APPENDIX B

DETAILED DESCRIPTION OF THE RESULTS OF CALCULATIONS

OECD-IAEA Paks Fuel Project
APPENDIX B of the Final Report of the OECD IAEA PAKS FUEL PROJECT contains the following papers provided by participants of the project:

B/I.  Jarne Varpoorten (Belgium, Suez-TRACTEBEL Engineering):
INTEGRAL CALCULATION WITH MELCOR 1.8.5

B/II.  Pasi Junninen (Finland, VTT):
THERMAL HYDRAULIC ANALYSIS OF PAKS-2 FUEL CLEANING TANK INCIDENT WITH APROS

B/III.  Kari Pietarinen (Finland, VTT):
FUEL PERFORMANCE ANALYSIS OF PAKS-2 FUEL CLEANING TANK INCIDENT WITH FRAPTRAN

B/IV.  Klaus Trambauer (Germany, GRS):
ANALYSES OF THE PAKS INCIDENT WITH ATHLET-CD

B/V.  László Perneczky (Hungary, AEKI):
THERMAL HYDRAULIC ANALYSIS OF THE PAKS-2 INCIDENT

B/VI.  István Trosztel (Hungary, AEKI):
THERMAL HYDRAULIC CALCULATION OF THE PAKS-2 INCIDENT BY ATHLET

B/VII.  Attila Molnár (Hungary, AEKI):
FUEL BEHAVIOUR CALCULATION OF THE PAKS-2 INCIDENT

B/VIII.  Emese Szabó (Hungary, AEKI):
ACTIVITY RELEASE CALCULATION OF THE PAKS-2 INCIDENT

B/IX.  Gábor Légrádi, Ildikó Boros, Attila Aszódi (Hungary, BME):
ONE-PHASE 3D CFD INVESTIGATIONS ON THE DEVELOPMENT OF THE SERIOUS INCIDENT IN PAKS NPP, APRIL OF 2003

B/X.  Gábor L. Horváth (Hungary, VEIKI):
THERMAL-HYDRAULIC AND SOURCE TERM CALCULATIONS FOR THE OECD PAKS FUEL PROJECT
B/XI. Yu. Zvonarev, V. Kobzar, A. Volchek (Russia, KI):
RESULTS OF THE PAKS-2 CLEANING TANK INCIDENT SIMULATION WITH ICARE/CATHARE CODE

B/XII. Peter Matejovic (Slovak Republic, IVS):
ANALYSES OF THE PAKS INCIDENT WITH THE ASTEC CODE

B/XIII. Martin Vogel (Slovak Republic, VUJE):
THERMAL HYDRAULIC CALCULATIONS OF PAKS INCIDENT USING RELAP5/MOD3.2.2

B/XIV. K.C. Wagner (USA, SNL):
ANALYSIS OF THE PAKS EVENT USING MELCOR
APPENDIX B/I.

INTEGRAL CALCULATION
WITH MELCOR 1.8.5

Jarne Varpooten
(Belgium, Suez-Tractebel Engineering)

OECD-IAEA Paks Fuel Project
I.1. Introduction

In the framework of the “OECD-IAEA Paks Fuel Project”, Suez-Tractebel Engineering (TE) developed a MELCOR 1.8.5 model for the cleaning tank facility, installed in the refuelling pool of the NPP unit Paks-2. This appendix describes the calculations which have been performed within this program and how they relate to the general observations made during the incident and to the calculation results of the other participants.

Paragraph I.3 of this appendix addresses the objectives of Suez-Tractebel Engineering with regard to this project. A description of the code and the developed model are given in § I.4 and § I.5. The results of the different calculations which have been performed at TE are presented in § I.6. In § I.7, the calculation results and the manner in which they relate to the general observations made during the incident and to the results of the other participants are discussed.

I.2. Conclusion

Suez-Tractebel Engineering performed two sets of calculations with regard to the “OECD-IAEA Paks Fuel Project”. The first set (blind calculation) has been presented at the final project meeting. The thermalhydraulic results of this calculation are in agreement with the observations during the incident.

The second set of calculations (open calculation, § I.6.2 and § I.7.3) has been performed after the final project meeting. These calculations take into account the reflections made during and after the final discussion with the other participants of the project. The open calculation presents not only a good agreement between the thermalhydraulic behaviour in the tank and the observations during the incident, but also shows that the degradation of the fuel assemblies follows a clear physical logic, which corresponds with the information available from the cleaning tank facility:

1. Start of uncovery of the fuel assemblies (2h20’);
2. Start of fuel assembly degradation (4h35’);
3. Amount of core degradation (fission product releases);
4. Amount of core degradation (maximal clad temperature – melting of the cladding material);
5. Heat transferred out of the facility with respect to the calculations performed by the University of Budapest [2].

It should be mentioned that certain elements of the thermalhydraulic behaviour of the TE open calculation are in conflict with those of other participants of the program. The main differences lay in the reduction of the water level in the downcomer, more in particular whether or not the outlet tube becomes uncovered, and the amount of heat which is evacuated out of the cleaning tank by radiation.

However, it is difficult to state which of both types of calculations reflects best the actual behaviour in the cleaning tank since only the information with regard to the fuel degradation (cladding temperature, fission product release) is available to evaluate the calculations. For both types of calculations, the behaviour of the fuel and the amount of fuel degradation are similar.
I.3. Objectives

An incident, during which degradation of the fuel assemblies occurs, is an ideal occasion for users of fuel behaviour codes to examine their code’s capabilities and restrictions. At the same time, the users can exchange information and experiences with regard to best engineering practices for thermalhydraulic and fuel degradation analysis.

The main reasons for Suez-Tractebel Engineering to participate to the “OECD-IAEA Paks Fuel Project” were to:

— Validate the general nodalisation approach applied at TE;
— Validate the in-home developed “Released Fission Product Activity Tool”;
— Gain experience in modelling VVER-type fuel assemblies;
— Exchange information with the other participants with regard to coupled thermalhydraulic and fuel degradation modelling.

I.4. Code description

Suez-Tractebel Engineering used the integral code MELCOR 1.8.5 to perform the calculations in the framework of the “OECD-IAEA Paks Fuel Project”. MELCOR 1.8.5 is a severe accident code developed by Sandia National Laboratories (SNL), which allows the user to perform integral severe accident calculations. As a result, MELCOR 1.8.5 is not only capable to calculate the thermalhydraulic behaviour of a facility, but also the possible degradation of the fuel assemblies and fission product releases.

I.5. Model description

I.5.1. Nodalisation scheme

An overview of the nodalisation scheme can be found in Fig. 2. Control volumes 201, 202 and 203 represent the air-filled space between the inner and the outer vessel walls of the cleaning tank. The coolant enters the cleaning tank in control volume 101, from where it goes through the fuel assemblies (represented by control volume 104) towards the cleaning tank head (control volume 102) and the downcomer region (control volume 103).

In Fig. 2, the flow paths between the different control volumes are indicated in blue. In order to obtain a proper modeling of the transient phase, a fictive control volume has been added (control volume 105). This control volume is situated between the downcomer region and the outlet tube. The different bypass flow paths (holes and accidental bypass) also inject in this control volume. The accidental bypass has a flow area of 0.012 m², while a total surface of 0.0083 m² has been attributed to the bypass flow through the holes at the bottom of the fuel assemblies.

I.5.2. Power distribution

Two axial zones (core and lower head zone) are defined for the core fuel assemblies. The first zone, representing the lower head, contains three axial levels of the fuel assemblies. Twenty axial levels have been modelled for the active core region. Based on the geometry in which the fuel assemblies are placed inside the tank, the core has been divided in three concentric radial rings. The tank core has been modelled as a BWR type reactor, in order to track the contribution of the cladding and canister zirconium to hydrogen formation.
Axial level input

In the AEKI database [1], the axial power density profiles of all the assemblies, present in the cleaning tank at the moment of the incident, are given as a function of time. Since the axial power density profiles of all the fuel assemblies are similar and since the evolution in time is not significant, the axial power density profile of fuel assembly 1 at 30 hours is considered as representative and is implemented in the model.

Ten axial power density profiles are considered in the AEKI database [1]. In the MELCOR 1.8.5 model, 20 axial levels are considered. The additional power levels have been obtained by linear interpolation.

In Table 1, input data, required by MELCOR 1.8.5, are indicated.

Table 1. MELCOR 1.8.5 axial level input.

<table>
<thead>
<tr>
<th>Axial level</th>
<th>Lowest point</th>
<th>Height</th>
<th>Corresponding HS</th>
<th>Rel. power dens.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Lower plenum</td>
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<td></td>
<td></td>
<td></td>
</tr>
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<td>1</td>
<td>0.359</td>
<td>0.162</td>
<td>10231</td>
<td>0</td>
</tr>
<tr>
<td>2</td>
<td>0.521</td>
<td>0.162</td>
<td>10232</td>
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<td>0.073</td>
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</table>

Radial ring input

In Table 2, the input data for MELCOR 1.8.5 are indicated. The area of the radial rings is represented schematically in Fig. 1.
Table 2. MELCOR 1.8.5 radial level input.

<table>
<thead>
<tr>
<th>Radial ring</th>
<th>Cross section [m²]</th>
<th>Heat structure [-]</th>
<th>Power per ring [kW]</th>
</tr>
</thead>
<tbody>
<tr>
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<td>10290</td>
<td>42.954</td>
</tr>
<tr>
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<td>10290</td>
<td>93.348</td>
</tr>
<tr>
<td>3</td>
<td>0.68</td>
<td>10290</td>
<td>105.336</td>
</tr>
</tbody>
</table>

Fig. 1. Schematic representation of the arrangement of the fuel assemblies in the Paks cleaning tank. The boundaries of the three concentric radial rings are indicated in the figure.

I.5.3. Inlet flow

The four inlet flow tubes have been modeled as flow paths only. Each of them has a cross section of 9.2E-3 m². The mass flow rate has been fixed, throughout the entire calculation, and corresponds to a speed of 0.155 m/s (20,45 ton/hr). After a series of preliminary calculations, it has been chosen to set the inlet temperature to 60°C.

I.5.4. Pressure boundary conditions

Before the start of the transient, the pressure in the upper part of the cleaning tank is 2.25 bar abs.
Fig. 2. Schematic representation of the nodalisation scheme of the Suez-Tractebel Engineering MELCOR 1.8.5 model for the Paks Fuel cleaning tank.

I.6. Calculation results

In this paragraph, the results of two calculations are presented: a *blind* calculation and an *open* calculation. The *blind* calculation has been performed without that the results and conclusions of the other participants were taken into account. Only the information in the AEKI database...
[1] was considered during the preparation of this calculation. At the final project meeting, the results of the blind calculation were presented.

At the final project meeting, a preliminary comparison of the calculation results of the different participants was presented. Based on this comparison, a discussion took place regarding the differences between the results of the participants. This discussion resulted in a set of suggestions for adaptations to the calculation model (for every participant separately) which could have an impact on the calculation results.

After the final meeting, the suggested adaptations for the TE model were examined in detail. Some of the suggestions were translated in specific values for certain parameters and have been incorporated in a new calculation, others have not been retained. An overview and discussion of the suggestions is presented in § 1.7.2. The calculation which considers some of the made suggestions is called the open calculation.

### I.6.1. Blind calculation

![Fig. 3. tank inventory. Evolution of the steam and water mass inside the cleaning tank throughout the Paks-2 incident. The steam mass can be found on the right vertical axis.](image)
Fig. 4. liquid level. Evolution of liquid level in the downcomer region throughout the Paks-2 incident.

Fig. 5. cladding temperature. Evolution of cladding temperature in ring 2 of the core region throughout the Paks-2 incident. Each line represents the temperature evolution in one of the twenty active core axial levels.
Fig. 6. produced hydrogen. Evolution of integrated hydrogen production in the cleaning tank throughout the Paks-2 incident.

Fig. 7. integrated hydrogen mass flow rate. Evolution of integrated hydrogen mass flow rate through the degassing line and outlet tube of the cleaning tank throughout the Paks-2 incident.
Fig. 8. Zirconium oxidation. Evolution of fraction of zirconium oxidised in the cleaning tank throughout the Paks-2 incident.
I.6.2. Open calculation

Fig. 9. tank inventory. Evolution of the steam and water mass inside the cleaning tank throughout the Paks-2 incident. The steam mass can be found on the right vertical axis.

Fig. 10. liquid level. Evolution of liquid level in the downcomer region throughout the Paks-2 incident.
Fig. 11. cladding temperature. Evolution of cladding temperature in ring 2 of the core region throughout the Paks-2 incident. Each line represents the temperature evolution in one of the twenty active core axial levels.

Fig. 12. produced hydrogen. Evolution of integrated hydrogen production in the cleaning tank throughout the Paks-2 incident.
Fig. 13. integrated hydrogen mass flow rate. Evolution of integrated hydrogen mass flow rate through the degassing line and outlet tube of the cleaning tank throughout the Paks-2 incident.

Fig. 14. zirconium oxidation. Evolution of fraction of zirconium oxidised in the cleaning tank throughout the Paks-2 incident.
Fig. 15. xenon-133. Evolution of Xe-133 activity throughout the Paks-2 incident.

Fig. 16. cesium-137. Evolution of Cs-137 activity throughout the Paks-2 incident.
I.7. Discussion

I.7.1. Blind calculation

At the start of the cooling phase, several fuel assemblies are not correctly positioned on the lower grid. A part of the coolant, injected into the cleaning tank, bypasses the fuel assemblies and causes a reduced cooling of the fuel assemblies. The fuel assemblies and the coolant heat up. When the saturation temperature is reached in the cleaning tank, steam is formed. The pressure in the tank rises instantly and the formed steam pushes the water, through the downcomer region and the outlet tube, out of the cleaning tank.

According to the Paks-2 incident reports and the AEKI database [1], the uncovery of the fuel assemblies in the cleaning tank started 2h20’ after the start of the cooling phase. This moment is well predicted by the blind calculation, as can be seen in Fig. 3. The liquid level in the downcomer region drops instantly to a level below the entrance of the outlet tube (Fig. 4).

The cladding temperature of the uncovered parts of the fuel assemblies rises and when a temperature of 1100K is reached (after 4h35’) hydrogen is formed by the oxidation of the cladding’s zirconium (Fig. 5 and Fig. 6). Due to the exposure of the entrance of the outlet tube, both the steam and the formed hydrogen can escape from the cleaning tank (Fig. 7). An open circuit exists which favours the formation of hydrogen and the rise of the cladding temperature. The oxidation of zirconium represents an important source of heat which causes a steep rise in cladding and fuel temperature up till values of around 2200°C (Fig. 5).

At these temperatures, the cladding material can relocate in molten form and a large degree of fuel assembly degradation is encountered. Up to 30% of the present zirconium is oxidised towards the end of the calculation, 8 hours after the start of the cooling phase (Fig. 8).
I.7.2. Suggestions

Although the timing and the rate of the assembly uncovering are well predicted, the extent of uncovering, as found in the blind calculation, is larger than that found by the other participants. In the calculations of the other participants, the entrance outlet tube remains covered by water. This is an important difference which has large consequences.

