocity</th>
<th>Material</th>
</tr>
</thead>
<tbody>
<tr>
<td>Turbine Extraction Steam System</td>
<td>110°C—230°C</td>
<td>Wet Steam</td>
<td>20m/s—70m/s</td>
<td>St45.8, Q235B</td>
</tr>
<tr>
<td>Main Steam System</td>
<td>270°C</td>
<td>Wet Steam</td>
<td>70m/s</td>
<td>SA-106B</td>
</tr>
<tr>
<td>Condensate system</td>
<td>50°C</td>
<td>Single-phase Flow</td>
<td>2m/s—6m/s</td>
<td>St45.8</td>
</tr>
<tr>
<td>1st Stage Reheat Steam System</td>
<td>230°C</td>
<td>Wet Steam</td>
<td>30m/s—50m/s</td>
<td>St45.8</td>
</tr>
<tr>
<td>2nd Stage Reheat Steam System</td>
<td>270°C</td>
<td>Wet Steam</td>
<td>25m/s—100m/s</td>
<td>SA-106B</td>
</tr>
<tr>
<td>High Pressure Heaters Drain Systems</td>
<td>140°C—200°C</td>
<td>Single-phase Flow</td>
<td>0.3m/s—0.5m/s</td>
<td>St45.8</td>
</tr>
<tr>
<td>High Pressure Feedwater System</td>
<td>130°C—220°C</td>
<td>Single-phase Flow</td>
<td>3m/s—6m/s</td>
<td>SA-106B</td>
</tr>
<tr>
<td>MSR Drain System</td>
<td>170°C</td>
<td>Single-phase Flow</td>
<td>0.4m/s—1m/s</td>
<td>St45.8</td>
</tr>
</tbody>
</table>

4 Critical Pipings for Ageing Management

Three types of carbon steel of St45.8, Q235B and SA-106B are used in the pipings of 300 MWe PWR nuclear power plant secondary systems. For St45.8 and Q235B, Cr content is not specified in the material specification. For SA-106B, although Cr content is required to control in the range of 0—0.4%, actually there is little Cr in the pipe. From the view of material, these types of carbon steel could not be resistant to flow-accelerated corrosion.

The fluid in turbine extraction steam system and 1st and 2nd stage reheat steam system are two-phase flow. Due to temperature of more than 100°C and velocity of more than 30m/s, flow-accelerated corrosion possibly occurs, especially in No.3 extraction steam system and 1st stage reheat steam system. For high pressure feedwater system, the feedwater pump outlet pipe, in which single-phase flow move, is susceptible to flow-accelerated corrosion due to velocity of 6m/s and temperature of 130°C. In MSR drain system and HP-5 heater drain system, the fluid are single–phase flow with temperature of 170°C and 140°C, respectively. The temperature are near to that of maximum flow-accelerated corrosion rate for single–phase flow, so flow-accelerated corrosion possibly occurs.

From the view of pipe geometry, the geometrical arrangements, such as orifice, valve, bent etc which induce high turbulence into the flow, lead to higher flow-accelerated corrosion rate. When wall thickness are measured in the pipings mentioned above, orifice and valve downstream pipes are the important region.
5 Summary

Ageing management of carbon steel pipings in secondary systems is an important part of ageing management of nuclear power plant, and concerns plant safety, efficiency and economic operation. For 300 MWe PWR nuclear power plant secondary systems, the pipings in turbine extraction steam system, 1st stage reheat steam system, MSR drain system, high pressure heater drain system and high pressure feedwater pump outlet pipe are the key components in ageing management. In addition to measurement of wall thickness, estimation of thickness thinning rate and assessment of reduced area will be performed for ageing management.

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AGING DEGREDATION PROBLEMS AND SOME COUNTERMEASURE CONSIDERATIONS OF PWR RPV AND RVI

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During the period of NPP operation, with the increasing of service time, systems, structures and components (SSCs) shall be affected by one or several degradation factors, such as general and local corrosion, erosion-corrosion, radiation and thermally induced embrittlement, fatigue, corrosion fatigue, creep, binding and wear, probably causing loss of safety function and availability. These factors should be considered during the design stage and ageing degradation in nuclear power plants must be effectively managed to ensure that the design functions remain available throughout the service life of the plant. From the safety perspective, this means, within acceptable limits, controlling ageing degradation of the systems, structures and components (SSCs) important to safety so that adequate safety margins could be kept. On the other words, the required integrity and functional capability of NPP SSCs are controlled under their normal operating requirements.

PWR pressure vessel is the most important pressure boundary component because its function is to contain the nuclear core under elevated pressure and temperature. And RPV functions are also to provide structure support for the reactor vessel internals and the core. In addition, RPV is the biggest component and can not be replaced during the plant service life. That means the plant service life depends on the life of RPV. The reactor vessel internals is to support the core, the control rod assemblis, the core support structure, and the reactor pressure vessel surveillance capsules. The reactor internals have the additional function to direct the flow of the reactor coolant and provide shielding for the reactor pressure vessel. Pressure light water reactor RPV and RVI experience service at the environment of high temperature, high pressure and high neutron fluence which will inevitable cause the ageing degradation.

The objective of this paper is to analysis the main ageing mechanisms and affect factors of PWR RPV and RVI according to their characteristic and then, give out the countermeasures. For pressure water reactor RPV, the main ageing degradation includes radiation embrittlement, thermal ageing, temper embrittlement, fatigue, corrosion and wear. Among them, the most important ageing is the radiation embrittlement what affected by neutron fluence, alloy composition, radiation temperature, microstructural characteristics and the neutron flux energy spectrum and so on. For the corrosion, the important problems are primary water stress corrosion cracking (PWSCC) of CRDM penetration, general corrosion and pitting on the inside surface, boric acid corrosion of outer surface.

For PWR vessel internals, the main ageing degradation includes radiation embrittlement, fatigue, irradiation accelerated stress corrosion cracking (IASCC), corrosion (such as general corrosion, SCC, erosion-corrosion), radiation induced creep, relaxation and swelling, mechanical wear and so on.
Considering the above ageing degradation mechanism, the countermeasure will be related to structure design and material improved, strict control the operation environment and condition, the way of in service inspection and mantance, etc.

For prolonging RPV service life, on the RPV structure design, the effect of neutron flux should be considered as low as possible. And, it is better that welding is not located at the zone of core belt to avoid the welding exposure to the biggest radiation dosage zone.

For prolonging RVI service life, the key factor is understanding the neutron flux distribution and predict the radiation embrittlement situation according to the current radiation embrittlement data base or tendency curve.
NEUTRON MONITORING SYSTEM AND ROD CONTROL SYSTEM UPGRADES FOR PLANT LIFE EXTENSION

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Most nuclear power plants in operation were built in the 1970’s and 1980’s. The most challenging aspect of I&C upgrades for these plants is the upgrade of the neutron monitoring system (NMS) and the Reactor Manual Control System (RMCS). These are specialized instrumentation and would require high quality design and engineering products to ensure safe and efficient plant operation. Specifically, GE Energy’s nuclear business provides a Wide Range Neutron Monitoring System (WRNM) to replace the existing Source Range Monitors (SRM) and Intermediate Range Monitors (IRM), a Power Range Neutron Monitoring System (PRNM) to replace the Average Range Power Monitor (APRM), and a Rod Control Management System (RCMS) to replace the original RCMS in the GE designed Boiling Water Reactors (BWR). The WRNM, PRNM, and RCMS are based on the Nuclear Monitoring Analysis and Control (NUMAC) platform, which is a microprocessor based system that provides improved system performance with standard features such as improved HMI, self-test and automatic calibration.

The WRNM is an example of design simplification that provides significant improvement over the current SRM and IRM. Both SRM and IRM are movable detectors and require mechanical equipment and external forces to drive the detectors into the core for a reading. This presents potential issues with stuck detectors or failure of mechanical equipment. The WRNM is a fixed in-core detector system. This allows the design simplification in eliminating the mechanical equipment associated with the SRM and IRM. Additionally, this design simplification reduces the dose for any outage maintenance work on the mechanical equipment. The WRNM further reduces the number of total detectors. Typical plants with 4 SRM detectors and 8 IRM detectors can be replaced with a system with 8 WRNM detectors. The NUMAC electronics also provide simplification for operations. The IRM is a manual range-based system. Thus, the operator has to change the range during startup or shutdown to prevent a reactor trip. The WRNM utilizes the NUMAC electronics to operate in an automatic range changing mode and replace the high level trip of the ranges with a period based trip. This allows the operator to focus on rod movements during startup and shutdown.

The PRNM enhances the performance as well as expanding the capability of the existing APRM. The typical APRM system is a 6-channel system. This creates a complexity for the Reactor Protection System (RPS), which is essentially a 4-channel system with a 1- out-of-2-twice logic system. The PRNM changes the APRM system into a 4-channel system. It incorporates a two-out-of-four logic voter that requires two positive channel inputs to initiates a trip signal to the RPS. This design eliminates the PRNM as a half scram contributor. It also allows the plant to operate with one channel being bypassed for maintenance purpose. The PRNM provides the capability for automatic uploading of the detector calibration data and downloading of the detector readings. It also allows the operation of a running detector IV curve without special setup. The most significant improvement of the PRNM is the ability to implement various long-term stability solutions without requiring hardware changes. The long-term stability solution is a regulatory
requirement and will be needed to support any plant performance enhancement, e.g. extended power uprate.

The RCMS replaces the existing RMCS and other rod control system, i.e., the RPI, RDCS, RWM, and Operator Control and Display, with one fully integrated system that provides redundancy and expanded capability for maintenance and operations, e.g. rod stroking and scram timing. The new system replaces the Transponder cards and Branch Amplifiers for the CRD hydraulics to improve their reliability. The position feedback feature ensures that the rod will stop at the targeted location. Additional capability such as rod prompting, pre-loading of rod sequences, continued operation with bypass or failure of a Transponder or Branch Amplifier, automatic rod drift suppression, and intelligent rod substitute position provide significant improvement in system operation. The HMI includes a core map with touch screen capability that allows the operator to recognize any equipment issue.

One of the NUMAC design philosophies is to apply modular design for hardware and software. The modular design minimizes the impact on replacement of obsolete parts. As time progresses, the current design of the NUMAC system, such as WRNM, PRNM and RCMS, will become obsolete also. However, new design of the NUMAC instruments will be developed to maintain the design viability. For example, the WRNM and PRNM to be implemented in ABWR and ESBWR will have new CPU designs due to the obsolescence of the CPU chip. However, the design of the new CPU does not require a re-design of the entire system. Therefore, the presented system can be maintained to support the plant operation for plant life extension.
REGULATORY ISSUES IN SPENT NUCLEAR FUEL MANAGEMENT IN BELARUS

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The report addresses regulatory issues in safe storage of spent fuel from mobile nuclear power plant (MNPP) “Pamir-630D” with 630 kW that was in operation during three years. That nuclear power plant consisted of gas cooled reactor (weight of 76 500kg), gas turbine-driven set (76 000kg), two control units (2×20 000kg), and auxiliary unit (20 000kg). The reactor and turbine-driven set were supposed to be put on transport platforms and carried by tractors. The control and auxiliary units were set on track beds. “Pamir-630D” was constructed and tested in appropriate building. The reactor core (height of 0.5 m and diameter of 0.506 m) consisted of 106 fuel assemblies, each of them contained 7 fuel rods surrounded by stainless steel claddings of wall (thickness 4×10^{-4} m and diameter of 6.2 ×10^{-3} m). Fuel spherical particles of UO_{2} enriched to 45% 235U were embedded in Ni-Cr matrix. The share of nickel and chrome in fuel composition was 40%. Weight of 235U in reactor core was 18.7 kg; weight of the reactor core - 5700 kg.

The mobile nuclear power plant “Pamir-630D” was shut down 26.11.1987 and up to 24.06.1988 the reactor was cooled by gas-liquid coolant circulation in main loop. After that time the reactor was cooled by liquid coolant circulation through auxiliary loop of accident cooling system of the reactor. The coolant was removed from all circuits 22.05.1989 and the reactor was filled in nitrogen and cooled by it until removing the fuel (January 1991).

The decommissioning plan included a short period of preparation for disposal followed by reactor and turbine units of “Pamir-630D” dismantling and safe short-term keeping of spent fuel in an appropriate temporary storage facility. Special equipment included tank for temporary keeping of radioactive pieces of reactor, turntable of biological shielding and turntable with devices and tank (height of 5.3m, a diameter of 2m) for the removing of the fuel assemblies which were designed and constructed to unload the reactor core under water. A gas tightness of the assemblies was checked under water. The removed fuel assemblies were placed into the flasks under water after a gas tightness checking. The fuel assemblies with cladding defects were placed in addition into sealed pills.

The temporary storage facility consisting of two pools (volume of 2×28 m^3) with a water shielding (thickness of 3.1m) and a concrete shielding (thickness of 1.8m) was built in the reactor hall. Since then the spent fuel assemblies have been kept under water. Surveillance, monitoring and inspections are carried out to ensure that the spent fuel storage remains in good condition. The water quality is maintained in accordance with appropriate chemistry requirements. The flasks with the fuel assemblies are tested for leak proofness by the check weighting under water quarterly if its weight is stable and weekly in case the weight changes. The flask is removed for testing and correcting the leak proofness in hot cell in accordance with appropriate procedure if the reference weight increasing is 0.1kg. Support plates on the bottom of the pools were presumably designed to space the fuel (in their storage cassettes) far enough apart to avoid criticality safety issues. The pool lids are
sealed by the IAEA inspectors.

At the time a decision was taken to shut down and decommission the mobile nuclear power plant it was foreseen that the spent fuel would be shipped to Russia afterwards. Unfortunately, due to the break-up of the Soviet Union, the spent fuel was left in the territory of the Republic of Belarus. As a Contracting Party to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, Belarus has a national policy for the management of spent fuel, in order to ensure that acceptable levels of protection of human health and environment, now and in the future, can be adequately achieved without composing undue burdens on the future generations. The national legislative and regulatory system for safety of spent nuclear management in Belarus deals with follow types of challenge.

The pools were built for MNPP operation and are not suitable for long term keeping of spent fuel. So, the ageing management programs have to be elaborated as soon as possible. We are also faced with using spent fuel facility that was not designed and constructed in accordance with the current knowledge and experience base and design criteria. The reprocessing technology for such kind of nuclear fuel has not been developed. Taking into account the fact that this particular sort of fuel has been designed only for “Pamir-630D” and has not been used in any other types of reactor, technology of its reprocessing perhaps will not be elaborated in the nearest future, so it is necessary to provide long-term safe keeping of spent fuel. However, the lack of financial resources for long-time storage, ageing the equipment of storage facility will not allow assuring safe storage of spent fuel in future. There is neither appropriate dry storage facility no reprocessing plant for spent fuel in Belarus, so the ‘wait and see’ option means passing an issue on to future generations. For Belarus the costs of sitting and developing a geological repository are enormous. Moreover, Belarus also has not suitable geological formation to host a geological repository.

Since decommission of MNPP the storage facility staff has consisted of operative personnel of nuclear reactor. Number of workers is being decreased and, moreover, now most of them have near pension age. Nonetheless, no one young scientist or engineer has joined spent nuclear fuel facility. May be in ten years we will have the situation when no one will be familiar with storage facility and spent nuclear fuel management.

Unfortunately, decommission fund was not established and budget financial resources for long-time storage of spent nuclear fuel is not enough. Now regulatory body has no possibility to change this situation. Besides, under the Convention on the Physical Protection of Nuclear Material this fuel has become Category I material due to the decrease in the radiation level. It requires application of strengthened physical protection measures. The best solution of this problem is to return the spent fuel from “Pamir-630D” to a reprocessing plant of the Russian Federation as it had been planned before decommissioning of MNPP in 1987. A positive decision on its return to the Russian Federation with the IAEA assistance will contribute to strengthening the physical protection as well as reducing its maintenance expenditures.
REGULATIVE ASPECTS AND DECISIONS AT LICENSING CONTEMPORARY I&C CHANGES OF NPP WITH WWER REACTORS IN THE CONTEXT OF PLiM

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Before 1996 NPP Kozloduy started producing a programme of modernization /PM/, for an increasing the safety of 5 and 6 units. PM crosses performance of a string from technical decisions and her performance is degree of the performance on PLiM - NPP Kozloduy.

The changes in the codex of the Bulgarian nuclear law from 2004, and the acceptance and implementation of the Modernization Program (PM) of the Nuclear Power Plant (NPP) aim not only to improve and ensure safety of the plant, but also to advance Bulgarian standards in nuclear safety to the criteria and requirements in Europe and the World.

An important part of the Modernization Program is the implementation of technical programs for upgrade of the I&C systems and components. The Bulgarian Regulator faced a series of problems at the time of reviewing, analyzing, assessing and licensing the new digital I&C systems, components and devices:

- Seismic qualification.
- Redundancy
- Diversity
- Reliability
- Principle of single mode failure
- Independence between separating loops (managing and controlling)
- Validation and verification of the modernizations. Demonstration of its good working condition and capability.
- Metrological assessment of the new elements and systems.
- Comparison between Bulgarian and other standards.
- Classification on safety of the new systems, devices and components

In the paper-presentation, I present to your attention a summary-analysis of the regulatory process. I also review specific practical examples, which capture the overall regulatory process and what is executed by PM of part I&C.

The changes in the codex of the Bulgarian nuclear law from 2004, and the acceptance and implementation of the Modernization Program (MP) of the Nuclear Power Plant (NPP) aim not only to improve and ensure safety of the plant, but also to advance Bulgarian standards in nuclear safety to the criteria and requirements in Europe and the World. An important part of the Modernization Program (PM) is the implementation of technical projects for upgrade of the I&C systems and components. The Bulgarian Regulator faced a series of problems at the time of reviewing, analysing, assessing and licensing the new digital I&C systems, components and devices.
The cover of PM in an digital I&C part are the following systems:

Unification Complex Technical Devices for reactor hall (UKTS–I), Unification Complex Technical Devices for turbine hall (UKTS–II), Computer Information System (CIS), Automated Turbine System (ASUT). These systems are Westinghouse provided with – OVATION controller, double redundancy, 3 channels.