If the entrance of the outlet tube remains covered by water, steam and hydrogen can not escape out of the cleaning tank. Hydrogen accumulates at the top of the cleaning tank and there the steam concentration becomes too low to enable oxidation of the zirconium. Steam starvation occurs. Due to the phenomenon of steam starvation less hydrogen is produced and less heat is generated. As a result, the cladding's temperatures remain lower than in the TE calculation.

The results of the blind TE calculation differ from the results of the other participants. Nonetheless, a clear physical logic exists between the different results of the blind TE calculation (water level in the tank, mass of H₂ and high cladding's temperatures).

There are two elements which indicate that the calculation of TE overestimates the amount of hydrogen formed and the maximum temperature attained in the cladding:

1. If a temperature of 2200°C would have been attained during the incident, a large part of the cladding material and the stainless steel of the upper grid would have molten. A visual inspection of the cleaning tank debris, by the NPP utility with a camera, showed no traces of molten zirconium or stainless steel;
2. Such large temperatures cause a large degree of fuel assembly degradation. The more the fuel pins degrade, the more fission products are released. During the incident, the activity in the containment was monitored. The measured activity corresponds to around 1% of the total radioactive fission product inventory. If a large degree of core degradation would have been encountered, more radioactive fission products would have been released into the containment and detected by the activity measurement chains.

At the final project meeting, a preliminary comparison of the calculation results of the different participants was presented. Based on this comparison, a discussion took place regarding the differences between the results of the participants. This discussion resulted in a set of suggestions for adaptations to the calculation model (for every participant separately) which could have an impact on the calculation results.

After the final meeting, the suggested adaptations for the TE model were examined in detail. The results of the examination for the three main suggestions are discussed in detail below.

I.7.2.1. System pressure

In comparison to the calculations of the other participants, the system pressure in the TE calculation is low. There is a difference of 0.4-0.6 bar.

The system pressure is determined automatically by MELCOR 1.8.5 based on the depth of the cleaning tank. An underestimation of the system pressure would therefore come from an underestimation of the location of the cleaning tank in the refuelling pool.
If the pressure at the exit is higher, there is a larger force acting against the flow of water leaving the cleaning tank through the outlet tube. This could lead to a smaller amount of water leaving the cleaning tank and thus a smaller reduction of the water level in the cleaning tank. As such, this could be the reason why the entrance of the cleaning tank remained covered in the other calculations, but not in that of TE.

### I.7.2.2. Emissivity

In comparison to the calculations of the other participants, the heat evacuated out of the cleaning tank through the vessel wall in the TE calculation is low. The blind calculation gave a heat transfer of only half what other participants found.

Radiation plays an important role in this incident because it represents an important mode of heat transfer out of the cleaning tank. An improper prediction of the emissivity, of the core materials and of the heat structures representing the inner and outer tank walls, could explain an underestimation of the heat transported out of the cleaning tank.

If less heat is evacuated out of the cleaning tank, more heat remains in the cleaning tank itself and higher temperatures are attained. As a result, there could be a direct link between an underestimation of the in-core and heat structures emissivities and the higher cladding temperatures, observed in the TE calculation.

### I.7.2.3. Zirconium oxidation model

Different users applied different correlations to model the oxidation of the fuel assemblies’ zirconium. In the blind calculation of TE, the standard MELCOR 1.8.5 Zircaloy oxidation correlation has been applied. However, this correlation has been developed for another type of Zircaloy than that used in VVER type assemblies. As a result, the oxidation of VVER type assemblies can not be correctly predicted by the standard MELCOR 1.8.5 correlation.

In the AEKI database [1], a correlation for the oxidation of VVER type assemblies is given. This correlation indicates an oxidation rate which is somewhat lower than that predicted by the standard MELCOR 1.8.5 correlation. As such, this difference could explain an overestimation of the amount of zirconium oxidation and the amount of hydrogen formed.

Since it is assumed that the blind TE calculation overestimates the zirconium oxidation, it has been suggested to use the oxidation correlation presented in the AEKI database [1].

### I.7.2.4. Refined nodalisation

For the nodalisation of the Paks-2 fuel cleaning tank, TE applied the same philosophy as used for the different Belgian NPP models. This means that one control volume represents the core region and one control volume represents the downcomer region. Also the lower and upper plenums are represented by single control volumes. More information with regard to the nodalisation can be found in the main text.

After a set of preliminary calculations, an extra control volume has been added to the downcomer region. Both the outlet tube, the accidental bypass as the fuel assembly holes are connected to this control volume which is situated right above the lower support plate of the cleaning tank.
The other participants applied finer nodalisations for their calculations. They are convinced that a finer nodalisation, which considers several axial control volumes for as well the downcomer as the core region, is able to predict in a more accurate way the evolution of the water level in the cleaning tank. For the problem of the cleaning tank, where gravity plays a particular role since the outlet of the tank is at the bottom of the downcomer region, a more refined nodalisation could indeed be mandatory to obtain a proper thermalhydraulic simulation of the Paks-2 incident. It has been recommended to TE to adapt the used nodalisation for the simulations of the Paks-2 incident.

However, a detailed nodalisation is not always required for reactor applications and therefore refining the nodalisation (for the reactor case) would not necessarily imply that the thermalhydraulic behaviour is predicted in a more correct way. A possible conclusion with regard to a more refined nodalisation for the Paks-2 incident would therefore not necessarily lead to a similar conclusion (and possible validation) with regard to reactor models. This was one of the main objectives of TE to participate in the “OECD-IAEA Paks Fuel Project”.

Another reason why this suggestion was not considered in the open calculation, was the available time. The use of a more refined nodalisation requires that more parameters have to be defined as there are far more control volume and flow paths.

On top of this modelling effort, there is also the validation effort which is to be performed. It is commonly known that the results of a MELCOR 1.8.5 calculation with regard to the liquid level in a vessel are very sensitive to the value used for the opening height of the flow paths between vertically stacked control volumes. It is only after all the adaptations have been validated during a set of preliminary calculations that they can be used in an incident simulation. Therefore, it is almost impossible to find a proper value and an exact thermalhydraulic justification for all these parameters in the remaining time.

Based on the two aforementioned arguments, TE has decided not to adapt the applied nodalisation.

**I.7.3.  Open calculation**

Based on the retained suggestions, new calculations have been launched. First, the impact of every suggestion has been addressed separately. After an evaluation of the impact, it has been decided whether or not to include the suggested adaptation in the final open calculation.

**I.7.3.1.  System pressure**

The location of the cleaning tank in the cleaning tank pool has been re-addressed based on the comparison of the system pressure obtained by the other participants. This has led to an adaptation of the water pool level above the cleaning tank.

A higher system pressure has an impact on the thermalhydraulic behaviour inside the cleaning tank. With the same thermalhydraulic parameters for the control volumes and junctions in the tank, the behaviour of the tank during the accident changes drastically. Hardly any fuel assembly uncovery is noticed when the same parameters are used as during the blind calculation.

In order that the output of the calculation would correspond to the data recorded during the accident, certain parameters (mostly friction in flow paths) have been adapted. Once a correct
behaviour has been obtained for the initial phase of the incident, a full incident simulation has been run.

During the analysis of the new calculation results, however, it appeared that the same amount of fuel assembly degradation has been obtained with the higher system pressure as in the blind calculation. The higher system pressure did not have the preferred effect on the fuel assembly uncovery and the evolution of the water level in the cleaning tank. Most likely the impact of the higher system pressure is compensated by the adaptations made with regard to the thermalhydraulic parameters.

I.7.3.2. Emissivity

MELCOR 1.8.5 considers different input parameters which play an important role in the radiative heat transfer. The impact of each of these parameters on the thermalhydraulic behaviour has been addressed in detail.

The first record which has been adapted with regard to the blind calculation is the input record in which the user defines the radiative exchange factors between different components of the core, the so called COR00003 record. This record defines five radiative exchange factors:

1. Radiative exchange factor for radiation from the canister to the fuel rod cladding;
2. Radiative exchange factor for radiation from the “other structure” to the adjacent canister walls or fuel rods and debris;
3. Radiative exchange factor for radiation radially outward from the cell boundary to the next adjacent cell;
4. Radiative exchange factor for radiation axially upward from the cell boundary to the next adjacent cell;
5. Radiative exchange factor for radiation from the liquid pool to the core components.

Of these five records, the latter has the largest impact on the thermalhydraulic behaviour of the cleaning tank. By changing this value, one can attain complete uncovery of the fuel assemblies or prevent any uncovery to occur.

The second set of records that have been changed are the records with regard to the radiative heat exchange between different heat structures, the so called HSRDCCCO0 records. In these records, the user can define the radiative heat transfer between two heat structures by indicating the emissivity of both surfaces and the view factor. These records have a clear impact on the thermalhydraulic behaviour of the cleaning tank during the incident, but not as large as the last parameter in the COR00003 record.

As mentioned, different combinations of the radiative exchange parameters have been tested in order to investigate the impact of the different parameters. However, for the final open calculations, the input has been defined as a function of the heat that is evacuated out of the cleaning tank. The estimation of the evacuated heat is based on the approach and results presented in [2], as this document is the most elaborated and detailed with regard to the heat loss of the cleaning tank in the AEKI database.

I.7.3.3. Zirconium oxidation model

The E110 zirconium alloy correlation which is mentioned in the AEKI database [1], has been implemented in MELCOR 1.8.5. The objective of this suggestion was that it would reduce the amount of zirconium oxidation and the amount of hydrogen produced. However,
implementing this new correlation had some side effects. Resolving these side effects required efforts which could not be delivered in the available timeframe. As a result, the impact of this suggestion has not been evaluated in detail.

I.7.3.4. Refined nodalisation

This suggestion has not been taken into account during the open calculations. The reason for this can be found in § I.7.2.4.

I.7.3.5. Final calculation

The final open calculation addresses the suggestion discussed in § I.7.2.1 and § I.7.2.2. This calculation is presented in the final report of the “OECD-IAEA Fuel Paks Project”.

Thermalhydraulic behaviour and fuel degradation

Initially, the thermalhydraulic behaviour of the open calculation is the same that found in the original blind calculation (§ I.7.1). As before, the timing of fuel assembly uncovering is well predicted (2h20’, Fig. 9). The extent of core uncovering has not changed, meaning that the entrance of the outlet tube becomes uncovered (Fig. 10). This means that also in this open calculation, steam and hydrogen (when present in the tank) can leave the cleaning tank (Fig. 13). Steam starvation does not occur in the open calculation. Up to this point the blind and open calculations present the same thermalhydraulic behaviour.

The rise in cladding’s temperature starts in both calculations at the same time and proceeds at the same rate. However, once the oxidation of the present zirconium starts, the two calculations evolve in different directions.

Due to the oxidation of the zirconium there is a new important heat source in the cleaning tank next to the residual heat. The cladding’s temperature rises further. However, with this rise in temperature, more heat is transferred by radiation from the fuel assemblies’ canister towards the inner wall of the cleaning tank. The temperature of this wall rises and it starts to heat up, again by radiation, the outer wall of the cleaning tank. At higher temperatures, this mode of heat transfer becomes more efficient and it becomes the most important way to evacuate heat out of the cleaning tank (instead of conduction).

A balance is established between the heat generated in the cleaning tank and the heat transported out of the tank. Due to the balance in heat transfer, the temperature in the cleaning tank reduces and a peak in the cladding temperature is noticed around 1400°C (Fig. 11).

The difference in evolution of the cleaning tank heat transfer between the blind and open calculation, is the origin of a difference in the thermalhydraulic behaviour. A different evolution in the water level is noticed which impacts directly the cooling of the fuel assemblies. Since water level, temperature and oxidation rate are strongly related and interfere with one another, a difference in heat transfer implies a change in the evolution of these three parameters (Fig. 10, Fig. 12 and Fig. 14). This explains the large difference with regard to the amount of hydrogen which is being formed and the amount of zirconium oxidation (Fig. 8 and Fig. 14).
Fission product releases

Based on the evolution of the activity of three isotopes (Fig. 15, Fig. 16 and Fig. 17), it can be stated that the moment at which core degradation starts is well predicted (4h35'). The magnitude of the releases, as calculated by MELCOR 1.8.5, does not coincide exactly with the values observed during the incident, but gives a good approximation given the detail of the calculation with respect to this issue.

For Xe-133 and Cs-137, the values obtained by TE are a factor 2 higher than the measured values. This most probably arises from the way in which the activity of the isotopes has been determined for the TE calculation. In fact the activities are not exactly the activities of the isotopes in the containment alone, but rather in the entire model. A sum is made of the total mass and activity of the isotopes in all the control volumes of the model.

The TE calculation seems to give an underestimation of the released I-131 activity. This could come from the fact that MELCOR 1.8.5 has the tendency to consider a large part of the present iodine under the form of CsI. The activity of the latter molecule has not been taken into account in Fig. 17.

I.7.3.6. Comparison blind/open calculation

<table>
<thead>
<tr>
<th></th>
<th>Blind calculation</th>
<th>Open calculation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Water level drop in the cleaning tank [hr]</td>
<td>2h20'</td>
<td>2h20'</td>
</tr>
<tr>
<td>Start of zirconium oxidation [hr]</td>
<td>4h35'</td>
<td>4h35'</td>
</tr>
<tr>
<td>Mass of hydrogen formed after 8 hr [kg]</td>
<td>35</td>
<td>13</td>
</tr>
<tr>
<td>Percentage of zirconium oxidised after 8hr [-]</td>
<td>33</td>
<td>12</td>
</tr>
<tr>
<td>Maximum cladding temperature [°C]</td>
<td>2200</td>
<td>1400</td>
</tr>
<tr>
<td>Maximum heat loss through tank wall [kW]</td>
<td>58</td>
<td>295</td>
</tr>
</tbody>
</table>
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E.Szabó, Z. Hózer (AEKI)

Attila Aszódi, Gábor Légrádi, Bogdán Yamaji
Budapest University of Technology and Economics (BME)
Institute of Nuclear Techniques (NTI)
THERMAL HYDRAULIC ANALYSIS OF PAKS-2 FUEL CLEANING TANK INCIDENT WITH APROS

Pasi Junninen (Finland, VTT)

OECD-IAEA Paks Fuel Project
FOREWORD

In this report the thermal hydraulic analysis results of the Paks-2 cleaning tank incident are described. The analysis was performed using the APROS simulation code. The results were presented in the Final Meeting of the OECD-IAEA Paks Fuel Project in Budapest, Hungary April 26.-27., 2007.
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1. INTRODUCTION

On April 10th, 2003 a serious incident happened at the Paks-2 nuclear power plant in Hungary, when nuclear fuel was cleaned in a special cleaning tank. The fuel had been contaminated by magnetic deposits from steam generators. The cleaning tank was designed to clean 30 fuel assemblies concurrently. Previously a cleaning tank for 7 fuel assemblies had been used successfully.