For example: a functions and subsystems UKTS – 1 are showing in the table 1

<table>
<thead>
<tr>
<th>Functional system (s)</th>
<th>Process system (s)</th>
<th>Function (s)</th>
<th>Drive</th>
<th>Safety class according to IEC(1)</th>
</tr>
</thead>
<tbody>
<tr>
<td>SIN40</td>
<td>UV, TL, TP, TU, TG, RY, TK, TS, UG, YC, YD; YD50,60; YT (4)</td>
<td>Primary circuit signalization</td>
<td>C (5)</td>
<td></td>
</tr>
<tr>
<td>SIN90</td>
<td>(Not shown) (4)</td>
<td>Signalization of the Reactor control and protection system</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>TAN00</td>
<td>TA20</td>
<td>Primary circuit Oil system</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>TBN10</td>
<td>TB10, TB11, TB12, TP00</td>
<td>Concentrated boric acid solution system Primary circuit</td>
<td>B (a, b) (2)</td>
<td></td>
</tr>
<tr>
<td>TBN20</td>
<td>TB20, TP00</td>
<td>Maintenance of the water chemistry in primary circuit</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>TBN30</td>
<td>TB30</td>
<td>System borated water primary circuit on</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>TCN10, TCN20, TCN30, TCN40</td>
<td>TC00, TC01 TC10, 20, 30, 40 YA12, 22, 32, 42</td>
<td>System for bypass purification of Primary circuit</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>TEN00, TEN10, TEN20</td>
<td>TE00, TK80 TB60, TE10, TE11, TEC1 TB70, TE20, TE21, TEC2</td>
<td>System for purification of Primary circuit and controlled drainage system</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>TGN21</td>
<td>TG11, TG12, TG13, TG20, TG21</td>
<td>Cooling of the spent fuel pool</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>Functional system(s)</td>
<td>Process system(s)</td>
<td>Function(s)</td>
<td>Drive</td>
<td>Safety class according to IEC(^1)</td>
</tr>
<tr>
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</tr>
<tr>
<td>TKN00, TKN21, TKN22, TKN23, TKN99</td>
<td>TF10, TK10—14,20,31,32,40-44,70,71,80, VF33, YC10, YC20, YC30, YP10 TK21, TK91, TKC2, VF13 TK22, TK92, TKC2, VF23 TK23, TK93, TKKC2, VF33 TK10, TK13, TK14, TK70, TK80, TKC0, TKC1, TKC7</td>
<td>Primary feed and bleed system (PFBS)</td>
<td>B (^3)</td>
<td></td>
</tr>
<tr>
<td>TLN02</td>
<td>TL02, TQ30, UM44, VB81, VB82, VB83</td>
<td>Air purification system in the restricted area</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>TLN03</td>
<td>TL03, VB50, YC01</td>
<td>Control rods cooling system</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>TNN00</td>
<td>TB40, TN20, TN21, TN22, TN23, TN40</td>
<td>Clean condensate system</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>TQN00</td>
<td>TQ12, TQ13, TQ14, TQ22, TQ23, TQ24, TQ32, TQ33, TQ34</td>
<td>Emergency core cooling, residual heat removal, Emergency high and low pressure baric acid injection system</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>TSN10</td>
<td>TK10, TS10, TS11, TS12, TS13, TS14, TS15</td>
<td>Hydrogen burning system</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>TVN00</td>
<td>TV11, TV31, TV41, TV51</td>
<td>Reactor Building Sampling System</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>TXN00</td>
<td>TX10, TX20, TX30, TX50, TX60, TX70, TX80, YB10, YB20, YB30, YB40</td>
<td>Emergency feed water system</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>TYN00</td>
<td>TY16, TY17, TY18, TY19, UR20</td>
<td>Controlled drainages and leakages primary circuit</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>Functional system (s)</td>
<td>Process system (s)</td>
<td>Function (s)</td>
<td>Drive</td>
<td>Safety class according to IEC$^{(1)}$</td>
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</tr>
<tr>
<td>TZN00</td>
<td>TP00, TZ00, TZ01, TZ20, TZ21</td>
<td>Primary circuit special drainages</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>UJN00</td>
<td>UJ06, UJ14</td>
<td>Fire protection system and fire alarm system</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>UVN06</td>
<td>UM45, UV06, UV55</td>
<td>Ventilation system in the Main Control Room MCR and in the Auxiliary control room (ACR)</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>UXN00</td>
<td>TL05, TL48, UX11, UX12, UX21</td>
<td>Cooling of the air in the compartments of the water supply systems</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>YBN00</td>
<td>RY10-14,21-24,30-34; YA10, YB10, YB20, YB30, YB40</td>
<td>Steam Generator Blow down System blow down water</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>YDN10, YDN20, YDN30, YDN40</td>
<td>TK30, TK51, VB11, YA10, YB10, YD10, YDC1 TK30, TK52, VB12, YA20, YB20, YD20, YD21, YDC1 TK30, TK53, VB13, YA30, YB30, YD30, YD31, YDC31 TK30, TK54, VB14, YA40, YB40, YD40, YD41, YDC1</td>
<td>Main circulation pumps – MCP 1, 2, 3, 4</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>YDN50, YDN60</td>
<td>YD50, YD51, YD52, YD53, YD71, YD72, VB31 YD60, YD61, YD62, YD63, YD81, YD82</td>
<td>Lubricating Oil System of the MCP</td>
<td>C</td>
<td></td>
</tr>
<tr>
<td>YPN00, YPN99</td>
<td>TN33, TP20, YP10, YP11, YP12, YP13, YP20</td>
<td>Pressurize System</td>
<td>B(a)$^{(3)}$</td>
<td></td>
</tr>
</tbody>
</table>
What problems in process of make a decision to license the system had Regulator? Classification on safety of the new systems, devices and components. We have to have compare between Bulgarian standards and IEC 61 226.

Seismic qualification systems, subsystems and components UKTS – 1 are different from old system. Old system has I category but new has II category, but old UKTS cover UKTS – 1, UKTS – II, UKTS – safety systems. We make assessment on functions and equipment new UKTS-1, are determining requirements from the mind of safety. After that we make decision that II category is correctly from UKTS – 1.

About Redundancy, Diversity, Reliability, Principle of single mode failure, Independence between separating loops (managing and controlling) we are not difficulty

Validation and verification of the modernizations. Demonstration of its good working condition and capability – these requirements we check with inspections. We check FAT, SAT, demonstration and results from vendor and NPP personal.

II. Other I&C digital systems:

- KUS 95 – Ensure Detection of Loose Parts System- Siemens, signal to CIS.
- Hydrogen Detection and Recombination System – with Siemens help, signal to CIS.
- RMS – Radiation Monitoring System
- Automated Device for Cold Overpressure Protection – Siemens, Teleperm – controller.

FLUES – System for Quick Detection and Localization of Leakage from the Reactor Upper Bloc and the Primary side – Siemens , signal to CIS.
FAMOS – System On-Line Monitoring of Component Fatigue – AREVA NP , signal to CIS.

In the finish of paper we have to mark that the replacement the ISC equipment with new digital, micro controller based are important for safety service life cycle NPP.
APPRAISAL AND COUNTERMEASURE OF PRESSURE PIPE LINES OF CI AND BOP

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Combining situation of daya bay nuclear power plant, according to industry pipeline (GC) the pressure pipelines of CI and BOP is divided into three kinds according to regulation\textsuperscript{[1]}. Selected pipelines is divided into three parts: 336 GC first class, 503 GC second class, 2057 GC third class. The main liquid medium in pipelines are steam, water and hydrogen etc. Hydrogen belongs to combustible gas of A-class according to GBJ16-87\textsuperscript{[2]}. There are 174 hydrogen pipelines with the diameters less than 25 mms, but they should be always the prime consideration.

Considering equipment of nuclear power plants and personal safety, GC first class pipeline will lead to serious harm once its confine is destroyed because of its high parameter. Therefore GC first class pipeline with the diameters more than 50 mms should be supervised principally, there are 334 such pipelines in daya bay.

Outside oxide film which gets in touch with the fluid will become thin and its protectiveness will be reduced when the natural oxide film of carbon steel or the low-alloy steel fused into mobile water or water-steam compound. Therefore the material will be corroded more rapidly. In a stable flow accelerated corrosion(FAC) state the material corrosion rate is equal to the oxide film resolving rate, and the process will continue with the operation of nuclear power plants\textsuperscript{[3]}. Feedwater pipeline, condensate pipeline, drain pipeline and partial steam pipeline of nuclear power plants are made of carbon steel, FAC will do harm to them during the long operation of nuclear power plants, so the pipeline should be tracked classified and supervised.

The pipelines of 22 systems is divided into three parts 336 F first class, 811 F second class, 1749 F third class by FAC criterion. The F third class pipelines is not influenced by FAC usually, so it is not in safe management outline.

Ultrasonic thickness measurement is used as a chief means of measuring residual thickness of pipe fittings during FAC track management. The criterion of choosing pipe fittings are as follows: 1. the pipe fittings of changing flow pattern, such as valve, pore plate, elbow etc; 2. complex geometric place, such as pipelines which distance is no more than twice diameters; 3. the components with support ring or counterbore.

When we inspect the multirow parallel pipelines whose parameter and position are similar, personal safety should be considered first. We should increase measuring point nearby or extend inspection of other pipelines when we have discovered that wall thickness is abnormal. Pipe fittings behind of control valve or pore plate of pipelines should be primary examined, the pipelines should meet the condition synchronal as follows: first fluid medium is wet steam or water; second design temperature is greater than 125°C; third pressure loss is...
greater than 0.4MPa; forth pipelines’ material is unalloyed steel.

The FAC first class pipelines in the light of important pipelines is mostly managed in daya bay. The thickness of selected pipe fittings was measured by stages and the measure result was input to database which EDF exploited for nuclear power plants. The database can calculate and analyze the FAC of pipelines. Finally, an evaluation of pipelines life analysis was gained.

Inspection place of FAC second class pipelines should be chose conservatively and the factors took into account are as follows: design parameter, operation condition, industrial experience and the specific condition of nuclear power plants. We draw lessons from foreign nuclear power plants presently and frame inspection plan based on analysis of sensitive region periodicity. The pipelines should be spot-checked termly considering its hazard degree.

Analysis production of reliability-centered maintenance (RCM) has been applied since 1999 in daya bay nuclear power plant and good success has been gained in pipelines of main system. We emphasized particularly on production of RCM analysis when we arranged inspection plan of pipeline. And it enhanced the level of life management of pipelines completely.

Furthermore we tracked experience feedback of national and international nuclear power plants nearly in daya bay and evaluated problem discovered on job site in time. The stress formed by vibration or blocked thermal expansion is maximal on elbow or end commonly[4]. Combining internal and external feedback and general investigation and checkout we chose 266 vibration pipe fittings and 758 sensitive apparatus pipe fittings. We mostly managed these pipe fittings and shorten their inspection cycle.

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DESIGN OF EARTHQUAKE RESISTANCE ENHANCEMENTS OF THE DUKOVANY NUCLEAR POWER PLANT BUILDING STRUCTURES.

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Keywords: nuclear power plant, earthquake resistance, safety related structures, seismic response.

At present time nuclear power plants of the WWER type operate for about three decades. In Czech Republic, the Dukovany NPP of the VVER-400/213 type has been put into operation in 1985. A steady high level of plant’s operational availability and reliability has been proved. No actual problems with operational safety have been noticed. An effective way of managing updated safety requirements ensures a high safety level during the whole plant’s design life. However, in accordance with general trends, there are increasing requirements of upgrading the plant and operating the plant in excess of its design life. The development of an effective system of management of ageing effects in structures and components is the key for maintaining the operational safety during the extended plant’s operating lifetime. Moreover, due to steadily increasing additional safety requirements, the operational safety has to be enhanced. Besides many other important problems, those related to earthquake have to be considered. The earthquake resistance of building structures during the predicted residual life has to be investigated, considering both updated and newly postulated earthquake risks.

The contributed paper deals with selected problems of enhancing earthquake resistance of safety related building structures of the NPP Dukovany.

A sophisticated spatial computation model with 427442 degrees of freedom has been developed for the seismic analysis using the FEM based ANSYS program package. The model involves the complex of all interacting structures of the selected main production block of the plant. Components of mutual constraints of analyzed structures and exposed structural components have been modeled in detail, in order to get directly the loads for eventual redesign. Subsoil properties influencing the seismic response have been respected by selecting seismic motion inputs. The energy dissipation in structures has been modeled assuming modal damping ratio of 5%.

Two years ago a detailed seismic response analysis of NPP Dukovany structures has been performed ([1]) using response spectrum method. Seismic motion inputs have been defined in accordance with the IAEA recommendations for seismic re-evaluation of operating nuclear power plants with VVER-type reactors. Responses have been computed using all significant modes out of the 3500 modes of vibration that have been computed. Resultant seismic response has been computed using standard procedures. This response has been combined with the response to operating steady loads using a simple superposition rule. The combined response has been considered for assessing the earthquake resistance of structures using the HCLPF approach. The earthquake resistance of structures has been specified as the least of HCLPF values computed for all main structural components. The analysis has
revealed high stresses in a number of structural members. Although ductility properties he load limits, structural modifications have been present a satisfactory reserve with respect to the load limits, structural modifications have been recommended. Consequently feasible structural modifications, which enhance the earthquake resistance of the analyzed structures, have been designed [1].