The total decay heat of 30 fuel assemblies was 241 kW. During the chemical cleaning process temperature of the tank was 90-95 °C and a powerful pump handled the cooling of the fuel assemblies. The total mass flow rate was nearly 50 kg/s. At 16.40 the chemical cleaning process was ended and the cooling mode of the cleaning process was initiated. A low-powered pump started to cool the fuel assemblies. The total mass flow rate of the cooling pump was some 6 kg/s. The coolant was pumped from a refuelling pit and the temperature of the coolant was 35 °C. There were only a few measuring elements in the tank. Only the outlet temperature of the tank, the liquid level of the refuelling pit and the liquid level of the pressurizer were the signals which could be used to determine the state of the cleaning tank. At 19.20 the pressurizer liquid level was increased by 70 mm in 20 minutes and afterwards it was concluded that liquid level had decreased in the cleaning tank because of boiling.

The cooling of the fuel assemblies was not adequate. There were 12 holes with diameter 9 mm on the lower and upper edge of the standard type fuel assembly. These holes were ignored in the thermal hydraulic design of the cleaning tank. However, these holes played an important role in the fuel assembly cooling, when the cooling mode of the cleaning tank was used. They formed a bypass channel from the cleaning tank inlet to the outlet side. Most of the coolant flowed through this bypass and an adequate amount of coolant did not flow through the assemblies. In addition, some of the fuel assemblies were not seated correctly and that incorrect seating formed an extra bypass.

The insufficient cooling of the fuel assemblies induced the heating of the fuel rods. Fuel rods had been heated far over 1000 degrees Celsius. The claddings were ruptured and heavily oxidized on April 11th, 2.15 at night when the cleaning tank was opened and cold water from the refuelling pit flowed into the tank. When cold water flew into the tank, the fuel rods shattered.

In this report the results of thermal hydraulic analysis of above described incident are given. The analysis was performed with 1-dimensional system code APROS. The incident was simulated from the beginning of the cooling mode to the opening of the cleaning tank. Reflooding of the tank was not simulated in this work.

2. APROS SIMULATION MODEL

The fuel cleaning tank of Paks2 nuclear power plant was modelled using APROS 5.06 simulation code. The six equation thermal hydraulics model of the code was used in the simulation.

The fuel rods were modelled using REACTOR components and the rods were axially divided into 23 or 24 nodes depending on the type of the fuel assembly. The standard type rods were 10 mm longer than the follower types. The lower and upper plenums were modelled with NODE components and the downcomer side of the tank was modelled with
TANK_WITH_HEAT_STRUCTURE component in order to model an internal vessel wall. The downcomer side was divided into 12 axial nodes. The downcomer consisted only of one flow channel. The bypasses were modelled with PIPE components. The flow area of the perforations was 83.3 cm² and an extra bypass area was 120 cm². The extra bypass described the flow channel which was formed by incorrect seating of the fuel assemblies.

The fuel assembly shrouds and the heat transfer from the shrouds to the downcomer side were modelled with HEAT_STRUCTURE_X components. External vessel wall was also modelled with HEAT_STRUCTURE_X component. The heat radiation between the internal and the external vessels, and the heat radiation between the fuel assembly shrouds and the internal vessel wall was modelled with HEAT_TRANS_SOLID components. Emissivity of the vessel walls was estimated to be 0.5 and emissivity of the fuel assembly shroud was set to 0.7. When defining the emissivities it was assumed that the surfaces were heavily oxidized [1]. The reference level, namely level zero, was on the bottom plate. Flow diagram of the cleaning tank is shown in Figure 1.

![Flow Diagram of the Cleaning Tank Simulation Model.](image)

Fig. 1. Flow Diagram of the Cleaning Tank Simulation Model.

3. **HEAT RADIATION BETWEEN FUEL ASSEMBLY SHROUDS AND INTERNAL VESSEL WALL**

The radiation heat transfer between the structures is very important phenomenon in heat transfer during the later phase of the incident and it has to be considered in the simulation. Positions of the fuel assemblies are shown in Figure 2. The fuel assemblies were divided into three concentric rings which was taken into account in defining the viewing factors. Fuel assemblies 1-6 formed the first ring, fuel assemblies 7-18 were in the second ring and fuel
assemblies 19-30 were in the third ring. It was assumed that the viewing factor between rings 1 and 2 was 0.5, and the viewing factor between rings 2 and 3 was 0.5. Likewise, the viewing factor between ring 3 and the internal vessel wall was set to 0.5.

In APROS simulation model, the fuel assemblies were divided into six groups according to their decay heats. Aforementioned viewing factors were divided further on for the fuel assembly groups by the number of fuel assemblies in the same group. The viewing factors are shown in Table 1. The heat radiation between the fuel assemblies in the same ring was neglected because they had nearly the same temperature. The heat radiation to the internal vessel wall was assumed only from ring 3.

![Fig. 2. Position of the fuel assemblies in the cleaning tank [2].](image)

<table>
<thead>
<tr>
<th>FA Group</th>
<th>FA Group</th>
<th>Viewing Factor</th>
</tr>
</thead>
<tbody>
<tr>
<td>1-6</td>
<td>7-11</td>
<td>0.208</td>
</tr>
<tr>
<td>1-6</td>
<td>12</td>
<td>0.042</td>
</tr>
<tr>
<td>1-6</td>
<td>13-18</td>
<td>0.25</td>
</tr>
<tr>
<td>7-11</td>
<td>19-24</td>
<td>0.104</td>
</tr>
<tr>
<td>7-11</td>
<td>25-30</td>
<td>0.104</td>
</tr>
<tr>
<td>12</td>
<td>19-24</td>
<td>0.021</td>
</tr>
<tr>
<td>12</td>
<td>25-30</td>
<td>0.021</td>
</tr>
</tbody>
</table>
4. BOUNDARY CONDITIONS

There were very few measurements taken and recorded during the incident. Only the coolant outlet temperature and the flow rate and the pressurizer water level were the measured data that could be used to comparison of the calculation and the measurements.

Center of the cleaning tank was located 10 meters under water. The end of the outlet pipe was located at the top of the spent fuel pool. Inlet temperature was 30 °C and constant inlet mass flow was 5.8 kg/s. Initial cleaning tank temperature was 58 °C.

Before the simulation was started the model was run 200 seconds with the following boundary conditions: Inlet temperature was 58 °C and inlet mass flow rate was 47 kg/s. That was done to achieve the correct initial state.

4.1. Decay Heats

The fuel assemblies in the tank were divided into six groups according to the different decay power levels of the fuel assemblies. There were six different decay heat distributions. The total decay heat of the assemblies was 241 kW. The decay heat distributions of six different assembly groups are shown in Table 2 and linear power distributions can be seen in Figure 3. Fuel assemblies 1-11 are the standard types and fuel assemblies 12-30 are the follower types.

Table 2. Decay Heats of the Fuel Assemblies

<table>
<thead>
<tr>
<th>Assembly</th>
<th>1-6</th>
<th>7-11</th>
<th>12</th>
<th>13-18</th>
<th>19-24</th>
<th>25-30</th>
</tr>
</thead>
<tbody>
<tr>
<td>Decay heat of one assembly [W]</td>
<td>7159</td>
<td>9006</td>
<td>8154</td>
<td>6694</td>
<td>8837</td>
<td>8719</td>
</tr>
<tr>
<td>Total decay heat of the group [W]</td>
<td>42954</td>
<td>45028</td>
<td>8154</td>
<td>40164</td>
<td>53022</td>
<td>52312</td>
</tr>
</tbody>
</table>
4.2. Heat Transfer Coefficient on Outer Surface of the External Vessel Wall

On outer surface of the external vessel wall heat was transferred by free convection. In the estimation of the heat transfer coefficients the proper equations for vertical wall and horizontal cover were chosen [1].

The Nusselt number for the vertical external vessel wall can be calculated by the following equation

\[
Nu = \left\{ 0.825 \frac{0.387Ra^{1/6}}{\left[1 + (0.492/Pr)^{9/16}\right]^{1/27}} \right\}^2
\]  

(1)

where

Pr is the Prandtl number and Ra is the Rayleigh number and it can be calculated as follows

\[
Ra = \frac{g \beta (T_w - T_\infty) L^3}{\nu \alpha}
\]  

(2)

Where \( g \) is gravitational acceleration, \( \beta \) is volumetric thermal expansion coefficient of water, \( T_w \) is wall temperature, \( T_\infty \) is water temperature, \( \nu \) is kinematic viscosity and \( \alpha \) is thermal diffusivity.

The Nusselt number for the horizontal external cover was calculated by the following equation

\[
Nu = 0.15Ra^{1/3}
\]  

(3)
Temperature on the outer wall of the external vessel was estimated to be \( T_s = 35 \, ^\circ \text{C} \). Water properties were determined in temperature of \( T = \frac{(T_s + T_\infty)}{2} \). The heat transfer coefficient on the outer vessel wall was calculated to be 1280 W/m²K and the heat transfer coefficient of the cover was 1450 W/m²K. The heat transfer coefficients were assumed to be constants during the simulation.

### 4.3. Heat Released from the Zirconium Oxidation Reaction

Heat generation due to oxidation was calculated with the FRAPTRAN fuel behaviour code [3] and heat flux was entered to APROS as a boundary condition with special Boundary condition modules of APROS. The cladding temperatures calculated with APROS were used as boundary condition in FRAPTRAN calculations. The first step was to calculate the transient from the beginning of the cooling mode to the instant when the cover of the tank was opened. After that, fuel behavior analysis was performed with FRAPTRAN and oxidation heat was transferred back to APROS and the transient was recalculated. Aforementioned iteration procedure was repeated three times and maximum oxidation reaction heat was aimed to correspond to the amount of total decay heat power of the fuel assemblies. Finally maximum oxidation reaction heat used in APROS calculation was 250 kW. Hydrogen production and steam starvation in the cleaning tank were ignored in this analysis. Oxidation reaction heat used in APROS calculation is shown in Figure 4.

![Fig. 4. Oxidation reaction heat and decay heat.](image-url)
5. SIMULATION RESULTS

After the first simulation run (i.e. without oxidation) the maximum cladding temperature was 1155 °C and it was calculated to be in fuel assembly 12 (see Figure 5). The maximum cladding temperature when the oxidation reaction heat was considered was 1628 °C (see Figure 6). Temperature difference between the fuel assemblies was not very large. The highest clad temperature of fuel assembly 12 among of 30 fuel assemblies can be explained by the highest and peaked decay heat. The maximum cladding temperature elevations are shown in Figure 7. The maximum temperatures at later part of the simulation were calculated to be on elevation 2.04 m from the reference level.

The liquid level of the cleaning tank settled down to 1.4 meters (see Figure 8). The pressurizer water level during the incident was increased about 70 mm at 19.20 in about 20 minutes. The liquid level decrease in the cleaning tank started 8400 seconds after the beginning of the cooling mode. It can be concluded that the voiding of the cleaning tank was well predicted by APROS code. Dotted vertical lines in Figure 8 describe the onset and the end of the voiding during the incident.

The fluid temperature at the outlet (see Figure 9) and the outlet flow rate (see Figure 10) were measured during the incident and recorded approximately once in an hour. The measured values in Figures 9 and 10 are shown with dots and the calculated values are shown with lines. Average calculated values correspond well to the measured ones. Because there was not a continuous temperature measurement, the temperature peak caused by tank voiding, can not be seen in the measured values. In calculation, outlet flow was very unsteady after the tank voiding and even backflow from the outlet pipe can be seen. To prevent the fluctuation of the solution, a large form loss coefficient of the outlet pipe had to be used.

After the cleaning tank was switched to the cooling mode, the coolant flow rate through the fuel assemblies began to decrease rapidly. Boiling started 7250 s after the onset of the simulation in fuel assembly 12. When the coolant was saturated and boiling was started inside the fuel assemblies, flow rate started to fluctuate very strongly (see Figure 11). Simultaneously as the coolant flow in the assemblies was decreasing, the coolant flow through the bypasses was increasing (see Figure 12). 1500 seconds after the onset of the cooling mode, 81 % of the coolant was flown through the bypasses. At time 5000 s, 89 % of the coolant was flown through the bypasses. As a result of this cooling conditions of the fuel rods became insufficient.

Temperatures of the fuel assembly 12 shroud, internal vessel wall, fluid inside the assembly 12, fluid in downcomer side and cladding at different elevations are presented in Figures 13-19.

In Figure 20 the total heat loss from the cleaning tank is shown. Heat losses were calculated through the cover of the tank and through the external vessel wall. Before the voiding of the cleaning tank, heat losses were almost negligible.
Fig. 5. Maximum cladding temperatures after the first simulation run.

Fig. 6. Maximum cladding temperatures after the third simulation run.
Fig. 7. Maximum cladding temperature elevation.

Fig. 8. Collapsed liquid level in the cleaning tank.
Fig. 9. Flowrate through the air letdown valve.

Fig. 10. Outlet temperature.
Fig. 11. Outlet flowrate.

Fig. 12. Bypass flowrates.
Fig. 13. Temperatures at elevation 0.5 m (Fuel assembly 12).

Fig. 14. Temperatures at elevation 0.9 m (Fuel assembly 12).
Fig. 15. Temperatures at elevation 1.3 m (Fuel assembly 12).

Fig. 16. Temperatures at elevation 1.7 m (Fuel assembly 12).
Fig. 17. Temperatures at elevation 2.1 m (Fuel assembly 12).

Fig. 18. Temperatures at elevation 2.5 m (Fuel assembly 12).
Fig. 19. Temperatures at elevation 2.7 m (Fuel assembly 12).

Fig. 20. Total heat loss from the outer surface of the tank to the storage pool.
6. CONCLUSIONS

The incident in the cleaning tank of Paks2 nuclear power plant was analyzed with APROS 5.06 simulation code. The 6-equation thermal hydraulic model of the code was used in the simulation.

The simulation was started from the instant when the cleaning tank was switched on the cooling mode. As the cooling mode was initiated, the bypass flow rates began to increase very rapidly and the coolant flow through the fuel assemblies decreased. Therefore cooling of the assemblies was not adequate. The voiding of the cleaning tank was well predicted by APROS. After the liquid level of the cleaning tank had collapsed, the fuel rods started to heat up. The minimum estimation for maximum cladding temperature (without oxidation) was 1155 °C and maximum estimation for maximum cladding temperature (with oxidation) was 1628 °C. Maximum heat from the oxidation reaction was 250 kW and it was slightly overestimated because steam starvation in the cleaning tank was ignored.

The radiation heat exchange between the fuel assemblies and between the fuel assembly shrouds and internal vessel wall was complicate to simulate. The fuel assemblies were divided in three concentric rings and the radiation heat exchange was modelled between rings 1 and 2, 2 and 3, 3 and the internal vessel wall. The radiation heat transfer inside the rings was not modelled.

Heat released from zirconium oxidation reaction was calculated with the FRAPTRAN code and it was transferred back to APROS. Thermal boundary condition for FRAPTRAN was calculated with APROS. Two-way data transfer was necessary due to the restriction of the zirconium oxidation model used in the current APROS version. Total of three iteration runs were performed in turns to achieve an oxidation heat which was in a magnitude of decay heat of the fuel assemblies.

ACKNOWLEDGEMENT

The work summarized in this paper was funded by Fortum Nuclear Services.

REFERENCES

APPENDIX B/III.

FUEL PERFORMANCE ANALYSIS OF PAKS-2 FUEL CLEANING TANK INCIDENT WITH FRAPTRAN
Kari Pietarinen (Finland, VTT)

OECD-IAEA Paks Fuel Project
FOREWORD

On 10 April 2003, an incident took place at Unit 2 of the Paks Nuclear Power Plant in Hungary that led to severe damage in a number of VVER-440 type fuel assemblies. The fuel behaviour in this cleaning tank incident has been recently analysed at VTT as a part of an international exercise under the established OECD-IAEA Paks Fuel Project. The FRAPTRAN fuel transient behaviour code was applied. The results summarised in this report are those that have been earlier presented in the Project’s final meeting on 26 to 27 April in Budapest Hungary.

Fortum Nuclear Services Oy is acknowledged for funding this study.
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1. INTRODUCTION

On 10 April 2003, a number of fuel assemblies were severely damaged in an incident at the Paks nuclear power plant Unit 2. The assemblies in question were placed in a cleaning tank submerged to a depth of 10 m at the spent fuel storage pool. The tank designed to accommodate 30 fuel assemblies had been used to chemically clean fuel bundles of the crud build-up that had been observed to deteriorate heat transfer in operation. Previously a smaller cleaning tank for seven assemblies had been used successfully.

During the cleaning, the total decay heat power was 241 kW, which was effectively removed by a forced flow, and the water temperature did not exceed 95 °C. After the cleaning was finished, the mass flow rate was decreased from 50 kg/s to some 6 kg/s. This reduced flow turned out to be insufficient, however, due to unforeseen flow conditions which were a consequence of a failure to properly seat a few of the assemblies in the tank, and of obvious flow bypass through 12 holes of 9 mm in diameter in each assembly, the effect of which had not been taken into account in the thermal hydraulic design. Under poor cooling that resulted, the maximum cladding temperatures rose well above 1000 °C leading to heavy oxidation and rupture of Zr-1%Nb claddings as well as to massive oxidation of the Zr-2.5%Nb shroud material. After 9.5 hours into the unsuccessful cooling phase, the lid of the underwater tank was opened. The hot, heavily oxidised fuel rods were quenched by the cold refuelling pool water and a large number of those were shattered.

In the visual inspection, all of the 30 assemblies were found to be severely damaged. In, 2005 they were removed from the tank and were enclosed in special canisters, temporarily stored in the spent fuel pool of the plant.

A model of a single rod was made to run in the FRAPTRAN code, taking into account the flow channel configuration with characteristics from the VVER-440 fuel assemblies placed in the cleaning tank. The tank thermal hydraulics and the system code are separately described in [1].

This report comprises the description and results of the six fuel transient behaviour calculations that were made in this exercise.

2. CASE DESCRIPTION

The basic task in fuel transient performance analyses is to provide an estimate whether the cladding of a rod will remain intact or not. Fuel cladding integrity may be challenged by such phenomena as heat transfer crisis, oxidation and embrittlement - potentially greatly enhanced by heat transfer crisis, and pressurisation by fission gases or pellet-to-cladding mechanical interaction (PCMI).

The initial fuel design parameters and boundary conditions (Tables 1 and 2, and Figures 2 and 3) given in OECD-IAEA Paks Fuel project database have been used in this analysis [2]. Thermal boundary conditions for the transient fuel behaviour analysis have been calculated using the APROS system code.
In the APROS simulation and in the fuel performance analysis, the fuel assemblies were divided into six groups according to their levels of decay heat [1]. In this report fuel assemblies are referred as groups, rather than by their fuel assembly numbers. The resulting division of the fuel assemblies into six groups is given in Table 3. The length of the simulated transient was 9 hours 37 minutes.

2.1. Tools for fuel behaviour analysis

The steady-state fuel performance code FRAPCON was used to calculate the burnup dependent properties for the code FRAPTRAN. The thermal boundary conditions for FRAPTRAN were calculated with the APROS code. A simple perl programming language script set was implemented to transfer initial and boundary condition data from the OECD-IAEA Paks Fuel Project database and from the APROS results. Due to highly voided coolant conditions, a system code like APROS was required for the determination of the thermal boundary conditions. Another perl script set was implemented to transfer calculated heat generated by zirconium oxidation with FRAPTRAN back to APROS. Two-way data transfer was necessary due to the restriction of the zirconium oxidation model used in the current APROS version. Three iterations were made running APROS and FRAPTRAN by turns. This iterative solution was used to determine maximum reaction heat generation by oxidation corresponding to decay heat of 30 assemblies.

2.1.1. Fuel performance codes

FRAPCON-3 is a fuel performance code for steady-state analyses and was used to compute the burnup-dependent parameters of a fuel rod and to generate initial input conditions for FRAPTRAN transient analyses. The main phenomena and output parameters from FRAPCON-3 include 1) radial burnup-dependent heat conduction through the fuel and cladding, 2) cladding elastic, thermal, creep, and plastic deformations, 3) fuel cladding mechanical interaction (PCMI), 4) fission gas release (FGR) 5) fuel rod internal gas pressure, 6) radial heat transfer between fuel and cladding, 7) cladding oxidation, and 8) heat transfer from the cladding to the coolant. The code contains necessary material properties, water properties, and heat transfer correlations.

FRAPTRAN is a FORTRAN language computer code developed for the U.S. Nuclear Regulatory Commission to calculate the transient thermal and mechanical behaviour of LWR fuel rods. FRAPTRAN code with VTT modifications includes a finite difference heat conduction model for the transient thermal solution, a mechanical model developed at VTT, and the MATPRO material properties package.

To take into account the effects of high burnup, burnup-dependent UO₂ thermal conductivity and a revised model for cladding mechanical properties with the effects of oxidation and hydrides, in addition to irradiation damage, have been introduced in FRAPTRAN in an upgrading campaign that started a few years ago. Burnup dependent fuel rod initial conditions can be obtained from the companion FRAPCON-3 steady-state code.

The FRAPCON and FRAPTRAN codes did formerly employ a rather simple 1D thin shell mechanical model for the stress-strain analysis of the cladding. A new approach with a Finite Element (FE) model has been implemented basing on a FE solver that had been earlier created at VTT.

To estimate cladding oxidation and heat generation from the metal-water reaction, conservative type Baker-Just and best-estimate type Cathcart-Pawel correlations are available
in the current FRAPTRAN version. Cathcart-Pawel oxidation correlation was not used in these analyses because of too high temperature diffusion gradient in zirconium cladding for the model at the end of the transient.

2.1.2. Fuel rod model

The pre-processing prior to calculation includes entering the fuel rod geometry and boundary condition data to input files of FRAPCON-3 and FRAPTRAN. The input files for FRAPCON and FRAPTRAN were generated by using a simple perl-script. The fuel rod geometry is schematically presented in Figure 1 and with details in Table 1.

The standard fuel rods and the follower type fuel rods were divided in 23 and 24 axial evenly spaced lengths, respectively. The length of the standard fuel assembly is 10 mm longer than the VVER-440 follower fuel assembly. The temperature distribution model used a mesh set-up, where the pellet stack was divided in 18 radial rings. For the FGR model, the pellet is divided in 45 equal-volume radial rings. The fuel pellet stack and cladding were divided in 10 and 2 equal-area radial rings, respectively. The first radial node is placed at the fuel pellet inner radius and the last at the fuel pellet outer radius.

The thermal boundary conditions for the single fuel rod model were calculated with the APROS system code. The maximum number of FRAPTRAN thermal boundary zones is ten. The 23 or 24 nodal temperatures calculated with APROS were reduced to 10 nodal temperatures for FRAPTRAN calculations by interpolation. The data transfer is visualised in Figure 2.

2.2. Initial state parameters

The description of the VVER-440 type fuel rod construction is based on the information that is provided in OECD-IAEA Fuel project database [2]. In the FRAPCON code, specific E110 (Zr-1%Nb) material properties are not implemented. Instead, Zircaloy material properties were used in the steady-state calculations. Being delivered in re-crystallised state, Zircaloy-2 was a preferred choice. The cladding specifications do not make a drastic difference and this approximation should be satisfactory. Regarding clad creep, specifically FRAPCON does not make any distinction between cladding materials. The FRAPTRAN code does contain E110 cladding material properties and these were in use in the analyses.

The decay heat profiles are presented in Figure 3. The decay heat profiles of fuel assembly groups 1-2 and 5-6 are cosine shaped while the decay heat power profiles of group 3 and 4 are more peaked towards the upper end of the rod. The same profiles were used in the base-irradiation and transient calculations.

3. SIMULATION RESULTS

The fuel behaviour analyses cover all of the six fuel assembly groups. However, mainly the calculation results of the third fuel assembly group are presented because it represents the worst case among the six groups. The length of the simulated transient (9 h 37 min) covers the time between the start of the cleaning and opening of the tank lid.

All calculation results presented in tables and figures of this report consider results of the third iteration run.
3.1. Base irradiation

The key fuel performance parameters for the base irradiation are presented in Table 4 and visualised in Figure 4 for the fuel assembly group 3. Emphasis in this work is on fuel transient performance, the base irradiation calculation results are not discussed in detail.

3.2. Transient analysis

The FRAAPTRAN transient calculations indicate that the metal-coolant reaction produces significant amount energy in relation to the decay heat during the incident. Because of this additional heat source, a decision to continue calculations with APROS and FRAAPTRAN by turns was made. The calculated chemical reaction heat here is from the oxidation of the fuel rod cladding only; the fuel assembly shroud, spacer grids etc. are not considered. It is also important to note that the calculated chemical reaction heat after the second iteration run is the one, which was transferred as the heat source for the third APROS calculation run. The constant decay heat and the calculated heat from the chemical reaction during the transient are visualised in Figure 5.

FRAAPTRAN calculations suggest that the chemical reaction between the cladding and steam essentially starts after 3 hours 57 minutes and reaches a value of 441 kW at 5 hours 57 minutes into the cooling mode. The characteristics of the self-accelerating reaction are clearly shown in Figure 4. The maximum reaction powers during the first and second iteration run were 145 kW and 250 kW, respectively. The reaction heat curves of the first and second iteration runs were transferred back to APROS as a boundary condition. The effect of potential steam starvation on the reaction rate was not modeled in this analysis.

The reaction energy release is highest in fuel assembly group 3 due to the peaked decay heat profile and the highest clad outer surface temperatures. Data of Table 5 reiterates the property of the correlation to activate at about 727 °C. Chemical reaction power, greater than the decay heat power, was achieved after two iteration runs calculated with APROS and FRAAPTRAN in turn. The oxidation rate decreases after the instant of its highest intensity. At this point the zirconium has been oxidised in the hottest part (middle section) of fuel rods, which leads to decreased oxidation rate. However the Zirconium oxidation continues in top portions of the fuel rods above the water level as illustrated in Figure 6.

In the claddings, the total zirconium mass is 989.58 kg. By the end of the transient, 46.7 % of zirconium has been oxidised and 15.1 kg of hydrogen produced. The hydrogen production rate and the cumulative production by oxidation of the fuel rod claddings are shown in Figure 7. The total average equivalent cladding reacted (ECR) value and the total amount of energy produced by oxidation are shown in Figure 8.

The liquid coolant level starts to decrease 2 hours 20 minutes after the beginning of the cooling mode. In this situation, the coolant in fuel assembly channels is voided. The rod surface heat transfer is immediately impaired and the cladding temperatures strongly rise due to the stored energy in the fuel. The bottom, top and maximum cladding temperatures of fuel assembly group 3 are visualised in Figure 9. The fuel rod key parameters at the beginning of oxidation and at the time of burst are presented in Tables 5 and 6. The fuel rod burst can be clearly seen in Figure 10 by observing the fuel rod internal pressure.

Calculated maximum fuel pellet and cladding outer surface temperatures in the fuel assembly groups are close to each other during the simulated period. The maximum fuel pellet and cladding outer surface temperatures during the transient are between 1866 and 2043 °C,
between 1582 and 1628 °C, respectively. The highest fuel centre and cladding temperatures were estimated for fuel assembly 12 due to the decay heat profile peaking at the elevation of 1.6 m from the bottom of the rod. The temperature differences between the fuel assembly groups are relatively small and derive mainly from differences in heat radiation and decay heat.

A conclusion from the fuel transient behaviour analysis is that in Paks-2 cleaning tank conditions, Baker-Just oxidation model predicted almost identical temperature threshold for all of the six fuel assembly groups. Temperature thresholds for zirconium oxidation and zirconium cladding burst are 727 °C and between 960 and 1127 °C, respectively. The cladding oxidation is predicted to start after 3 h 50 min to 4 h 3 min into the cooling mode. The cladding burst for all groups was predicted to occur 1 h 2 min later on the average. The cladding burst is predicted to occur first in fuel assembly 12 after 4 h 34 min from the beginning of simulation, 36 min before the measured activity increase. The rods in the rest of the fuel assembly groups were predicted to lose their integrity within 31 minutes.

The activity concentrations in the coolant and the release through the tank chimney were regularly measured and the first activity increase was observed 5 h 10 min after the beginning of the cooling mode [2]. This result indicates that cladding temperatures and/or cladding oxidation were overestimated. As noted above the metal-water reaction heat is significant compared with the decay heat.

The top and middle parts of the assemblies were predicted to deform and become heavily oxidised while the bottom part of fuel assemblies would be expected to remain intact as shown in Figure 11. The cladding state in all other fuel assembly groups at the end of the transient is qualitative the same.

4. CONCLUSIONS

The purpose of the calculations was to simulate the fuel rod behaviour during Paks-2 cleaning tank incident and to determine whether the fuel rod integrity is lost due to high temperature oxidation. The OECD-IAEA Paks Fuel project database was used for the steady-state and transient fuel behaviour calculations. The thermal boundary conditions for the transient analysis were calculated with the APROS system code. The calculated burnups of the analysed fuel assembly groups are between 10.5 and 26.2 MWd/kgU.

The main conclusion of the analysis is that fuel failure was predicted for all fuel assembly groups. The cladding burst was calculated to occur at first in fuel assembly group 3, at 4 h 34 min from the beginning of simulation; rods in all other groups were predicted to lose their integrity within 31 minutes. The cladding burst was calculated to occur 36 minutes earlier than what was detected. However, the effect of potential steam starvation due to hydrogen production in the cleaning tank on the reaction rate was not modeled in this analysis.

The Baker-Just oxidation model was used to simulate the high temperature metal-water oxidation. The top and middle parts of the assemblies are expected to deform and become heavily oxidised while the bottom part of fuel assemblies would remain intact.
5. REFERENCES


## APPENDIX I. VVER-440 FUEL ASSEMBLY DESIGN PARAMETERS

Table 1. The VVER-440 fuel assembly design parameters [2].