Recently, seismic response analysis of the NPP Dukovany building complex has been accomplished using the time history method. The analysis has been carried out in order to verify the conclusions of the seismic response spectra analysis prior to the final design of structural modifications mentioned above. Seismic excitation of the computation model has been defined by synthesized site-specific design acceleration time histories. The envelopes of computed seismic response time curves have been determined. The selected response extremes have been combined with the responses to operating steady loads. The combined response extremes have been compared with the extremes obtained by response spectrum method.

Some important conclusions have been drawn from the seismic analyses. The correctness of results of the seismic analysis of NPP structures depends principally on the correctness of modeling the physical process going on in the course of the earthquake event. The modeling of the process involves the modeling of the structure and the modeling of the seismic motion of the base of the structure.

It has been concluded, that the modeling of NPP structural complex as a whole represents the fundamental condition to get reliable results. The response displacement fields have shown the practical impossibility to formulate the boundary conditions for a correct solution of separated structures. It has been proved that, due to very complicated vibration mode shapes, reliable results in stresses can be obtained only when even minute details of the structure are modeled. It has been shown, that simplified modeling of mutual constraints of structural parts commonly used in standard analyses cannot be applied in the seismic analysis. Special constraining elements modeling unambiguously the load transfer have to be used.

The seismic response of the NPP Dukovany complex of structures has been first computed using the response spectrum method. The obtained fields of extreme values of both response displacements and response stresses have been applied for the design of structural modifications enhancing the seismic resistance of structures. However, when deducing detailed conclusions from the response spectrum analysis results, some supplementary questions have arisen. It has been found as reasonable to perform another independent seismic analysis in order to verify the response spectrum analysis results.

The seismic analysis using acceleration time histories has proved its importance. Compared to the seismic response spectra analysis, the new analysis has shown a more realistic view on the character of seismic loads. Although the differences of mean response levels for structural parts have been mostly unsubstantial, the newly computed local responses have been in many cases considerably higher. Consequently, the new data obtained by the seismic analysis using acceleration time histories have been respected in the final design of the proposed structural modifications.
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REGULATORY CONTROL OF AGEING RESEARCH REACTOR IN INDONESIA

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According to the requirements of BAPETEN, safety review (SR) to the report on ageing assessment should be carried out for the license renewal applications of research reactors that have been operated more than 25 years. Ageing management is one of the important safety factors to be reviewed for license renewal applications. In order to assess continued operation of a research reactor from the reactor safety standpoint, a methodical approach should be taken. Such an approach will utilize data from the ageing management program and should incorporate the following considerations: A safety review of the reactor tailored to establish the actual status of the systems regarding degradation from ageing or other specific mechanisms; An overview of the potential refurbishment needs, by establishing a comprehensive list of systems and components, categorizing and prioritizing them; A selection of the critical items and identification of the relevant ageing mechanisms in order to perform a preliminary evaluation of the critical items; The establishment of the technical feasibility of the refurbishment program; and The identification of further studies and inspections to refine the preliminary assessment. This paper described the regulatory control of ageing research reactor in Indonesia, including the rules and regulation, the licensing process and the periodic review of ageing management.
ANALYSIS AND EVALUATION ON HYDROGEN BLISTERING OF TITANIUM TUBES ON TUBE SHEET OF RCW HEAT EXCHANGER

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Conductive titanium tubes of RCW (recirculation cooling water) heat exchanger with material specification of ASME SB338 grade 2 failed after operation of only two and a half years in Third Qinshan Nuclear Power Co., Ltd. Fig.1 shows morphology of one of the failure titanium tubes. The heat exchanger was of the shell-and- tube type with seawater inside the tubes and desalted water at the shell side, which has allowable design life of 40 years. The service temperature of the media was about 30–42°C. Both sides of each tube were fixed by 78mm-thickness titanium-steel composite plate with material ASME SB265 grade 1 thin titanium clad at the tube side and material ASME SA515 grade 65 carbon-steel at the shell side. The hydraulic expanding technique without seal weld was used to connect tube sheet and nozzle with expansion joint ratio of 80%.

The failed heat exchanger tubes were subjected to chemical composition analysis (inductively coupled plasma atomic emission spectrometry, C/S gas analyzer and N/O/H chromatography meter), metallography analysis (scanning electron microscope), microstructure examination (transmission electron microscope) and a series of mechanical property test including tensile test, hardness test, flaring test and flattening test. The testing results show that the material of the heat exchanger tubes in service is conformity with the relevant materials standards.

The cause of the failure was thoroughly investigated using stereo/optical/scanning electron microscope equipped with energy dispersive spectrometer, X-ray Photoelectron Spectroscopy, (XPS), secondary ion mass spectroscopy (SIMS), X-ray Diffraction (XRD), Raman spectrum and attenuated total reflectance – Fourier transform infrared spectroscopy (ATR-FTIR).

The study revealed that there are galvanic corrosion and crevice corrosion between titanium tubes and titanium-steel composite plate due to local gap at the expansion-joint region connecting tube sheet and nozzle. The product of electrode reaction induce local hydrogen blistering at some small areas of gap. Furthermore, the brittle titanium hydride which was resulted from the hydrogen absorption reaction between the surface of titanium and atomic hydrogen through oxide film, has been certified by XRD, XPS and SIMS (as shown in Fig.2). The existence of the face centered cubic crystal system titanium hydride made the hexagonal closed-packed crystal system α-Ti matrix lattice distortion and grain-boundary embrittlement and easy to cracking under tensile stress. Both of the hydrogen blistering and the product of hydrogen absorption reaction made the tube wall inward oriented bubble, which induced the eddy erosive wear of the tube by seawater containing sand. The interaction effects of galvanic corrosion, crevice corrosion and the erosive wear finally lead to break of the tube and thus enlarge the crevasse.
Based on those analysis, we provide the following recommendations for the titanium tubes of RCW heat exchanger: seal weld or high performance polymer adhesive agent should be used between tube sheet and nozzle in order to eliminate galvanic corrosion.

The study firstly approved the hydrogen absorption reaction of titanium and its product TiH$_{1.924}$ at about 40°C, which provide a new suggestion and direction for analyzing the failure reason of titanium tubes. Also, a useful proposal for ensuring titanium tube for long-term safety service is put forward.

(a) outer surface  
(b) inner surface
FIG. 1. Morphology of the failure titanium tube.

FIG. 2. The concentration and distribution of outer surface elements along the depth of titanium tube (SIMS).
PRELIMINARY STUDY ON STRATEGY OF PLANT LIFE MANAGEMENT FOR THE FIRST NUCLEAR POWER PLANT IN INDONESIA

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Indonesia plan to operate the first PWR type nuclear power plant in 2016. The strategy of plant life management for the first PWR should be studied well. The objective of the plan life management is to maintain reliability and safety for a long term operation as long as the plant nominal design of life. Besides, it is important also to preserve the option of extending plant by maintaining good safety and economic performance\cite{1}.

Scope of strategies which have already been studied including:
1. General methodology of plant life management.
2. Defining technical elements of plant life management.
3. Data and record keeping activities.
4. Relationship between maintenance and plant life management to achieve the long term operation goals.
5. Research and development requirements of plant life management for long term operation.

1. General methodology of plant life management:

A general methodology of PLiM has been studied based on the reference, such as screening and ranking to identify the majority critical system, structures and components. Ranking of the majority critical components are based on the safety or reliability aspect which indicate the importance of key components related with the major impact on safety or plant operational life and cost. As an initial work screening and ranking have already been set up for three BATAN research reactors, one of them is TRIGA MARK II Kartini Reactor in Yogyakarta, given in Table 1.

Table 1. Screening and Prioritization of Kartini Reactor Components\cite{2}

<table>
<thead>
<tr>
<th>Rank</th>
<th>Major Kartini Reactor Components</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.</td>
<td>Reactor tank</td>
</tr>
<tr>
<td>2.</td>
<td>Reactor internal</td>
</tr>
<tr>
<td>3.</td>
<td>Heat exchanger</td>
</tr>
<tr>
<td>4.</td>
<td>Primary cooling system piping</td>
</tr>
<tr>
<td>5.</td>
<td>Secondary cooling system piping</td>
</tr>
</tbody>
</table>
2. Define technical elements of plant life management:

Technical elements of plant life management has already been defined, those are: ageing characterization for critical components, maintenance strategy, inspection and surveillance, repair and refurbishment, and verification/compliance with licensing basis.

3. Data and record keeping activities:

Plant life management requires data and good record keeping. A good documentation allows operating organization to follow the progress of plant ageing. The data/information which is necessary for detecting plant degradation such as : base data of component specification from fabricator (manufacturing material, initial properties, codes and standard, etc), age or failure tracking data (operational history, in-service inspection, in-service monitoring, etc.), stressor and root cause data, failure or degradation mechanism data, test and maintenance data. BATAN has a good documentation system for its research reactors. In-service inspection of reactor tank, reactor tank internal, heat exchanger tube and secondary cooling piping have been conducted regularly. Assessment and analysis of inspection results including root cause analysis are conducted when abnormal condition observed.

4. Relationship between maintenance and plant life management:

From plant life management point of view it is essential to manage and assess the maintenance program on the system, structure and components, because maintenance program is part of plant ageing evaluation and plant life management. Plant life management for achieving long term operation goals are depend on good maintenance practices. A routine maintenance especially for component important to safety is conducted as well for the three BATAN research reactors.

5. Research and development requirements of plant life management for long term operation:

Research and development to assess the reactor pressure vessel integrity and to support plant life management is very important. The objective of R&D programs is to characterize through R&D all plausible degradation mechanism of critical components and to develop inspection, monitoring and mitigation techniques to manage ageing[1,3]. Preliminary research work on ageing degradation and structure integrity of reactor pressure vessel and its similar material has already been started by BATAN, especially in the Centre for Reactor Technology and Nuclear Safety. Result of some research works such as effect of irradiation temperature in ductile to brittle transition temperature shift and hardening and it also effect of Phosphor, Copper and Nickel Doped on Mechanical Property of Ferritic Alloy have already been known[4,5]. Other research topics related to the effect of thermal and fatigue ageing of reactor pressure vessel materials will be conducted for the future.
A simple proposed model of plant life management for critical components is given in Figure 1.

\[ \text{FIG. 1. Proposed model of plant life management for critical components} \]

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ACTIVITY IN KAZAKHSTAN RELATED TO NPP LIFE MANAGEMENT

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It is well known, neutron damage to the permanent internal structures of light-water (LWR) reactors is one of the limiting factors in the usable life of these reactors. Although neutron damage to LWR components can be simulated by various ion bombardment techniques, there is always some modeling extrapolation necessary, so that there is no substitute for direct measurement on irradiated specimens. Unfortunately, there are very few reactor facilities world wide for specimen irradiation and, when available, such irradiations take about the same time to achieve end-of-life neutron doses as in actual LWRs. Similarly, the alternative method of obtaining samples from the permanent internal structures of decommissioned plants is extremely difficult and expensive. In consequence, the case for life extension of pressurized water reactors (PWRs) and the Russian equivalents (VVERs) is presently made on the basis of extrapolating trends observed in lower dose materials.

At present time there are no operating nuclear power plants in Kazakhstan, the only BN-350 fast breeder reactor is under decommissioning since 1999. In the framework of ISTC K-437 Project it was suggested to investigate the materials irradiated in BN-350 reactor keeping in mind the low inlet temperature of BN-350 (280\textdegree{}C) which means that the irradiation conditions in-reactor (dose rate, dose and temperature) bound the values occurring in LWRs and to use these data for predicting behavior of LWR internals after its operation within long period of time.

When implementing this project the main attention was paid to investigations of the changes in structure and mechanical properties of the steels used as the ducts of BN-350 reactor Fuel Assemblies (FA) depending on neutron flux / fluence and irradiation temperature. Project objective was an obtaining data on BN-350 reactor materials structure and properties changes for extension of the lifetime of Light Water reactors of commercial nuclear power plants.

Specimens of 08Cr16Ni11Mo3 (AISI 316 analogue) and 12Cr18Ni10Ti (AISI 321 analogue) austenitic stainless steels cut off from the different places of chosen fuel assembly ducts irradiated in BN-350 reactor have been investigated by means of optical metallography, TEM and SEM, X-ray analysis, mechanical tensile tests, microhardness measurements, density measurements, thermodesorption spectroscopy. ISTC Contact Expert Group on Plant Life Management (ISTC CEG PLIM) has supervised this activity. Preliminary results of this research were published in [1-3]. The most interesting results obtained, which relate to the life extension of light water reactors are as following:
The voids were found in both 08Cr16Ni11Mo3 and 12Cr18Ni10Ti steels irradiated at rather low temperature 280°C up to only 1.3 dpa damage dose at dose rate $3.9 \times 10^{-9}$ dpa/s and 0.65 dpa at $1.2 \times 10^{-9}$ dpa/s respectively. Previously, there was a common opinion that 305°C was a threshold temperature for pore formation in austenitic stainless steels and these steels swelled only at higher temperatures. The data obtained show that pores could be found in the internals of light water reactors operated at similar irradiation conditions.

It was found that voids with the diameter 10-15 nm and density $1 \div 2 \times 10^{21}$ void/m$^3$ are nucleated in 08Cr16Ni11Mo3 austenitic stainless steel irradiated at $T=302 \div 311^\circ C$ up to damage doses 7÷13 dpa with dose rates $2.2 \div 3.3 \times 10^{-8}$ dpa/s, while voids were not found in the austenitic steel samples irradiated within the same temperature range up to the same damage dose but with dose rates $19 \div 26 \times 10^{-8}$ dpa/s. 12Cr18Ni10Ti austenitic stainless steel demonstrates similar behavior.

Simple model has been developed to explain the dose rate dependence of swelling in austenitic stainless steels.

There were obtained dose rate dependencies of yield strength, ultimate strength and ductility for 08Cr16Ni11Mo3 steel irradiated up to 7-13 dpa at 302÷311°C. These dependencies show a decrease in both yield strength and ultimate strength when dose rate decreases. This effect was not found for 12Cr18Ni10Ti steel samples. But for both steels, there was found an apparent decrease in total elongation at low dose rates. This is an important issue for light water reactors internals because, possibly, there was no previously an expectation to find a significant lost of ductility in light water reactors internal components irradiated at relatively low temperatures up to low doses and at low dose rates.

There was found unexpectedly high value of embrittlement of 08Cr16Ni11Mo3 steel irradiated up to damage dose 11 dpa only at dose rate $2 \times 10^{-7}$ dpa/s and irradiation temperature 346°C. Presumably, steel properties degradation was caused by a creep effect under mechanism of grain boundary sliding.

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GRADE MANAGEMENT OF PWR STEAM GENERATOR POTENTIAL AGEING MECHANISM BASED ON RISK

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PWR steam generator (SG) has two primary functions: one is to provide steam flow with required flowrate and moisture, the other is to guarantee structure integrity of the pressure boundary. With increase of operation time, SG degradations caused by aging would occurs, which may serverly affect the achievement of the two SG functions. Varied construction, design, materials and environment factors may result in the degradation mechanisms. Ageing mechanisms analysis is fundamental and necessary step for SG ageing management. In this paper, the potential ageing mechanisms of TQNPC CANDU 6 SG are showed in Fig 1.

Generally, there is a formula that the risk= occurrence frequency \times consequence. Here, the risk management concept is introuced into the SG ageing management, and the potential ageing mechanisms are graded into three levels of level I, II, and III based on the risk. The occurrence frequency is determined by the lab qualification test data of materials, design analysis results of components and the interior and external operation experience feedbacks of NPPs etc.

For different level ageing mechanism, the management requirements in the course of PDCA cycle will be varied. For level I ageing mechanism of SG, most strict inspection and maintenance requirements must be used to keep the degradation caused by level one mechanism under control. Continuous operation optimized and periodic state evaluation are also necessary. For level II and III, the inspection and maintenance requirements can be lowered, grading management of SG ageing mechanism should be dynamically operated, the level of the existing ageing mechanism will be upgraded or downgraded on interior and external experience feedbacks of NPPs during the periodic ageing management evaluation.

In this paper, the principle and method to grade SG ageing mechanisms are introduced, and the varied management requirements for different level mechanism are described as well. The idea has been used for SG management of chinese NPP of QNPC and TQNPC. Moreover, the grading management of ageing mechanism can be extended to other component ageing management and NPP ageing management.
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GENERAL METHOD OF THERMAL STRESS CALCULATION ON PRESSURED PARTS IN NUCLEAR POWER PLANT

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All the pressured parts in the nuclear power plant suffer from the alternative stress during the startup, shutdown, and load following process. Failure occurs at stress much below the ultimate tensile strength of the part material, due to accumulation of damage. The alternative stress is composed of thermal and pressure stress, which are related with temperature differentiate and pressure in pressured parts. The paper presents a general method to calculate thermal stress on pressured parts.

In order to obtain thermal stress, temperature field should be calculated first. Generally the temperature differential equation is listed as follow.

\[ \theta(r, \tau) = t(r, \tau) - t_0 \]
\[ \frac{\partial^2 \theta(r, \tau)}{\partial r^2} + \frac{1}{r} \frac{\partial \theta(r, \tau)}{\partial r} = \frac{1}{a} \frac{\partial \theta(r, \tau)}{\partial \tau} \]
\[ \theta(r, 0) = 0 \]
\[ \theta(R_1, \tau) = f(\tau) \]
\[ \frac{\partial \theta(r, \tau)}{\partial r} \bigg|_{r=R_i} = 0 \]

Through Laplace transfer, the temperature fields are list.

\[ \theta(r, s) = A \cdot I_0(\omega r) + B K_0(\omega r) \]

\[ A = \frac{\phi(s) \cdot K_1(R_i \omega)}{I_1(R_i \omega) K_0(R_i \omega) + I_0(R_i \omega) K_1(R_i \omega)} \]
\[ B = \frac{\phi(s) \cdot I_1(R_i \omega)}{I_1(R_i \omega) K_0(R_i \omega) + I_0(R_i \omega) K_1(R_i \omega)} \]

\[ I_0(\omega r) \text{ is Bessel function, } K_0(\omega r) \text{ is Bessel function, } A \text{ and } B \text{ is the function of } S. \]
\[ I_0(\omega r) \text{ is Bessel function, } K_0(\omega r) \text{ is Bessel function.} \]

The thermal stress can be listed below.

\[ \frac{\phi(R_i, s)}{s \phi(s)} = \frac{\alpha_i \cdot E}{1-\nu} \cdot \frac{2 R_i D - (R_i^2 - R_1^2) \omega C}{s (R_i^2 - R_1^2) \omega C} \]
\[ \frac{\phi(R_2, s)}{s \phi(s)} = \frac{\alpha_i \cdot E}{1-\nu} \cdot \frac{2 R_2 D - (R_2^2 - R_1^2) \omega C}{s R_2 (R_2^2 - R_1^2) \omega C} \]
\[ C = I_1(R_2 \omega) K_0(R_1 \omega) + I_0(R_1 \omega) K_1(R_2 \omega) \]
\[ D = I_1(R_2 \omega) K_0(R_1 \omega) - I_1(R_1 \omega) K_1(R_2 \omega) \]
Although the stress influence function is got, the thermal stress can’t be put in practice. The function is a super function. There are infinite zeros and polar points. It needs to simplify the result. Polar method can be used to simplify the influence function into lower rank function. The result can be expressed below.

\[
\frac{\phi(r,s)}{s\phi(s)} = M\left[\frac{p_1}{s+s_1} + \frac{p_2}{s+s_2} + \ldots\right] = \frac{A}{Bs+1}
\]

In practical online monitoring system, the working stress can be calculated from the configuration below. Operators can inspect the actual stress during operation.

**FIG 1 Configuration of the working stress calculation**

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MAINTENANCE AND LIFE ASSESSMENT OF STEAM GENERATORS AT EMBALSE NUCLEAR STATION

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The Embalse Nuclear Generating Station (ENGS) has four vertical I-800 U-tubes Steam Generators (SGs) manufactured by Babcock & Wilcox (B&W). They are one of the most important components from the point of view of safety and cost-related elements for potential life extensions in case of a replacement thereof. A Plant Life Management program has commenced covering the entire plant and one of the first pilot studies was the SG Life Assessment (LA), which consisted in a systematic way to evaluate the aging mechanisms focused on the plant refurbishment and life extension.

Because of this, maintenance-based ageing assessment from beginning of operation has been analyzed and LA-frame maintenance and inspections programs were carried out. The most important taken actions have been the Eddy Current (EC) In Service Inspection Program (ISI) which performs 100% of the tubes of two SG every 1.5 years started in 1992, the mechanical cleaning by blasting of the internal tube surface, the sludge removal from the secondary side tubesheet, the divider plate replacement, the installation of antivibration bars (AVB’s), installation of TSP inspection ports and an exhaustive inspection of the secondary internals as a preliminary result of the LA.