<table>
<thead>
<tr>
<th>Fuel assembly</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Shroud material</td>
<td>Zr-2.5%Nb</td>
</tr>
<tr>
<td>Number of fuel rod per fuel assembly</td>
<td>126</td>
</tr>
</tbody>
</table>

### Cladding

<table>
<thead>
<tr>
<th>Cladding material</th>
<th>Zr-1%Nb (E110)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cladding outer diameter [mm]</td>
<td>9.15</td>
</tr>
<tr>
<td>Cladding thickness [mm]</td>
<td>0.65</td>
</tr>
<tr>
<td>Cladding inner diameter (calculated) [mm]</td>
<td>7.85</td>
</tr>
<tr>
<td>Fuel rod pitch in bundle [mm]</td>
<td>12.2</td>
</tr>
</tbody>
</table>

### Pellets

<table>
<thead>
<tr>
<th>Pellet shape</th>
<th>Chamfered</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel pellet material</td>
<td>UO&lt;sub&gt;2&lt;/sub&gt;</td>
</tr>
<tr>
<td>Fuel stack height (active length) [mm]</td>
<td>2420&lt;sup&gt;b&lt;/sup&gt;, 2320&lt;sup&gt;c&lt;/sup&gt;</td>
</tr>
<tr>
<td>Pellet diameter [mm]</td>
<td>7.57</td>
</tr>
<tr>
<td>Central hole diameter (average) [mm]</td>
<td>1.2 – 1.8 (1.6)</td>
</tr>
<tr>
<td>Pellet average height [mm]</td>
<td>10.5</td>
</tr>
<tr>
<td>Pellet density (average) [g/cm&lt;sup&gt;3&lt;/sup&gt;]</td>
<td>10.6</td>
</tr>
<tr>
<td>Relative density [%]</td>
<td>96.715</td>
</tr>
<tr>
<td>Pellet &lt;sup&gt;235&lt;/sup&gt;U enrichment [%]</td>
<td>3.6</td>
</tr>
<tr>
<td>Pellet surface mean arithmetic roughness [µm]</td>
<td>1.0</td>
</tr>
<tr>
<td>Fuel grain size [µm]</td>
<td>15.0</td>
</tr>
</tbody>
</table>

### Rod

| Fill gas pressure [Mpa] | 0.5 |
| Plenum free volume (calculated) [cm<sup>3</sup>] | 4.26 |
| Plenum length (calculated) [mm] | 90.0 |

<sup>a</sup> Geometrical values are at room temperature 20 °C.
<sup>b</sup> Standard fuel assembly, groups 1 and 2.
<sup>c</sup> Follower type fuel assembly, groups 3 – 6.
Fig. 1. Schematic of the FRAPCON and FRAPTRAN fuel rod model.
Table 2. The steady-state operation parameters for FRAPCON-3.3.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coolant pressure at the inlet of fuel assembly [Mpa]</td>
<td>12.3</td>
</tr>
<tr>
<td>Coolant temperature at the inlet of fuel assembly [°C]</td>
<td>267.0</td>
</tr>
<tr>
<td>Coolant mass flux [kg/m²/s]</td>
<td>2900.0</td>
</tr>
<tr>
<td>Fast neutron flux coefficient [(n/m²/s)/(W/g)]</td>
<td>2.46⋅10¹⁶</td>
</tr>
</tbody>
</table>

Table 3. Fuel assembly groups [2].

<table>
<thead>
<tr>
<th>Fuel assemblies</th>
<th>Group</th>
<th>Average decay heat [kW/m]</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 – 6</td>
<td>1</td>
<td>2.96</td>
</tr>
<tr>
<td>7 – 11</td>
<td>2</td>
<td>3.72</td>
</tr>
<tr>
<td>12</td>
<td>3</td>
<td>3.51</td>
</tr>
<tr>
<td>13 – 18</td>
<td>4</td>
<td>2.89</td>
</tr>
<tr>
<td>19 – 24</td>
<td>5</td>
<td>3.81</td>
</tr>
<tr>
<td>25 – 30</td>
<td>6</td>
<td>3.76</td>
</tr>
</tbody>
</table>
Fig. 2. Data transfer between the database, the APROS code and the fuel performance codes.

Fig. 3. Relative decay heat profiles.
Fig. 4. Key parameters of fuel assembly group 3 during base irradiation.
Table 4. The properties of irradiated fuel assemblies calculated with the FRAPCON-3.3 code.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Group 1</th>
<th>Group 2</th>
<th>Group 3</th>
<th>Group 4</th>
<th>Group 5</th>
<th>Group 6</th>
</tr>
</thead>
<tbody>
<tr>
<td>Max. linear heat generation rate [kW/m]</td>
<td>17.5</td>
<td>21.9</td>
<td>22.4</td>
<td>20.6</td>
<td>22.7</td>
<td>22.1</td>
</tr>
<tr>
<td>Max. fuel temperature [°C]</td>
<td>856.0</td>
<td>964.0</td>
<td>1020.0</td>
<td>966.0</td>
<td>1040.0</td>
<td>1020.0</td>
</tr>
<tr>
<td>Max. average fuel temperature [°C]</td>
<td>697.0</td>
<td>738.0</td>
<td>800.0</td>
<td>766.0</td>
<td>814.0</td>
<td>800.0</td>
</tr>
<tr>
<td>Max. rod internal pressure [MPa]</td>
<td>1.43</td>
<td>1.61</td>
<td>1.53</td>
<td>1.56</td>
<td>1.51</td>
<td>1.50</td>
</tr>
<tr>
<td>Min. gap [µm]</td>
<td>48.0</td>
<td>11.0</td>
<td>18.0</td>
<td>19.0</td>
<td>28.0</td>
<td>30.0</td>
</tr>
<tr>
<td>Max. cladding axial deformation [mm]</td>
<td>7.0</td>
<td>7.2</td>
<td>6.8</td>
<td>6.9</td>
<td>7.0</td>
<td>6.9</td>
</tr>
<tr>
<td>Cladding axial deformation [mm] (^d)</td>
<td>-0.04</td>
<td>-0.14</td>
<td>-0.02</td>
<td>-0.22</td>
<td>-0.12</td>
<td>-0.10</td>
</tr>
<tr>
<td>Max. burnup [MWd/kgU] (^d)</td>
<td>12.6</td>
<td>31.2</td>
<td>15.9</td>
<td>28.2</td>
<td>16.0</td>
<td>15.7</td>
</tr>
<tr>
<td>Average burnup [MWd/kgU] (^d)</td>
<td>10.5</td>
<td>26.2</td>
<td>10.3</td>
<td>21.6</td>
<td>13.6</td>
<td>13.4</td>
</tr>
<tr>
<td>FGR [%] (^d)</td>
<td>0.077</td>
<td>0.193</td>
<td>0.102</td>
<td>0.168</td>
<td>0.099</td>
<td>0.098</td>
</tr>
<tr>
<td>Rod internal pressure [MPa] (^d)</td>
<td>0.630</td>
<td>0.686</td>
<td>0.651</td>
<td>0.673</td>
<td>0.651</td>
<td>0.649</td>
</tr>
</tbody>
</table>

\(^d\) At the end of base-irradiation at temperature of 20 °C and pressure of 1.01 bar.
**APPENDIX IV. SIMULATION RESULTS: TRANSIENT FUEL BEHAVIOUR**

Table 5. Fuel and cladding key parameters at the time of cladding oxidation initiation.

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>4:3</td>
<td>825</td>
<td>727</td>
</tr>
<tr>
<td>2</td>
<td>4:1</td>
<td>840</td>
<td>727</td>
</tr>
<tr>
<td>3</td>
<td>3:50</td>
<td>882</td>
<td>727</td>
</tr>
<tr>
<td>4</td>
<td>3:57</td>
<td>853</td>
<td>727</td>
</tr>
<tr>
<td>5</td>
<td>4:1</td>
<td>833</td>
<td>727</td>
</tr>
<tr>
<td>6</td>
<td>4:1</td>
<td>832</td>
<td>727</td>
</tr>
</tbody>
</table>

Table 6. Fuel and cladding key parameters at the time of cladding burst.

<table>
<thead>
<tr>
<th>Group</th>
<th>Time [h:min]</th>
<th>Elevation [m]</th>
<th>Fuel temp. [°C] e</th>
<th>Clad temp. [°C] e</th>
<th>ECR [%] e</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>5:5</td>
<td>1.46</td>
<td>1400</td>
<td>1127</td>
<td>16.7</td>
</tr>
<tr>
<td>2</td>
<td>5:5</td>
<td>1.46</td>
<td>1470</td>
<td>1126</td>
<td>16.8</td>
</tr>
<tr>
<td>3</td>
<td>4:34</td>
<td>1.76</td>
<td>1339</td>
<td>960</td>
<td>5.6</td>
</tr>
<tr>
<td>4</td>
<td>4:45</td>
<td>1.66</td>
<td>1344</td>
<td>967</td>
<td>5.9</td>
</tr>
<tr>
<td>5</td>
<td>4:35</td>
<td>1.66</td>
<td>1318</td>
<td>977</td>
<td>6.5</td>
</tr>
<tr>
<td>6</td>
<td>4:35</td>
<td>1.66</td>
<td>1321</td>
<td>981</td>
<td>6.4</td>
</tr>
</tbody>
</table>

* e In burst elevation.
Fig. 5. Decay heat and metal-coolant reaction heat.

Fig. 6. Oxidation of fuel rod claddings.
Fig. 7. Hydrogen production by oxidation of fuel rod claddings.

Fig. 8. Total zirconium reacted and reaction energy.
Fig. 9. Fuel pellet and cladding thermal properties of FA group 3.
Fig. 10. FA group 3 fuel rod internal pressure and clad state during transient
Fig. 11. FA group 3 cladding state at the end of the transient.
APPENDIX B/IV.

ANALYSES OF THE PAKS INCIDENT WITH ATHLET-CD

Klaus Trambauer (Germany, GRS)

OECD-IAEA Paks Fuel Project
I.1.1. Description of the code

The system code ATHLET-CD (Analysis of Thermal-Hydraulics of LEaks and Transients with Core Degradation) [1] is designed to describe the reactor coolant system thermal-hydraulic response, core damage progression, fission products and aerosol behaviour during severe accidents, to calculate the source term for containment analyses, and to evaluate accident management measures [2]. It is being developed by GRS in cooperation with the Institut für Kernenergetik und Energiesysteme (IKE), University of Stuttgart. The development and validation of ATHLET-CD are sponsored by the German Federal Ministry of Economics and Technology (BMWi). The system code also includes the aerosol and fission product transport code SOPHAEROS, which is being developed by the French Institut de Radioprotection et de Sûreté Nucléaire (IRSN).

The ATHLET-CD structure is highly modular in order to include a manifold spectrum of models and to offer an optimum basis for further development. It contains the original ATHLET models for a comprehensive simulation of the thermal-hydraulics in the reactor coolant system. The ATHLET code [3] comprises: (a) a thermo-fluid-dynamic module, (b) a heat transfer and heat conduction module, (c) a neutron kinetics module, (d) a general control simulation module, and (e) the general solver of differential equation systems FEBE (Forward Euler - Backward Euler [4]). The thermo-fluid-dynamic module includes two different fluid-dynamics equation systems: (a) a six-equation model, with fully separated balance equations for liquid and vapour, complemented by mass conservation equations for up to 5 different non-condensable gases and by a boron tracking model, and (b) a five-equation model, with a mixture momentum equation and a full-range drift-flux formulation for the calculation of the relative velocity between phases. Specific models for pumps, valves, steam separators, mixture level tracking, critical flow, etc. are also available in ATHLET.

The rod module ECORE consists of models for fuel rods, absorber rods (AIC and B4C) and for the fuel assembly including BWR canister and absorber. The module describes the mechanical rod behaviour (ballooning), zirconium and boron carbide oxidation (Arrhenius type rate equation), Zr-UO2 dissolution, as well as melting of metallic and ceramic components. The melt relocation (candling model) is simulated by rivulets with constant velocity and cross section, starting from the node where rod failure is predicted to occur. The model accounts for oxidation, freezing, re-melting, re-freezing and melt accumulation due to blockage formation. The feedback to the thermal-hydraulics considers steam starvation and blockage formation [5]. Besides the convective heat transfer, energy can also be exchanged by radiation between fuel rods and to surrounding core structures.

The release of fission products is modeled by rate equations or by a diffusion model within the module FIPREM [6]. The transport and retention of fission products and aerosols in the reactor coolant system are simulated by the code SOPHAEROS [7].

For the simulation of debris bed a specific model MESOCO [8] is under development, with its own thermal-hydraulic equation system, coupled to the ATHLET fluid-dynamics on the outer boundaries of the debris bed. The transition of the simulation of the core zones from ECORE to MESOCO depends on the degree of degradation in the zone. The code development comprises also late phase models for core slumping, melt pool behaviour and vessel failure.

The code system ATHLET/ATHLET-CD can be coupled to the containment code system COCOSYS, and it is the main process model within the German nuclear plant analyzer ATLAS. The ATLAS environment allows not only a graphical visualization of the calculated results but also an interactive control of the simulation.
The code validation is based on integral tests and separate effect tests, as proposed by the CSNI validation matrices [9], and covers thermal-hydraulics, bundle degradation as well as release and transport of fission products and aerosols. Post-test calculations have been performed for the out-of-pile bundle experiments CORA-13, CORA-W2, QUENCH-03, QUENCH-05, QUENCH-06 [10], QUENCH-07 [11], QUENCH-08, QUENCH-09, QUENCH-10, and QUENCH-11, as well as for the in-pile experiments PHÉBUS FPT0, FPT1 [12], FPT2, FPT3, and FPT4. The TMI-2 accident is used to assess the code for reactor applications [13]. Also the Lehrstuhl für Energiesysteme und Energiewirtschaft (LEE) of the Ruhr Universität Bochum substantially contributes to the code validation [14], [15].

I.1.2. Model extension for the simulation of the cleaning tank

The model approach of BWRs is applied to the simulation of VVER-440 with fuel assemblies surrounded by shrouds. For this it is intended to suppress the simulation of the absorber elements used by default in BWRs with an additional input option. This option was not programmed yet. The model was accordingly amended. A careful examination of the energy balance of all convective and radiative heat fluxes of the shrouds in the core and the core surroundings indicated also some discrepancies which have been corrected.

The model review also hinted the incorrect allocation of assemblies to the radial subdivision. To fulfill the energy balance for the radiative heat transfer between the shrouds, the number of assemblies from the inner to the outer rings must increase by a constant increment. Therefore the number of assemblies allocated to the four rings was corrected as described in the nodalisation scheme.

I.1.3. Description of input model and nodalisation scheme

The ATHLET-CD input model consists of 5 main sections:

1. Description of the thermofluid dynamic objects (TFO)
2. Description of the heat conduction objects (HCO)
3. Description of the core components (rods and shrouds)
4. Input of general simulation control module (GSCM)
5. Table input section for boundary conditions, GCSM functions and material properties

The configuration of the entire model of the cleaning tank is defined by the TFOs. Its nodalisation scheme is depicted in Fig. 1. There are four groups of TFOs:

1. The inlet section consist of BOTTOM and INLET providing the inlet flow.
2. The top section consists of TOP1 and TOPVALVE to simulate the air outlet flow.
3. The intermediate objects are the main flow channels ASSEMB-1, ASSEMB-2, FOLLOW-1, FOLLOW-2 and the surrounding tank volume BYPASS1 and BYPASS2.
4. A separated fluid system is the annular GAP to model the cylindrical cleaning tank wall.
Fig. 1. ATHLET-CD Nodalisation scheme for cleaning tank.