The most relevant aging mechanism up to 2004 was the Flow Accelerated Corrosion (FAC) of U-bend supports and consequent fretting of tubes. The eddy current inspections allowed the fretting degradation to be detected and mitigated by installing AVB’s. Currently, efficiency of this mitigating action is followed by vibration measurements and visual inspections. However, other degradation mechanism that could have origin due to the U-bend FAC like loose part damage (LPD) it is being analyzed since could be an issue in the near future. At present, FAC degradation on the cold leg side and sludge deposition on the hot leg side of the carbon steel Tube Support Plates (TSP) are the main ageing issues at the point that SGs life extension is almost discard, and even the design life was compromised. During the LA-frame secondary side inspections of oct-nov 2005, many internals of SGs 2 and 4 were inspected for the first time and some important and minor degradation issues that are being assessed were found: possible crevice corrosion at the thermal plate-tube gap; incipient FAC at the steam separators and loose part/foreign object detection.

The results of the 2005 inspections and the conclusions of a set of studies performed by AECL (the plant designer) in relation to the TSP issue, which include structural integrity analysis and feed water chemical improvement to reduce FAC rate, has allowed to complete the Life Assessment and the recommendations needed for the refurbishment (repair/replacement of steam generators, modifications of the In Service Inspection and Maintenance programs for the future, etc)
The Life Assessment systematic approach to evaluate ageing in the steam generators has permitted to obtain important results related to the SG design life attainment and life extension. TSP structural analysis shows that the design life will be reached, however, the TSP FAC issue resulted to be the SG life limiting and last decisions have been oriented to the replacement options during the refurbishment outage in 2011.
FRAMEWORK FOR MANAGING AGING EFFECTS OF ENVIRONMENTALLY ASSISTED CRACKING OF NICKEL BASED ALLOYS IN PWR ENVIRONMENT

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As the nuclear power plants get older, aging becomes an issue, because aging degradation can affect the structural integrity of systems and components. With the advent of risk-informa regulation, the assessment of risk by aging becomes an important issue as new power plant installations are delayed and concern with global warming increases. The license renewal of Nuclear Reactors has been recognized as a key solution to this issue. In U.S., such applications, if approved, will extend nuclear plant operation for an additional 20 years beyond the current 40-year limit. To ensure the safe operation of nuclear power plants in the United States, it is essential to assess the effects of age-related degradation of plant structures, systems, and components. This concern becomes even more important in the view of the submittals of license renewal applications to the NRC under 10 CFR Part 54 License Renewal Regulation that focuses on aging effects associated with long-lived passive structures and/or components through time-limited aging analyses. [NRC 1995] [Braverman, et al. 2004]

This paper will provide a framework for probabilistic assessment of aging passive components, with particular attention to Environmentally Assisted Cracking (EAC) of pipings and nozzles on Nuclear Power Plants (NPP). The framework for EAC aging management should be founded on the fundamental and the mechanism-based EAC study and the established database.

EAC of piping and nozzles have occurs mainly in weld regions including heat affected zones when material, environment, and stress factors are superimposed in unfavorable manners. In the aging-related assessment, it is necessary to select components that have non-negligible potential for EAC occurrence and to perform an integrity assessment taking into account of three factors mentioned above. In operating plants, a number of EAC phenomena of different mechanism have been observed so far. From the analysis of such cases, three kinds of EAC have been realized as very important phenomena for the age-related technical assessment.[JNES 2005]

They are:

(i) Intergranular Stress Corrosion Cracking (IGSCC) of low carbon stainless steels;
(ii) Primary Water Stress Corrosion Cracking (PWSCC) occurrence in Ni base alloy; and
(iii) Irradiation Assisted Stress Corrosion Cracking (IASCC) induced in core internals

PWSCC is one of the main degradation processes of passive components in PWR’s. But, several mechanisms have been proposed to account for the PWSCC behavior of nickel-based alloys in the high temperature water (~ 300°C). The mechanism that controls PWSCC in nickel-based alloys has not been clearly identified and only a few quantitative models,
largely based on databases, have been developed for the prediction of crack growth behaviors. [Rebak, et al. 1996, Kwon, et al. 2005]

Nevertheless, a probabilistic assessment methodology based on degradation mechanism of passive components including Ni base alloy materials is needed for quantifying potential risk of aging. For the first level modeling of this process, the EAC-induced failure process has been described by crack initiation phase, stable crack growth phase and the final unstable fracture phase. The overall failure probability of Ni based alloy piping has been determined by taking into account the inspection uncertainties. Probabilistic Fracture Mechanics (PFM) analysis for the assessment of final fracture also depends on parameters with uncertainties. In this paper we have examined the effect of uncertainties in key parameters in models for EAC, fracture and inspection, based on a sensitivity study.

REFERENCES

Regulatory guidelines for the continued operation of CANDU reactor in Korea were introduced in this paper. Wolsong unit 1, which is a CANDU 600 reactor in Korea, is approaching to its design life of 30 years in 2011. The licensee who wants to operate a nuclear power plant beyond its design life should submit the periodic safety review (PSR) report conducted based on the eleven safety factors required by the Enforcement Decree of the Atomic Energy Act Article 42-3 (Contents of Periodic Safety Review) clause 1. In addition, two more factors should included in the application in compliance with clause 2 of the same article: the first is the life evaluation of systems, structures, and components (SSC) considering the period of continued operation, and the second is the assessment of radiological environmental impacts considering changes since the start of operation. As per the Notice of the MOST for CO (No. 2005-31) Article 3 subparagraph 1, the licensee should provide the items as follows: (1) Scoping and screening results of aging management, (2) Aging management program, (3) TLAA including the continued operation term, and (4) Operation experience feedback and important safety research results. In this study, the 53 regulatory guidelines for the above 4 items for the CANDU reactor in Korea were developed.

The guideline provides the method for determining the components, systems, and structures (SSC) within the scope of the continued operation application. The lists of SSCs subject to aging management and aging management program for each item shall be presented. In principle, the scoping process should be focused on the selection of the SSCs that are screened as “passive” and “long-lived”.

The guideline for the aging management program (AMP) contain ten elements: the scope of program, preventive actions, parameters monitored/inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls and operating experiences. In addition to these elements, special considerations is required for operating experiences and safety research results as requested in the safety review guidelines. The licensee should provide the information about the items to be reviewed for continued operation, measures to be taken prior to continued operation, additional measures to be taken posterior to the start of continued operation in consideration of characteristics of design and operation, and operating experiences in AMP.

The time-limited aging analysis (TLAA) is to address plant specific safety analyses that are based on plant design life. The licensee is required to provide a list of TLAAAs as defined in the Notice of the MOST No.2005-31. The seven items subject to TLAA are provided in table 3 of the above notice. The licensee shall demonstrate one of the followings: (1) The analyses remains valid for the period of continued operation, (2) The analyses have been projected to the end of the continued operation, and (3) The effects of aging on the intended functions will be adequately managed for the period of continued operation.
The operational experience feedbacks are those items that have not been covered during the safety review of initial operating license but that have been selected to be important from operational experiences and safety research results, whether they are based on our own experiences or other countries’ ones. Five items are provided in the guidelines such as fire protection, seismic qualification of equipments, management program for active components, thermal stratification in piping, and concentration of combustible gas.
ROLE OF ORGANIZATIONAL LEADERSHIP IN PLANT LIFE MANAGEMENT

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The nuclear power plant (NPP) operational trend shows that the plants of the same design and brought to service about the same time demonstrate a wide range of life time operational performance. Based on years of performance assessment experience from various types of industry audits, it can be seen that there is a strong relationship between organizational leadership and the good performing plants. A review based on this relationship is provided to suggest important characteristics needed in management and leadership team for an organization to have a successful life management program in a NPP.

The required characteristics and attributes are discussed in the following three important organizational elements: Environment, People and Process

ENVIRONMENT

The senior executive team, starting from the Board of Directors or equivalent body needs to create an environment in the organization which is seen by the management team and the staff at all levels as a long term commitment to the safe operation of the NPPs. A policy to support this vision needs to published and communicated.

The strategic planning part of the organization needs to look into the short, medium and long term impact of the potential forces, internal and external, which can impact this vision and the organization Business Plan should include mitigating measures. The next level of detailed planning should include the capital and cash outlay to match the resources needed to sustain a defined life management plan.

There needs to be ongoing communication of the plans in place to preserve the assets of the NPPs, which will support a culture for sustaining long term safe operation and will keep the staff engaged.

PEOPLE

People and their knowledge are one of the important resource for NPP operation and life management. This becomes even more important in specific application to life cycle management as knowledge and expertise play a major role in capturing the information acquired over long period of time. A formal knowledge retention program and succession planning play an important role in this area.
The training of new employees and ongoing training need to include aspects of long term planning, asset preservation and life management. This will encourage staff to make appropriate decisions.

PROCESS

As we know, suitable processes play an important role in managing safe operation of NPPs. However to ensure the long term operational focus is not overlooked due to the overriding short term demands, processes need to be carefully put in place for managing and directing the life cycle management. The structure has to be clear where the decisions are made for the short and long term needs and respective roles and responsibilities must be clearly defined.

The strategic planning and short term operational planning need to be linked but carefully managed in parallel. Data and technical facts on equipment performance, technology, aging mechanisms etc. need to be retained in properly aligned groups in the organization.

A model is proposed by which day to day operational and outage programs and activities can be linked to gathering information leading to a living life management program. A structure for arranging the policy, programs and procedures etc is explained. This is designed to help in making the programs sustainable in an efficient way which will lead to the development of a suitable living life management plan.
PNRA EXPERIENCE WITH LICENSING OF KANUPP BEYOND DESIGN LIFE

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Abstract. Global surge in energy demand and the post-Kyoto environment conscious scenario is pushing the international community to re-evaluate its utilization of different energy resources. Under the circumstances, nuclear option is coming forward as a sensible choice for increased share in generation mix. PLiM/PLEX adds to the sensibility of this choice. Approaches followed for this must be discussed mutually which would increase both harmonization and rationalization.

Life extension of present plants would also make a smooth transition from plants installed in 60’s to 80’s to projects envisaged for future energy generation. This would also preserve the know-how and experience gained over the years.

Pakistan’s experience with the life extension of Kanupp, a 137 MWe CANDU and its licensing could give important insight to the issues needing to be addressed in plant’s life management. The paper describes, firstly, the issues faced by K1 and modifications/upgradations made to counter them and secondly, the methodology followed by Pakistan Nuclear Regulatory Authority in its relicensing.

The last part of the paper puts forward some lessons learned through the process and how they are being used for Pakistan’s other nuclear power plant, C1.

Synopsis
The regulatory body gives prime importance to the safety margin available in nuclear power plants while making regulatory decisions. This margin fluctuates with the life of a plant as a result of continuous changes in the condition of plant equipment (in term of aging, modifications etc.). The goal of regulatory framework is to keep this margin within an acceptable band. As the plant ages, this margin would narrow and unless the utility aims for a margin greater than minimally acceptable, this margin would diminish.

Life extension of plants (PLeX) made to earlier standards makes the task of keeping a ‘verifiable’ acceptable safety margin more difficult. Moreover, as a number of modifications are made during plant life extension, maintaining the design integrity of the plant adds in as a challenge. There are other factors which must also be accounted for such as change in regulatory framework, the economic justification of the job, technology involved in modifications, compatibility of older systems with new modifications, etc.

Pakistan’s experience with life extension of KANUPP and its licensing beyond its design life has been a challenging task. KANUPP is a first generation 137MWe CANDU which completed its design life of 30 years in October 2002.
PNRA carried out safety assessment of KANUPP and issued a report in September 2001, realizing its responsibility and commitment for the safety of general public, operator and environment. The objective was to identify safety significant weaknesses and inform operators well in time to initiate suitable corrective actions for operation of plant beyond its design life if so desired. The report identified 24 issues of safety concerns – 17 in category 1, while 7 in category 2. Category-1 issues were those which were considered of being highest safety concerns and needs immediate action to resolve the issue. Category 2 issues were those which reflected departure from CANDU and/or International practices. A prioritization of action based on safety significance was to be undertaken by KANUPP to resolve these issues.

PAEC submitted a formal request to PNRA for licensing beyond the design life of KANUPP in October 2001. In November 2001, KANUPP was asked to submit a work plan providing commitment to undertake the safety jobs with target completion dates. Successful accomplishments of these jobs along with others required under Regulation PAK/909 were mentioned as a pre-requisite for licensing of the plant beyond its design life of 30 years.

Accordingly, KANUPP developed an action plan to complete the jobs in two phases. In phase – 1, all the critical safety issues were chosen for execution. It was agreed that on satisfactory completion of Phase-1 jobs, as per PNRA’s satisfaction, conditional license for plant operation would be issued to KANUPP. After completion of Phase-2 jobs PNRA may consider “issue of licence for longer duration” depending upon the results of safety assessments to be performed before issue of such license. For licensing beyond the design life, KANUPP has implemented a number of modifications to improve the safety of the plant in the light of results of PNRA assessment. Accordingly, KANUPP performed containment integrity and leak rate tests successfully, conducted a comprehensive sector and site emergency drill, revised the OP&P, re-qualified all plant operating personnel, imposed new restriction on maximum excess reactivity in the core, revised the reporting requirements and relevant station instruction, developed accident management procedures for LOCA and LOECI event and for feeder pipe break, updated and upgraded plant documentation to reflect the “as modified and upgraded” system.

Upon satisfactory completion of the phase I tasks, plant was allowed to restart and operate at a lower power for an interim period to carry out commissioning and testing of the upgraded equipment and system such as performance of I&C system, control loops, regulating computers, to make necessary arrangements for procurement of equipments required to complete the remaining tasks etc. The plant was synchronized with national grid in the third week of January 2004 and the load was restricted to a maximum of 33% thermal power. The power was later allowed to be raised to about 45% FP based on safety demonstrating by the plant. Plant operated at a power of 45% F.P until December 12, 2005 when it was shut down to carry out remaining safety up-gradation jobs. Major jobs that were planned have almost been completed successfully during the 2006 shut down. These included installation & commissioning of Forced Injection Water System (FI JW), installation & commissioning of Redundant Injection Water System (RI JW), replacement of three MH Pump motors with environmentally qualified motors, ISI of Reactor Fuel Channels, inspection of Channel Feeders, replacement of two Active Drainage Pump Motors with environmentally qualified motors, ISI of Steam Generator for tube integrity, seismic retrofits, containment building pressure leak test at 11 psig, replacement of 113 control & instrumentation loops.
In compliance with the requirements as per PAK/909, KANUPP has submitted an updated Periodic Safety Review (PSR), Revised Final Safety Analysis Report (FSAR), Probabilistic Safety Analysis (PSA) level-1 plus based on actual plant design and operation data, Decommissioning Program and an Environmental Impact Report. Review of KANUPP FSAR is in progress and it may be completed in September, 2007. Presently, KANUPP has implemented nearly all the modifications required for licensing beyond the design life and is allowed to operate at 90 MWe under interim license on the basis of actions taken so far by KANUPP. However, formal licensing decision will be made after completion of revised FSAR review. KANUPP’s relicensing has set many benchmarks for how nuclear power is regulated in Pakistan. There are issues which need to be closely monitored if plant may apply for licensing beyond its design life. Spent fuel storage is one of the issues which can be highlighted as most plants have storage for waste generated up till the design life. Closely monitoring the fatigue cycles of vital components as a part of Aging Management Regime is another factor. Maintaining close liaison with Designer (Vendor), strengthening design support organizations, management of knowledge and experience of plant personal are all factors emphasized and enforced by PNRA due to its experience feedback from relicensing of KANUPP.

REFERENCES

[1] Regulation for Licensing of Nuclear Installation(s) in Pakistan
Almirante Álvaro Alberto Nuclear Power Station, referred herein as Angra NPP, comprises two generating units operated by Eletrobrás Termonuclear - Eletronuclear. Unit one, a Westinghouse designed PWR, operates commercially since 1985 generating 657 MW, while Unit two, designed by Siemens, is a 1350 MW PWR in operation since 2000. This paper focuses on Eletronuclear experience in developing and implementing ageing programmes addressing structural integrity of pressurized components.

Steam Generator programme, fatigue and flow accelerated corrosion monitoring integrate a set of ageing related programmes established to control ageing effects on equipment and piping components in order to ensure their capability to perform their safety functions during their remaining life.

Steam Generator Programme is part of Angra 1 alloy 600 management plan, established to detect tube cracking before it impacts plant safety and operability. Eletronuclear follows the Nuclear Energy Institute document NEI 97-06 Steam Generator Program Guidelines, which defines the performance criteria as defined by NEI 97-06. Condition monitoring evaluations and operational assessment are performed in sequence to the inspection of SG tubes, during outages, respectively, to verify their structural integrity after the last cycle and to predict their condition at the end of the next.

The SG operational assessment is the most important analysis to define the operational cycle period of Angra Unit 1. The procedure establishes an adequate safety margin to the maximum flaw sizes expected to occur at the end of the upcoming cycle.

Additionally to the normal procedure established in the SG programme, during the last outage (June, 2007), a Helium leak test was performed. Some tubes with through wall cracks were identified and the defects were confirmed by visual inspections. The helium tracer technique showed to be an effective tool to complement the Eddy Current inspections.

Fatigue Monitoring Programme, also necessary to future evaluations for plant life extension, is an important tool to compare the actual operational transients (amplitudes and number of occurrences) with the parameters used in the design of the plant as well to re-evaluate some loadings not considered in the original design, as thermal stratification loads.

The fatigue monitoring system was implemented in Unit 1 using the software FatiguePro, developed by Structural Integrity Associates (SIA) and Electric Power Research Institute (EPRI). The system FatiguePro uses information available in the Integrated Computational System - version 2 - (SICA2), like temperature, pressure and flow rate, to automatically compute operational cycles and perform fatigue assessment at selected locations. The operational transients obtained from direct measurements are, in general, significantly lower.
than those ones estimated by design which leads to a more realistic fatigue evaluation for plant components.

The automatic cycle counting has been recently introduced and is being performed in parallel to the manual current procedure.

The Thermal Stratification Monitoring System has been implemented in Unit 2, in the following systems:

- Residual Heat Removal System
- Main Feedwater System
- Emergency Feedwater System
- Reactor Cooling System (hot leg)
- Pressurizer Surge Line

The thermal stratification monitoring system detects and evaluates, for some selected locations, the occurrence of operational transients that produce thermal gradients at the piping transverse section. Based on the collected data, the fatigue damage caused by thermal stratification loading can be assessed.

Sets of thermocouples have been installed at piping sections potentially subject to thermal stratification and at those ones for which operational experience of thermal stratification had been reported. For the first type of location, a simple configuration with 2 thermocouples is used, measuring temperatures at the upper and at the lower points of the section. For the second type, a complete configuration, with 5 thermocouples is used. If indications of thermal stratification are detected by a simple thermocouple set, it may be replaced by a complete set.

Flow Accelerated Corrosion Programme provides guidance to control the piping thickness of the high-energy systems with potential flow accelerate corrosion degradation process. The FAC programme, covering Units 1 and 2 systems as indicated in Table 1, introduced the computational system COMSY, developed by Areva, to assist FAC management activities like:

- Identification of potentially susceptible locations
- Estimation of wear rates
- Recording of inspection results
- Analysis of wear rate evolution
- Lifetime prediction and inspection scheduling

<table>
<thead>
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<th>Table 1 - Systems covered by the Flow Accelerated Corrosion Programme</th>
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<tr>
<td><strong>Unit 1</strong></td>
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<tr>
<td>Main Steam (MS)</td>
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<tr>
<td>Feedwater (FW)</td>
</tr>
<tr>
<td>Extraction Steam (EX)</td>
</tr>
<tr>
<td>Heater Drain (HD)</td>
</tr>
<tr>
<td>Turbine Gland Steam and Drain (GS)</td>
</tr>
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</tbody>
</table>
Since the implementation of the computational system COMSY, in 3 outages of Unit 1, 690 piping elements have been inspected with 32 replacements performed. In Unit 2, after 2 outages, 350 piping elements have been inspected, resulting in the replacement of 2 of them.

REFERENCES


THE LARGE PROJECTS AT KOZLODUY NPP – WITH FOCUS ON LONG TIME OPERATION AND AGEING MANAGEMENT

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Kozloduy NPP p.l.c. is a single-owner joint-stock company with 100% of its shares held by the state. The Company was granted a license the State Energy Regulatory Committee for generation of electric power and heat.

The share of output of Kozloduy NPP in the total generated power in Bulgaria in 2006 was 42.2%.

All of us witness the rate at which the situation in the local and regional energy markets changes. The challenges faced by the energy industry increase as Bulgaria prepares to join the free electricity market in Europe. That joining implies compliance with the strict EU requirements for safe power generation meeting all environmental protection conditions and based on the energy efficiency principles.