The four main flow channels are connected at the inlet and outlet with BOTTOM and TOP1. The surrounding tank volume is connected at the top with TOP1 and both TFOs are interconnected by BY-CROS over the entire height by cross flows. The bottom of BYPASS1 is linked via the shroud penetrations BOTHOL-1 and BOTHOL-2 with the entrance of ASSEMB-1 and ASSEMB-2. The flow due to incorrect seating from BOTTOM to BYPASS2 is simulated by one single flow path BOTCRO-1. The cleaning tank outlet flow is simulated by the flow path OUTLET or OUTLET2.

The four main channels and the surrounding tank volume are axially subdivided into 24 nodes, 20 equally spaced along the heated length of 2.44 m and each two nodes in the unheated part below (0.64 m) and above (0.48 m). This subdivision defines the nodalisation of the connected core components and heat conduction objects too. Therefore the annular GAP has the same axial subdivision.

The geometrical data and hydraulic characteristics of the main flow channels have been taken from the GRS standard data sets for VVER analyses, they remain unchanged since the beginning of the analyses in 2003. The bore cross section of the assembly seating is 20 cm². The diameter of the 12 penetrations per fixed assembly is 9 mm with a total form loss coefficient of 1.53. The bypass due to incorrect seating has a cross section of 120 cm², a hydraulic diameter of 6 cm, and total form loss coefficient of 1.17. The surrounding tank
volume is characterized by a total cross section area of 1.2 m² and a hydraulic diameter of 4 cm. The cross section area of inner part BYPASS1 is 0.8 m², the remaining 0.4 m² are assigned to the outer part BYPASS2. The upper plate at elevation 3.0 m is simulated by an additional form loss coefficient in BYPASS1 only.

The INLET mass flow is constant in time (6.0 kg/s, 30 °C). The TOPVALVE is linked with a constant pressure boundary condition (1 bar) and has a time dependent cross section of 0.7 cm² (time < 8600 s), 1.4 cm² (8800 s < time < 10800 s), and 2.1 cm² (time > 12600 s) to allow higher venting flow later in the transient. The pressure in the cleaning tank is imposed by the OUTLET and the constant pressure boundary condition (2.3 bar) during the first 3 h of the transient. In order to avoid strong oscillation of tank pressure and outlet flow, the hydraulic model has been replaced by OUTLET2 with controlled mass flow by the end of the transient.

For the simulation of the heat losses from the cleaning tank to the water pool two HCOs VESSELINN and VESSELOUT have been introduced as inner and outer structure of GAP. The inner diameter and the wall thickness are 1.488 m and 6 mm for the inner wall and 1.780 m and 10 mm for the outer one. The inner surface of VESSELINN is connected with BYPASS2. The outer surface of VESSELOUT is linked with a heat transfer coefficient of 1000 W/m²/K to the constant pool temperature (30 °C). GAP is filled with nitrogen (2.3 bar). For the convective heat transfer between the structures and the nitrogen, a constant heat transfer coefficient of 10 W/m²/K has been defined. An emissivity coefficient of 0.8 was used for the thermal radiation between the structures across GAP.

The fuel rods and shrouds are simulated by the core components ROD1, ROD2, ROD3, and ROD4. The four components are identical except the power distribution. The overall length is 3.0 m, the heated length 2.44 m and the unheated nodes 0.32 m and 0.24 m at the bottom and top. The model assumes the same length for fuel rods and shrouds. Therefore an average value was chosen to adequately simulate the surfaces contributing to the oxidation potential. The axial discretisation is taken from the TFO, i.e. 20 axial nodes along the heated length and one node for the lower and upper plug respectively. The radii of fuel pellet, inner clad and, outer clad are 3.785 mm, 3.86 mm, and 4.55 mm. The pitch of the triangular grid is 12.2 mm. The shroud thickness and perimeter are 2.0 mm and 540 mm. The initial oxide layer thickness of cladding and shroud inner and outer surface is 5.0 µm, and of the cladding inner surface 0.5 µm. An emissivity coefficient of 0.7 was used for the thermal radiation of the rod and shroud surfaces. The void caused by the central pellet hole was considered by increased fuel porosity of 4 %. The upper plug plenum volume is 4.26 cm³, the initial gas pressure 1.0 MPa at room temperature (293 K).

In previous and recent calculations the four components were assigned to the assembly groups as depicted in table 1. ROD1 and ROD2 were assigned to the fixed assembly groups one and two. ROD3 was assigned to the follower groups three and four and accordingly ROD4 to the group five and six. The power of ROD3 and ROD4 corresponds to the average power of the merged assembly groups. As described in the previous chapter, this irregular number of assemblies per component violates the energy balance for the radiative heat transfer between the shrouds. Therefore the assignment was corrected as shown in the second last column by keeping the input data for the bundle power. The code normalizes the power distribution accordingly to the total input power which results in the actual power distribution in the last column. The final arrangement of the core components and the axial power distribution are depicted in Figure 2.
Table 1. Assignment of bundle power

<table>
<thead>
<tr>
<th>Assembly Number</th>
<th>Power (W)</th>
<th>ATHLET-CD</th>
<th>Number of assemblies</th>
<th>Input data Power (W)</th>
<th>Number of assemblies</th>
<th>Actual data Power (W)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 - 6</td>
<td>7159</td>
<td>ROD1</td>
<td>6</td>
<td>7159</td>
<td>3</td>
<td>7101</td>
</tr>
<tr>
<td>7 - 11</td>
<td>9006</td>
<td>ROD2</td>
<td>5</td>
<td>9006</td>
<td>6</td>
<td>8932</td>
</tr>
<tr>
<td>12</td>
<td>8154</td>
<td>ROD3</td>
<td>7</td>
<td>6903</td>
<td>9</td>
<td>6847</td>
</tr>
<tr>
<td>13 - 18</td>
<td>6694</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>19 - 24</td>
<td>8837</td>
<td>ROD4</td>
<td>12</td>
<td>8778</td>
<td>12</td>
<td>8707</td>
</tr>
<tr>
<td>25 - 30</td>
<td>8719</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Fig. 2. Arrangement of core components and axial power profile

The core components ROD1 and ROD2 are thermally coupled with ASSEMB-1 and ASSEMB-2 as well as with BYPASS1. ROD3 is linked with FOLLOW-1 and BYPASS1. The outermost component ROD4 is thermally coupled with FOLLOW-2 and BYPASS2 as well as by radiation heat transfer with the inner surface of VESSELINN. Accordingly to the number of assemblies of the core components, the TFOs ASSEMB-1, ASSEMB-2, FOLLOW-1, and FOLLOW-2 represent 3, 6, 9, and 12 flow channels.

1.1.4. Results

In this chapter the results of the calculation from 19th May are described by means of the data specified for the benchmark [16]. The data provided for the comparison adopt the specified format:
—— Time = 0 corresponds to time of the start of intermediate cooling mode with low water flowrate (10/APR/2003 at 16.40).
—— Time = 34200 s corresponds to the opening of the hydraulic locks of the cleaning tank cover (11/APR/2003 at 02.10).
—— The transient data are recorded approximately every 10 s.
—— Node position is given in distance from the lower plate (lower plate position is at level 0 and the heated length of the assemblies ranges from 0.64 to 3.08 m).
—— The rods are axially subdivided in 22 nodes, 20 of them with equidistant spacing of 0.122 m in the heated region (length = 2.440 m). The lower and upper unheated part are respectively one node with 0.320 m and 0.240 m length.

As described above the calculation simulates 4 representative rod types. In the tables provided for the comparison columns at the positions of assembly 12 and assemblies 19-24 are filled with zero to keep the identical assignment of the columns.

Over the entire transient time (32400 s) 840000 time steps have been executed and 14000 time steps were recorded. These recorded data were reduced to 9000 records by eliminating data within a time step increment of about 1 s. The remaining data were smoothed by a sliding averaging over 20 time steps. These smoothed data were further reduced to 2957 records by eliminating data within a time step increment of about 10 s. The total calculation time was 35 h.

The inlet mass flow rate is 6 kg/s (21.6 t/h) and constant in time. The inlet temperature is equal the water pool temperature of 30 °C. The initial temperature in the cleaning tank was 58 °C.

![Graphs showing mass flow rate and temperature of the cleaning tank outlet.](image)

**Fig. 3. Mass flow rate and temperature of the cleaning tank outlet.**

Fig. 3 shows the mass flow rate and temperature of the cleaning tank outlet. The outlet temperature decreases fast (35 min) to about 34 °C and remains then almost constant. The outlet mass flow rate is approximately the same as the inlet flow minus the mass flow rate through the air letdown valve, which is negligible compared with the main flow. At 7000 s
(2 h) the cleaning tank water becomes saturated and over a very short time of 5 min the cleaning tank is voided (8640 – 9000 s) which is indicated by the sharp peak of the mass flow (20 kg/s) and outlet temperature (80 °C). After the voiding the outlet temperature approaches slowly the inlet temperature.

The voiding is also seen in Fig. 4, which presents the pressure and water level in the cleaning tank, by the steep water level drop and the pressure increase. This pressure increase results in the acceleration of the water outflow. With the voiding the hydraulic system becomes unstable due to the submerged gas volume, which is clearly seen by the pressure oscillations. To avoid unrealistic outflow oscillations in a hydraulic system with very low damping, the hydraulic simulation of the outflow (pipe with pressure boundary condition) was replaced by a controlled mass flow with decreasing deviation (from ± 0.5 to 0.005 kg/s) of the average value (6.0 kg/s) with increasing time. After about 4 h the water level (0.7 m) and the average pressure (2.2 bar) in the cleaning tank remain constant.

The next figure shows the maximum cladding temperature and the fluid temperature at the assembly outlet and in the top of the cleaning tank. In the first two hours there is a slow temperature increase up to 7060 s. At this time the fluid becomes saturated in FOLLOW-1, node 23 at elevation 3.20 m, and the temperatures remain almost constant up to 9140 s, when the voiding of the tank is completed. Afterwards the temperature increase is accelerated. While the temperatures of the three inner components (ROD1, ROD2, ROD3) are very similar, the outer most assembly ROD4 temperatures are much lower in the later phase, when the heat transfer by radiation becomes more and more important. The maximum temperatures approach 1381 °C (ROD2), 1375 °C (ROD1), 1339 °C (ROD3), and only 1027 °C in ROD4. The plotted fluid temperatures are significantly lower, since the rod temperatures drop from the maximum to the upper end by about 120 K in the three inner components and by 150 K in the outer. It is interesting to note that the temperature of BYPASS1 governs the temperature in the TOP of the cleaning tank. It is the highest value in the time less than four hours and lies between the fluid temperatures of ROD3 and ROD4 later on. The fluid temperature at the top of BYPASS2 is in between the TOP and ROD4 temperatures. The fluid temperature oscillations reflect the mass flow oscillations, particularly in the time between 3 and 5 h.
Fig. 5. Maximum cladding temperature and fluid temperature at assembly outlet.

Fig. 6. Mass flow rate at bundle inlet and through bypass.

The mass flows are depicted in Figures 6 and 7a. For better comparison the mass flow rate in the assemblies are taken at the Bottom Of Fuel (BOF). The mass flow rate in the assemblies and through the perforations is the mass flow per assembly. Before the saturation commences (2 h) the mass flows in the assemblies have declined to 0.02 to 0.03 kg/s. In the aftermath of 5 h the flow oscillates in the range of ± 0.4 g/s with the tendency of higher amplitudes in the followers. In the time between (2 – 5 h) the flow oscillates very heavily as it can be seen in the figure. At 2 h the mass flow rates are 3.85 kg/s due to incorrect seating and 0.158 kg/s through the perforation per assembly or 1.42 kg/s for the 9 fixed assemblies ROD1 and ROD2. The residual flow of 0.73 kg/s is going through the assemblies. At 6 h the mass flow rates are 4.21 kg/s due to incorrect seating and almost 0.2 kg/s through the perforation per assembly.
The averaged residual flow going through the assemblies is in the order of 3 g/s as derived from the mass flow rate in air outlet.

Fig. 7. Mass flow rate at air letdown and heat losses to water pool.

Table 2. Mass flow rate in air outlet valve

<table>
<thead>
<tr>
<th>Time (h)</th>
<th>Area (mm²)</th>
<th>( \frac{p_{\text{hydrogen}}}{p_{\text{total}}} )</th>
<th>Density (kg/m³)</th>
<th>Mass flow (g/s)</th>
<th>Hydrogen mass flow (g/s)</th>
<th>Velocity (m/s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>71</td>
<td>0.00</td>
<td>970.0</td>
<td>13.00</td>
<td>0.00</td>
<td>0.19</td>
</tr>
<tr>
<td>3</td>
<td>141</td>
<td>0.00</td>
<td>0.973</td>
<td>0.98</td>
<td>0.00</td>
<td>7.1</td>
</tr>
<tr>
<td>4</td>
<td>212</td>
<td>0.00</td>
<td>0.743</td>
<td>2.03</td>
<td>0.00</td>
<td>12.9</td>
</tr>
<tr>
<td>5</td>
<td>212</td>
<td>0.50</td>
<td>0.250</td>
<td>1.00</td>
<td>0.10</td>
<td>20.0</td>
</tr>
<tr>
<td>6</td>
<td>212</td>
<td>0.92</td>
<td>0.070</td>
<td>0.53</td>
<td>0.30</td>
<td>36.0</td>
</tr>
<tr>
<td>8</td>
<td>212</td>
<td>0.98</td>
<td>0.048</td>
<td>0.43</td>
<td>0.32</td>
<td>42.0</td>
</tr>
</tbody>
</table>

To discuss briefly the mass flow in the air letdown some characteristic data are compiled in Table 2. In previous calculations the oxidation was significantly underestimated. Therefore the cross section area was increased from 71 to 212 mm² to allow higher flow rates thus increasing the hydrogen generation rate. At 3 h the steam flow rate is equal the estimated value about 1 g/s and later on at 4 h two times larger, which lies in the general uncertainties of the investigation. After this time the oxidation commences and hydrogen – steam mixture escapes via the air outlet with increasing hydrogen content, which is seen by the relative partial pressure \( \frac{p_{\text{hydrogen}}}{p_{\text{total}}} \) and subsequently decreasing mass flow rate. On the other hand the volume flow rate increases as seen by the velocity. After 6 h the hydrogen concentration is almost constant as well as the molar flow rate with 0.167 mol/s, which is equivalent to an evaporation rate of 3 g/s or to the maximum hydrogen generation rate of 0.33 g/s.
It is worth to use this information by looking to the heat losses from the vessel outer wall to the water pool in figure 7b. At the time of 8 h these heat losses have reached 256 kW, value which is 15 kW higher than the decay heat (241 kW). To close the energy balance one has to consider the additional heat from the oxidation (48 kW) and the heat to the fluid, which is leaving the system via the air outlet (14 kW). The difference between the enthalpy flow at the inlet and the outlet can be neglected. The residual heat of 19 kW corresponds to the stored heat which is relatively small compared to the overall heat flow. This means that the system is almost in steady state conditions, which has been already seen in Figure 5 with the very small heat up rate of 23 K/h (0.006 K/s) at that time.

![Graph](image1)

**Fig. 8. Axial cladding temperature profile at time 14690 s and 32400 s.**

Figure 8 shows the axial cladding temperature profiles at time 14690 s, the time when the maximum cladding temperature reaches 800 °C, and at the end of the calculation at 32400 s. During the early phase (up to 4 h) the temperature profile reflects the power profile. In the later phase the temperature profile is more influenced by the heat losses. As mentioned above, the upper ends of the rods are 120 to 150 K colder than the maximum temperatures.