To be sure, that process requires serious investments. However, we believe that it is the only alternative in the long term that can assure our success in the energy context of Bulgaria and Europe.

What does Kozloduy NPP invested into?

The largest-scale investment project at Kozloduy NPP is the Modernization Program that is being performed on Units 5 and 6.

Developed on the basis of the full scope of IAEA recommendations for VVER-1000 Units (Model B-320), the Modernization Program for Units 5&6 was organized as a set of 212 specific measures, distributed in groups according to their main purpose. The expected effect of their implementation is to achieve:

- Improvement of the safety of Units 5&6 through implementation of new design solutions;
- Validation of an adequate safety level by means of various analyses and additional studies in conformity with internationally adopted regulations;
- Safety upgrading through replacement of the equipment with expiring design life and of critical equipment.
- Improvement of work efficiency and operating conditions.
The contractors for implementation of priority measures were selected by international bidding in 1996. It was won by two bidders – Westinghouse of USA, and the European Consortium Kozloduy constituted by three leaders in the nuclear industry of the Old Continent – Framatome, Siemens and Atomenergoexport.

The investments required for implementation of that project are really impressive. The total resources envisaged for implementation of the Modernization Program amount to 491M€. Of them, about 135M€ is the planned equity that Kozloduy NPP will invest in the Modernization Program, and about 356 M€ was secured through credit agreements with various creditor organizations such as EURATOM, CITYBANK and ROSSEXIMBANK.

Performance of the Modernization Program was organized in two Phases:

Phase 1 – “Engineering”

Phase 2 – “Implementation”

Main areas of changes/improvement/ implementation are:

- Technological equipment upgrade.
- Control systems upgrade.
- Analyses.
- Documentation.
- Decommissioning.

Status of measures as per June, 2007:

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<th>Partner</th>
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<tr>
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<td>10</td>
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<td>212</td>
</tr>
</tbody>
</table>

More detail information for modernization of I&C systems of Units 5 & 6:

Operational time of I&C systems of Bulgarian NPP is about 30 years, though the lifetime of individual parts of I&C systems is limited by 10-15 years. I&C systems were designed in 60-70-th in accordance with the existing regulations and available technical solutions.

Modernization of I&C systems at operating NPP is under way in Kozloduy now.

Some examples (measures) from Modernization Program are presented below:

Installation of completely new systems, unforeseen in the original design of the unit:

- H2 monitoring and recombination system,
- System of protective measures for upgrading main steam and feed water lines against break;
- System for RPV (reactor pressure vessel) level measurement and control,
- Reactor pressure vessel cold overpressurization protection automatic system;
- Filtered ventilation system for beyond design-basis accidents ,
- System for continuous monitoring of 6 kV motor insulation status;
- Uninterruptible power supply system with greatly improved working parameters
- Replacement of Safety System 6 kV switchgear,
- New generator breaker for short circuit current switching,
- New system for radiation monitoring of greater preciseness,
- New “Ovation” computer information system (CIS),
- New digital unit process control system (UKTS),
- New automated turbine control system (ACYT),
- New technologies and equipment for facility status monitoring and preventive detection and elimination of defects are implemented.

The Modernization Program:
- ensures long-term effective operation of Units 5, 6 (35 and 39 fuel companies residual life time )due to their enhanced safety and reliability;
- catalyzes the erection of Belene NPP with the transfer of competence and experience;

Utilizing the potential of the best in the European and American nuclear energy, and proceeding from the latest requirements of the regulatory documents of the Chief Design Engineer and IAEA, we are sure that, upon completion of the Modernization Program, our power units will be definitely among the safest, most reliable ones in the world and that their service life can be extended by 15-20 years beyond the design time limits.
REFERENCES

PERSPECTIVES OF PLANT LIFE MANAGEMENT WITH TRANSBOUNDARY EFFECTS

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Perspectives of Plant Life Management (PLIM) with trans-boundary effects are determined by analyzing available technological, licensing, legal, economical and public documentation also with respect to the use for Long Term Operation (LTO). The outcome of compiling relevant documents and putting it into perspective is discussed.

Initiated by the OECD/NEA effort to provide for substantial information on Plant Life Management, the Austrian Government decided to establish a project to acquire current knowledge on issues related to this topic. Essential findings about the current status of safety and the envisaged service conditions of the various nuclear power plants are investigated and the declared positions as well of the entities responsible for the safe operation of the nuclear installations in the countries adjacent to Austria.

For this purpose the activities of the stakeholders in the Plant Life Management processes in these countries are analyzed, commented and incorporated into an information data base. At the same time confirmation is sought of the development of safety culture, safety features and operational safety for the introduction of Long Term Operation at several of these plants.

The interaction between the stakeholders on both sides Austria and the countries with nuclear installations, as mentioned, is an essential corrective and it is therefore enhanced substantially by the effort to acquire in depth understanding of PLiM and LTO including the related transition provisions.

Sources of information are described and the processing as well as the patterns of usage as a source of qualified technological information, but also in the policy discussion and the information to the public.

Information on nuclear installations around Austria and the detailed list of topics addressed within the scope of the compilation to be used by the experts are provided.
PWR-LTO AND AGEING: A STUDY OF AVAILABLE PUBLIC MATERIAL

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The Project with the title “PLiM LTO-2007” was assigned to the University of Vienna (IRR) by the Austrian Ministry of Environment with a 17 month implementation phase starting in summer 2006. The main goals of the study are:

- To assist in preparing the OECD-NEA document on “Impacts of Nuclear Power Plant Life Management and Long Term Operation”[1]
- To collect, overview and summarise openly available documents and material on NPP plant ageing to create a know-how base for safety evaluation of LTO measures for the Austrian Government.

The second task performed focussed on safety relevant ageing related topics especially for WWER-440/213, WWER-1000 and Westinghouse PWR 1000 plants. The ongoing plant life extension process at the NPP in Paks was monitored as an example of special interest to Austria. Information and material devoted to generic questions in LTO contexts are also included into the literature research and the evaluation process.

A set of default general key parameters was used as a Thesaurus to classify considered documents in a collection roster with the structure induced close to the IAEA systematic. A “literature sheet” was generated for each of the documents determined relevant, it contains on one or two pages an information summary about the contents.

Two paragraphs on these sheets concentrate a short abstract and an appraisal of the content. The literature sheets and the related documents are enlisted according to the content classification structure of five chapters or designated “off record”. The main chapters for sorting out are the following:

(1) Safety and Technical Considerations of PLiM  
(2) Technical Limitations of PLiM  
(3) Decision-making Process  
(4) Resource Management Issues  
(5) International Context

In a second level classification the documents were associated with a larger set of subtopics for a more detailed identification of the major topics treated. About 200 key documents have been identified, evaluated and introduced into the knowledge documents’ data base with literature sheets, no doubt that a many more documents had to be screened for relevance to safety related topics in association with NPP ageing and LTO.
The knowledge documents’ data base will soon be available for the use in the safety monitoring process and it is expected that an enhanced evaluation of future cases in connection with PLIM and LTO issues will be accomplished. Questions about the relevance of PLiM related events are expected to be answered more adequately.

ACKNOWLEDGEMENTS

• Federal Ministry of Agriculture, Forestry, Environment and Water Management

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FRACTURE MECHANICS-BASED LIFE MANAGEMENT OF STRUCTURAL MATERIALS OPERATING AT ELEVATED TEMPERATURES

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SYNOPSIS

All aspects of plant safety are paramount in the nuclear industry and therefore advanced methodologies have to be developed to cater for high levels of conservatism and extreme caution in design and plant life extension. Fracture mechanics based design and remaining life prediction methods coupled with multi-disciplinary materials modelling are the likely candidates for this task. Since the likelihood of failure is increased when cracks or defects are present in the structure and in particular in regions of weldments it becomes imperative to further develop predictive techniques. Fracture mechanics is routinely applied at British Energy Plc. [United Kingdom] codes of Practice R5/R6 to predict failure behaviour in equipment operating in the creep and creep/fatigue range by assessing crack initiation and growth from existing defects. This paper considers the relevant structural integrity aspects of a high temperature fracture-based methodology by considering the background to the present codes and standards that are used, role of residual stresses and the theory which deals with cracked structures at elevated temperatures for components. An international collaborative effort under the auspices of the Versailles Agreement on Materials and Standards (VAMAS) is now underway to deal with these aspects and develop recommendations for standard bodies.

Aging structures and components which are at the end of their operation design life need to prove safety levels in order to get approval for life extension. Even more stringent criteria for both clean and safe nuclear energy production are needed to improve public perception of the long term safety and societal compatibility with nuclear energy. Also the trends towards higher operating temperatures extend the life of existing nuclear power plant, mean that new and more accurate and reliable experimental data and fracture mechanics models on newer steels will be needed.

Another important aspect of improving life predictions is the treatment of the models in the presence of residual stresses arising during welding as well as fabrication, repair and unplanned overloads. These factors need to be quantified in order to improve lifetime predictions. Post weld heat treatment (PWHT) can reduce the magnitude of residual stresses, but not completely remove them nor can they remove any prior strain damage in the weld region. Current codes for design and assessment of high temperature components do not take full advantage of recent advances in the mechanistic understanding of the deformation and failure processes involved. Thus, they often lead to too a small critical defect sizes and short remaining lifetimes, particularly when pessimistic assumptions are made about residual stress. The scientific challenge is to develop methods to assess the structural integrity of components subjected to prior straining and/or containing residual stresses and to determine realistic inspection intervals for such components.
It is evident from previous work that the important criteria for modelling and life assessment in terms of fracture mechanics is first to develop an agreed testing procedure and analysis method and second to validate the models and results in the laboratory to conform to plant behaviour. Otherwise nuclear licensing authorities are unlikely to support new proposals for life extension based on invalidated data and material information. Improved accuracy in testing and analysis of data is therefore an important factor in deriving the appropriate validated crack growth properties for both parent and weld related materials. A collaborative effort to identify methods in testing and analysis of components at elevated temperature was setup in 1988. Under the auspices of the Versailles Agreement on Materials and Standards (VAMAS) committee standardization methods for testing analysis, modelling and novel applications are developed. This paper highlights the ‘Structural Integrity ‘aspects of life management for elevated temperature crack initiation and growth.

In conclusion life management of nuclear components containing defects need to be robust and based on fundamental fracture mechanics principals. With improvements in damage and crack detection techniques it is clear that evaluating failure behaviour needs to be considered at various length scales. Experimenters, modelers, standards and code writers in the nuclear field need to consider more advanced multi-disciplinary approaches in order to satisfy future stringent safety levels that will be asked of the nuclear industry.