Figure 9 depicts the maximum fuel and cladding temperatures as function of time. These temperatures are nearly identical. Due to the relatively low decay heat and low heat losses from the fuel to the fluid the temperature differences between fuel and cladding are less than 2 K after the voiding of the system. The maximum cladding temperatures as function of time have been commented already with figure 5.
Fig. 9. Maximum fuel and cladding temperatures as function of time.

Figure 10 presents the internal rod pressure and the maximum relative oxidation degree of cladding and shroud. For the simulation of the mechanical cladding behaviors, the default options for Zircaloy® have been used (failure at relative strain of 38%). Furthermore the model does not consider the tight packed rod bundle, which constrains the ballooning. Thus the plots showing the mechanical cladding behavior are included for completeness and should not be discussed in detail. Nevertheless the cladding burst of the inner components fits fairly well the recorded data of the facility (first release detected at 18600 s). The failure time of the ROD1 to ROD4 is 16966, 16036, 16434, and 19842 s.

Fig. 10. Internal rod pressure and the relative oxidation degree of cladding and shroud.
The cladding and shroud oxidation is calculated with the Arrhenius equation using Sokolov’s data for mass increase as presented in Table 3.

Table 3. Coefficients of Arrhenius equation

<table>
<thead>
<tr>
<th>Coefficient</th>
<th>Dimension</th>
<th>$T &lt; 1800$ K</th>
<th>$1800 &lt; T &lt; 1900$ K</th>
<th>$T &gt; 1900$ K</th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td>(g/cm²)s²/s</td>
<td>1.59D0</td>
<td>8.97022750D+05</td>
<td>0.982D0</td>
</tr>
<tr>
<td>B</td>
<td>J/mol</td>
<td>191554.D0</td>
<td>3.89743250D+05</td>
<td>172931.D0</td>
</tr>
</tbody>
</table>

For the initial oxide layer thickness of cladding and shroud a value of 5.0 µm was input. The maximum value of the one-sided cladding oxidation was $C_{MAX} = 53.5\%$ and of the two-sided shroud oxidation $K_{MAX} = 19.8\%$. Considering the thicknesses of cladding (0.69 mm) and shroud (2.00 mm), the maximum oxidized metal layer thickness is for both, cladding and shroud, about 400 µm.

The same trend as for the maximum local oxidation degree is seen in Figure 11 for the relative oxidation of all claddings and shrouds with a maximum value of $C_{KXI} = 5.29\%$ as well as for the accumulated hydrogen generation with $A_{CH2} = 4.63$ kg. The relative low oxidation degree indicates that the oxidation process was limited by the steam availability. In the same figure the hydrogen generation rate is plotted. As already mentioned above, the averaged hydrogen generation rate is approximately 0.33 g/s in the last two hours ($t > 7$ h).

![Graph](image1.png)

**Fig. 11. Relative oxidation degree and hydrogen generation (mass and rate).**

The figures 12 and 13 present the axial profile of the rod diameter (mm) and oxide layer thicknesses (µm) at the end of the calculation (32400 s). As it is seen in the figure the fuel rod bursts if the relative plastic strain exceeds 38%, i.e. if the outer rod diameter reaches 12.5 mm which is larger than the pitch (12.2 mm). As described above already, the model does not consider that the ballooned cladding contacts the neighboring rods before reaching the burst criteria. Therefore the model under predicts the extent of the ballooned region.
Fig. 12. Axial profile of the rod diameter and cladding oxide layer.

The figure with the cladding oxide layer thickness shows that the oxidation in the two fixed assemblies (ROD1 and ROD2) is less than in the follower ROD3. As mentioned already above, the steam flow in the follower during the oxidation phase is somewhat larger than in the fixed assemblies, thus providing more steam available for oxidation. The very sharp drop of the oxide layer thickness after the maximum value indicates that no steam is available for oxidation at the higher elevation. The much lower oxide layer thickness in the outer follower (ROD4) is due to the lower temperature and therefore limited by the kinetics. Also the shape of the oxide layer profile, which is the same as the temperature profile, is a clear indication that there is no steam starvation.

Fig. 13. Oxide layer profile of the inner and outer surfaces of the outer shroud wall.
The shroud model calculates individually the oxidation for the four different shroud walls. In the figure 13 the oxide layers of the inner and outer surfaces of the outer wall are plotted. The oxide layer profiles of the inner surface reflect those of the claddings, only the absolute numbers are somewhat smaller due to the slightly lower temperatures. The oxide layer profile at the outer surface is similar to the temperature profile. Due to the fact that the flow in the cleaning tank is simulated by two parallel flow channels with cross connections which allow a two dimensional flow pattern, the concentration profile is very flat and a sharp oxidation front is not seen as in the one dimensional flow inside the assemblies. Nevertheless, the partial pressure of the hydrogen is relatively high and thus the oxidation is limited by the steam availability and by the kinetics. Therefore, the oxide layer thickness of the relatively cold outer most wall is much smaller than the others.

Table 4. Final temperature and oxide layer thickness distribution at elevation 1.8 m.

<table>
<thead>
<tr>
<th></th>
<th>ROD1</th>
<th>ROD2</th>
<th>ROD3</th>
<th>ROD4</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cladding temperature</td>
<td>1186</td>
<td>1233</td>
<td>1203</td>
<td>967</td>
</tr>
<tr>
<td>Fluid temperature, main flow</td>
<td>1183</td>
<td>1229</td>
<td>1196</td>
<td>957</td>
</tr>
<tr>
<td>Shroud temperature, inner wall</td>
<td></td>
<td>1188</td>
<td>1182</td>
<td>1033</td>
</tr>
<tr>
<td>Shroud temperature, side wall</td>
<td>1163</td>
<td>1204</td>
<td>1162</td>
<td>913</td>
</tr>
<tr>
<td>Shroud temperature, outer wall</td>
<td>1167</td>
<td>1199</td>
<td>1115</td>
<td>760</td>
</tr>
<tr>
<td>Fluid temperature, bypass flow</td>
<td>1022</td>
<td>1022</td>
<td>1022</td>
<td>784</td>
</tr>
<tr>
<td>Relative strain of cladding</td>
<td>11.1</td>
<td>12.2</td>
<td>2.3</td>
<td>12.9</td>
</tr>
<tr>
<td>Cladding outer oxide layer thickness</td>
<td>101</td>
<td>187</td>
<td>322</td>
<td>77</td>
</tr>
<tr>
<td>Shroud inner oxide layer thickness, inner wall</td>
<td></td>
<td>140</td>
<td>273</td>
<td>117</td>
</tr>
<tr>
<td>Shroud inner oxide layer thickness, side wall</td>
<td>87</td>
<td>155</td>
<td>256</td>
<td>50</td>
</tr>
<tr>
<td>Shroud inner oxide layer thickness, outer wall</td>
<td>89</td>
<td>151</td>
<td>198</td>
<td>13</td>
</tr>
<tr>
<td>Shroud outer oxide layer thickness, inner wall</td>
<td></td>
<td>141</td>
<td>146</td>
<td>67</td>
</tr>
<tr>
<td>Shroud outer oxide layer thickness, side wall</td>
<td>121</td>
<td>154</td>
<td>141</td>
<td>30</td>
</tr>
<tr>
<td>Shroud outer oxide layer thickness, outer wall</td>
<td>124</td>
<td>152</td>
<td>112</td>
<td>10</td>
</tr>
</tbody>
</table>

For better understanding of the model approach used in the calculation, the complex radiation heat transfer between the shrouds of the different core components is discussed in the following. It has already been mentioned that in ATHLET-CD the BWR canister model is used for the description of the VVER shrouds. For VVERs the cruciform absorber elements are not simulated. Therefore, the outer shroud wall sees the inner shroud wall of the next radial component. The side walls see each other i.e. the net heat flow by radiation is zero. Table 4 depicts the final temperature and oxide layer distribution (time = 32400 s) at elevation 1.8 m just below the mid plane. ROD2 is one of the inner components, it has the highest bundle power and the axial power peak is just below the mid plane. Therefore ROD2 has at this elevation the highest temperatures. The cladding temperatures are higher than the shroud temperatures i.e. all inner surfaces of the shroud receive heat from the rods. The outer surfaces of the inner and outer shroud walls of ROD2 transmit heat to the outer surfaces of the outer wall of ROD1 and the inner wall of ROD3. Thus in ROD2 the temperatures of the inner and outer shroud walls are lower than the side wall temperatures. In contrast in ROD1 the
outer shroud wall temperature is higher than the side wall temperature and in ROD3 the inner shroud wall temperature is higher than the side wall temperature. The largest differences of shroud wall temperature are found in ROD4, the most outer core component which sees the cold cleaning tank wall. The shroud outer wall transmits heat to that tank wall and the shroud inner wall receives heat from ROD3. Hence in ROD4 the shroud inner wall temperature is even higher than the rod temperatures. The large differences of the temperatures affect beside the steam availability the formation of the oxide layers. The inner oxide layer thicknesses range from 13 µm to 273 µm and the outer oxide layer thicknesses from 10 µm to 154 µm. It is evident that such complex situation requires adequate models as they are provided by the code. On the other hand it is assumed that a higher resolution of the transport processes for the hexagonal shrouds would not effect in general the heat flux distribution by radiation.

Finally the fission product and fuel release is discussed for completeness. The total relative release is plotted in figure 14. Following a suggestion made in the final meeting the burst release was suppressed in this calculation. The release was calculated by simple rate equations based on data from ORNL [17]. The options for low burn-up and low oxidation degree of fuel were set. From this it follows that the release rate of the high volatile fission products (Xe, Cs, I) is reduced by a factor of 2 and the fuel (U) release is not enhanced (as for hyper-stoichiometric fuel oxides). For Te the release is reduced by a factor of 40 compared to I (for high burn-up fuel), as long the oxidation degree of the cladding is less than 90 % (Due to a coding error, this condition was not set and the calculated release of Te overestimated. In the figure this error is corrected). For better visibility the release data have been magnified as indicated by the legend (Te * 10, U * 1000). The relative release at the end of the calculation is 7.68E-3 for Xe and Cs, 6.16E-3 for I, 3.05E-4 for Te, and 8.51E-7 for U.

![Histories of released mass from the fuel rods](image)

**Fig. 14. Total relative release of fission products and fuel.**

**I.1.5. Conclusion**

The first analyses of the Paks incident with ATHLET-CD were already performed in 2003. In the frame of the OECD-IAEA Paks fuel project the data basis and code models have been successively improved as well as weaknesses and errors in the code detected and eliminated.
The calculation described in this report has been finished on May 19th. The computing time of the 9 h transient was 35 h. The integral calculation covers the thermal-hydraulics, the fuel behavior, and the activity release. The calculation results were post processed to create the tables to be delivered for the benchmark. According to the specification a set of 6 tables were produced:

1. Histories of thermal-hydraulic parameters (29 variables as function of time).
2. Profiles of thermal-hydraulic parameters by the time the maximum cladding temperature exceeds 800°C and at \( t = 32400 \) s (12 variables as function of elevation).
3. Histories of fuel parameters (23 variables as function of time).
4. Profiles of fuel parameters before quenching (24 variables as function of elevation).
5. Histories of release from the fuel rods (5 variables as function of time).
6. Profiles of axial power distribution (6 variables as function of elevation).

The transient data consist of 2957 records with a time increment of approximately 10 s. The axial profiles consist of 22 records. 20 nodes with equidistant spacing of 0.122 m in the heated region and one node with 0.320 m and 0.240 m length for the lower and upper unheated part respectively.

Besides the bottom and top region, the thermal hydraulic system is modeled by 6 parallel flow channels, four to simulate the flow within the assemblies and two radial interconnected channels for the surrounding fluid. Heat losses to the water pool are simulated only over the double-walled cylinder barrel (water pool temperature 30 °C). The 30 fuel assemblies are modeled by 4 representative components, each two for the fixed assemblies and for the followers. The number of assemblies per component is 3, 6, 9, and 12 and the bundle power is 7101, 8932, 6847, and 8707 W/assembly.

The total decay power is 241 kW. The inlet flow rate and temperature are 6 kg/s and 30 °C. The cleaning tank outlet flow is modeled during the first 3 h by a pipe with constant pressure boundary condition (2.3 bar) and during the later 6 h by a controlled mass flow to limit the flow oscillations in the system. The mass flow through the air letdown decreases from 2.0 to 0.4 g/s over the time from 4 to 8 h. The average hydrogen generation rate is limited to 0.33 g/s, the total hydrogen generation is 4.63 kg. At the end of the calculation the maximum temperature is 1380 °C at almost stationary conditions (heat up rate 23 K/h). The temperatures of the outer most assemblies are 350 K lower, i.e. the calculated temperatures lie in the expected range.

The chronology of the main event is in the order of appearance:

- Beginnig of saturation in the cleaning tank 7060 s.
- Beginning of water level drop in the cleaning tank 8640 s.
- Beginning of core super heating 9140 s.
- Maximum cladding temperature exceeds 800 °C 14690 s.
- Start of intense Zr oxidation 15840 s.
- Rod failure in ROD2 16036 s.
- Rod failure in ROD3 16434 s.
- Rod failure in ROD1 16966 s.
- Rod failure in ROD4 19842 s.
- Maximum cladding temperature exceeds 1200 °C 21240 s.
The timing of the water level drop is well captured by the calculation, but the release is estimated to early by 40 min.

The maximum oxide layer thicknesses are in the range of 100 to 500 µm except the cold walls of the outer most shrouds. The total relative oxidation of all claddings and shrouds is only 5.3 % due to the steam starvation in the cleaning tank.

The relative release (released mass / inventory) of fission products and of the fuel at the end of the calculation is 7.68E-3 for Xe and Cs, 6.16E-3 for I, 3.05E-4 for Te, and 8.51E-7 for U.

The end state of the simulation is depicted in figure 15. The colors indicate the temperatures (yellow > 1300 °C, red > 600 °C, blue saturation) and the void (white = 1, blue = 0) in the system. The water level is at the same height in the whole system at about 0.7 m elevation. Furthermore the ballooning of the cladding and the oxidation profile is visualized. Most oxidation layer are too thin for proper display, only in ROD3 (FOLLOW-1) the oxide layer is visible just below the maximum ballooning.

The conclusion is that the Paks incident was well predicted and the most processes were simulated adequately by the code. The estimated maximum temperatures are sensitive to the hydraulic simulation, the air letdown flow and the heat losses to the water pool.

Fig. 15. End state of simulation with ATHLET-CD.
REFERENCES


APPENDIX B/V.

THERMAL HYDRAULIC CALCULATION OF THE PAKS-2 INCIDENT
László Perneczky (Hungary, AEKI)

OECD-IAEA Paks Fuel Project
**Code description**

The RELAP5/MOD3.3 code has been developed for best-estimate transient simulation of light water reactor coolant systems during postulated accidents. The code models the coupled behaviour of the reactor coolant system and the core for loss-of-coolant accidents and operational transients such as anticipated transient without scram, loss of offsite power, loss of feedwater, and loss of flow. A generic modelling approach is used that permits simulating a variety of thermal hydraulic systems. Control system and secondary system components are included to permit modelling of plant controls, turbines, condensers, and secondary feedwater systems.

**Nodalisation**

Two different nodalisations have been developed for the analysis. A more detailed one lumped the 30 fuel assemblies into 4 groups, but non-physical oscillations were produced by this one as soon as the upper plenum turned to two-phase conditions. Therefore a simple nodalisation was applied, as shown by Fig. 1, where all assemblies were represented by a single flow channel. The main parts of the cleaning tank are modelled as follows:

- Vol 4, Vol 6 and Vol 590 = pool water volume
- Jun 5 = pump for cooling mode A
- Jun 7 = pump for cooling mode B
- Vol 58, Vol 60 and Vol 62 = single flow channel representing 30 fuel assemblies
- Vol 570 and Vol 500 (1 –4) = upper part of tank
- Vol 60 (2 – 6) = 5 nodes for active part of fuel rods (5 x 0,50 m). In the calculation presented the active part of the fuel rods was divided to 20 nodes: Vol 60 (2 – 21), i.e. 20 x 0,125 m.
- Hs 100 – 106 = average heat slabs
- Hs 110 – 116 = hot pins

The following by-pass areas were considered:

- incorrect seating of assemblies = 120 cm² Jun (Vol 54 - Vol 576)
- perforation (below active part) = 84 cm² Jun (Vol 58 - Vol 578)
- perforation (above active part) = 84 cm² Jun (Vol 62 - Vol 572)
Cleaning tank inlet flow rate was imposed to represent the different cooling modes, with a 5 min. interruption in forced flow due to transition from one mode to the other (Fig. 2):

**Boundary conditions**
time < -300 sec     47.20 kg/s,  
-300 < time < 0 sec    0.00 kg/s  
time > 0 sec    5.83 kg/s

Initial tank temperature 57 °C  
Pool temperature 30 °C  
Total fuel assembly power 241 kW
Results

The analysis starts from stabilised conditions in the cooling mode C, with high forced flow through the tank. A 5 min. period was assumed before effective switch-over to the low flow rate of the cooling mode B was established, where flow rate to the tank was zero. In this period natural circulation was established in the tank with upward flow through the assemblies, downward in the tank and back via the perforations and by-pass area due to incorrect seating.

With the low flow rate of cooling mode B the flow rate entering the heated part of the assemblies is continuously decreasing, while the by-pass flow rate is continuously increasing (Figs. 9 and 10). As a consequence, the coolant temperatures start to rise. Due to the rather fast decrease of the flow rate, the saturation temperature is reached at the assembly outlet already at 2830 s (Fig. 8). In spite of that, it is not before 10560 s that significant level decrease (Fig. 6) can be observed in the cleaning tank, which is much later then the phenomenon observed during the incident, where about 2 h 20 min after change-over to cooling mode B the level of the pool rose by ~7 cm in cca. 20 min. At about 11600 s the level finally stabilises at an elevation of 1.6 m. As shown by Fig. 7, already before the level stabilisation fuel temperatures start to increase and by the end of the calculation the maximum temperature reaches 600 °C.

As soon as the saturation temperature at fuel assembly outlet is reached, oscillations in the tank flow rates (Figs. 3, 9 and 11) as well as the system pressure, Fig. 5, can be observed.

Results for the main parameters are presented in Figs. 4.1 to 4.15. Due to the simplified nodalisation the inlet flow rates in Figs. 4.8-9 and 4.10-11 were estimated in the following way:

\[
\text{Inlet flow rate for working assembly} = \frac{[\text{Jun} (54 - 58) - 19*\text{Jun}(58-60)/30]}{11}
\]

\[
\text{Inlet flow rate for follower assembly} = \frac{\text{Jun} (58 - 60)/30}{30}
\]

The sequence of the events can be summarised as follows:

-300 s  47,2 kg/s forced flow

0 s  natural circulation is established via the perforations and by-pass

5,83 kg/s forced flow, heat-up to saturation

2830 s  void at outlet of assemblies

10560 s  vessel liquid level decreases

10730 s  fuel cladding temperature increases

13650 s  end of calculation
Conclusions
Saturated conditions are reached too early in the transient due to rather fast decrease of the fuel assembly low rate. Since elevation heads are correctly calculated in RELAP, it is assumed that this is a consequence of inappropriate friction losses in by-pass areas and/or fuel assemblies. On the other hand, the accumulation time of steam in the upper head is overestimated that may be due to non-physical oscillations caused by transition from low to higher void fractions in the condensation models.
Fig. 3. Cleaning tank coolant outlet flow rate

Fig. 4. Cleaning tank upper plenum and coolant outlet temperatures
Fig. 5. Pressure in the upper volume of the cleaning tank

Fig. 6. Collapsed water level in the cleaning tank
Fig. 7. Max. cladding temperature in assemblies

Fig. 8. Temperatures in upper plenum and at outlet of assemblies
Fig. 9. Inlet flow rate of assemblies and to the heated part

Fig. 10. Bypass flow rates due to incorrect seating and perforations
Fig. 11. Flow rate through the air letdown valve
APPENDIX B/VI.

THERMAL HYDRAULIC CALCULATION OF THE PAKS-2 INCIDENT BY ATHLET
István Trosztel (Hungary, AEKI)

OECD-IAEA Paks Fuel Project
1. Introduction

In order to understand the physical processes taking place in the cleaning tank during the Paks-2 incident, in April/May 2003 a series of calculations were performed by AEKI for different batches of fuel assemblies cleaned in the tank. The thermal-hydraulic analyses were performed by the ATHLET code. In what follows, the “best estimate” calculation results are presented for Batch No. 6.

2. Geometrical model and initial/boundary conditions

The geometrical model included the following elements as represented by Fig. 1:

- Simplified container geometry with lower and upper plena and outlet piping. Inlet piping was only modeled by imposed pump flow rates.
- Container wall, fuel assemblies (FA) including shrouds, support plates modelled as heat structures.
- FAs grouped into 5 parallel channels with 20 axial nodes
- Tank volume outside the assemblies represented by a single channel with 20 axial nodes
- Shroud perforation of working FA at bottom and top (A= 7.634 cm²)

Initial and boundary conditions were selected on the basis of Batch No. 6 data:

- Total decay heat: 241 kW
- 19 follower FAs represented by 3 groups with different power
- 11 working FAs represented by 2 groups with different power
- Axial power profile: from decay heat calculations for Batch No. 6
- Pump flow rate: 170 t/h in cooling mode C and 21 t/h in cooling mode B, with instantaneous switch-over from mode C to B
- Cleaning tank initial temperature: 57°C, uniform
- Pool temperature: 30°C
- Degassing line not modelled explicitly, only constant outflow of 0.0015 kg/s assumed.
- Besides the shroud perforation of the working FAs an additional by-pass cross section due to incorrect seating of 117.5 cm² was assumed.

3. Results

The mass flow rates at FA inlet and in the heated part of the different assemblies are given in Figs. 2 and 3, respectively. In Fig. 2 the mass flow rate at the inlet to the five groups of assemblies (working: AV, follower: FO) can be seen. While the flow into the working assemblies is continuously increasing, the opposite is true for the follower ones. However, looking at the flow rates entering the heated part of the assemblies, i.e. above the elevation of the shroud perforation, (Fig. 3), all the flow rates display similar behaviour, which means that in the case of the working FAs the difference between the two flow rates exits via the shroud perforations. The flow rates of the different assemblies are slightly different due to the different power. The by-pass flow rate is continuously increasing and reaches at 10000 s almost 75 % of the total flow at the FA inlet.
As a consequence, the coolant temperatures start to rise. The heat-up of the first working FA is shown in Fig. 4: the saturation temperature is reached at the outlet after 6000 s. This value varies among the five groups of FAs between 5400 and 6400 s, as it can be seen from Fig. 5. Figure 9 displays the axial variation of coolant temperature in the tank, outside the FAs. (Volume numbering is from top to bottom, i.e. elevations 1 and 20 indicate the top-most and bottom volumes, respectively.)

The void formation under the tank cover (V-UP in Fig. 6) starts at 6500 s and by 8000 s it is completely voided. It takes about a further 1000 s that the level in the tank drops to roughly the lower 1/3 elevation of the tank. This is in good agreement with the phenomenon observed during the incident, where about 2 h 20 min after change-over to cooling mode B the level of the pool rose by \( \sim 7 \) cm in cca. 20 min.

Figures 7 and 8 represent the calculated and measured temperature and flow rate, respectively, at the tank outlet. Since the measured values were noted with 1 h time interval, they miss the information that might have been captured at this point of measurement. The outlet temperature remains practically constant for most of the time, with a significant peak during the 30 minutes of void formation in the tank. The same is true for the flow rate variation.

4. Uncertainties affecting analysis results

There are a number of parameters in the calculations that are known with considerable uncertainty. A review of their effect and different parametric calculations performed indicate the importance of the following parameters:

- **Power of fuel assemblies.** The power of each of the fuel bundles is known with fairly good accuracy. Although fuel power is one of the most important factors in the heat-up phase until saturation, the uncertainties related with it are not affecting the heat-up time considerably.

- **Axial power distribution.** This is also known with fair accuracy. The assumed axial distribution has no impact on the FA coolant outlet temperature, it only slightly affects the axial temperature distribution.

- **Tank heat losses.** Since the tank is submerged in the 35°C pool, some heat loss can be expected in spite of the fact that it was designed with a double wall, with air in between. However, in the heat-up phase to saturation heat loss only represents a few percent of the total energy balance.

- **Hydrostatic head terms.** These terms are dominating the tank behaviour, their correct representation is important. The analysis results are affected by the number of axial nodes applied to represent the FAs and the tank. Parametric studies performed with the ATHLET code indicate that changing the number of nodes from 10 to 20 resulted in enhanced heat-up: it is not expected that finer subdivision would influence the results any more.

- **FA loss coefficients.** Considerable uncertainties can be expected in the loss coefficients of the FAs. Data are known for nominal operating conditions, i.e. for high Re-numbers and they include both frictional and form loss terms. The latter are relatively insensitive to Re-number, but have a tendency to increase as Re-number decreases. The frictional term increases much more with decreasing Re-numbers, especially, in the laminar flow region, as it is the case in the FAs, and largely outweigh form losses. The codes should be able to correctly predict these loss terms.
• By-pass flow loss coefficients. They are of high importance, since the pressure drop across the by-pass areas is a major term in the momentum equation: due to the low flow velocities in the FAs and the tank it practically balances the hydrostatic pressure drop terms. The “inherently present” by-pass area, i.e. the holes in the FA shroud are of great importance: this was evidenced by parametric cases with different numbers of working assemblies in the tank that led to large differences in heat-up times to saturation. On the contrary, the loss coefficients of the by-pass areas created by not perfectly seated FAs in the lower support plate have a negligible effect, which can be explained by a “saturation” effect shown by parametric studies: the time to saturation is mainly defined by the perforation by-pass, additional by-pass areas have only very little effect.

The nominal value of the loss coefficient for the shroud perforation is 2.7, but it was evaluated for the nominal operating conditions and (probably) takes into account the situation in the reactor core, where the FAs are positioned with a very tight pitch with respect to the cleaning tank. In order to justify the loss coefficient used in the analyses measurements were performed by AEKI in a simplified geometry, representative of the flow conditions on both sides of the holes, with a wall thickness of 6 mm between the two volumes. The diameter of the hole was 9 mm and cases with one and two holes across the wall were investigated. The Re-number (based on the conditions in the hole) was varied between 1000 and 7600, the pressure drop across the holes changing between 10 and 900 Pa. Two series were carried out with abrupt area change at the entrance to the hole, while the series with a single hole was repeated with the hole entrance machined in 45° on a distance of 1 mm.

The results of the measurements are summarised in Fig. 10. A sharp increase of the loss coefficient can be observed in the laminar flow region. The actual Re-number in the shroud perforation in the cooling Mode B is around 2500. At this value of Re-number the loss coefficient with abrupt entrance is around 2.4-2.6 that support the application of the nominal value of 2.7.

5. Conclusions

The calculation indicates that the mass flow rate along the heated part of the FAs is decreasing with time, while the by-pass flow rate is continuously increasing and reaches at 10000 s almost 75 % of the total flow at the FA inlet.

As a consequence, the coolant temperatures start to rise and the saturation temperature is reached at the outlet of the FAs between 5400 and 6400 s, depending on FA power. The void formation under the tank cover starts at 6500 s and by 8000 s it is completely voided. It takes about a further 1000 s that the level in the tank drops to roughly the lower 1/3 elevation of the tank. This is in good agreement with the phenomenon observed during the incident, where about 2 h 20 min after change-over to cooling mode B the level of the pool rose by ~7 cm in cca. 20 min. The phase with the heat-up of the fuel rods could not be calculated, since the analysis aborted due to the strong flow and void oscillations.
Fig. 1 ATHLET nodalisation of the cleaning tank
Fig. 2  Mass flow into the assemblies

Fig. 3  Mass flow into the heated part of the assemblies
Fig. 4  Coolant liquid temperature

Fig. 5  Coolant liquid temperature at the outlet of assemblies
Fig. 6 Void in the assemblies and in the tank

Fig. 7 Measured and calculated outlet temperature
Fig. 8 Measured and calculated flow rate

Fig. 9 Coolant temperature in the tank
Fig. 10  Measured loss coefficient of FA shroud perforation
APPENDIX B/VII.

FUEL BEHAVIOUR
CALCULATION OF THE PAKS-2 INCIDENT
Attila Molnár (Hungary, AEKI)

OECD-IAEA Paks Fuel Project
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1. Fuel behaviour calculations of the Paks-2 incident with the FRAP-T6 code

On 10 April 2003 severe damage of fuel assemblies took place during an incident at Unit 2 of Paks Nuclear Power Plant in Hungary. The assemblies were being cleaned in a special tank below the water level of the spent fuel storage pool in order to remove crud buildup. That afternoon, the chemical cleaning of assemblies was completed and the fuel rods were being cooled by circulation of storage pool water. The first sign of fuel failure was the detection of some fission gases released from the cleaning tank during that evening. The cleaning tank cover locks were released after midnight and this operation was followed by a sudden increase in activity concentrations. The visual inspection revealed that all 30 fuel assemblies were severely damaged. The first evaluation of the event showed that the severe fuel damage happened due to inadequate coolant circulation within the cleaning tank.

We simulated the incident with the FRAP-T6 code in order to evaluate the level of damage of fuel rods. The code simulates the thermal and mechanical behavior of one piece of fuel rod considering the heat transmission of the given subchannel. The model was a one dimensional, we set the coolant flow rate to constant and we did not take into account to the radial heat loss and we did not model the wall of the fuel assemblies. According to the thermohydraulic calculations (ATHLET, RELAP) in thermohydraulic phase, all coolant flowed out at the bottom of the fuel assemblies, not cooling the heated section. In calculations primarily we wanted to examine the fuel rod behavior in low heat transport case therefore we did not take into account the coolant oscillation – observed in thermohydraulic calculations – and the temperature distribution in the tank.

In the FRAP-T6 calculations, we set the coolant circulation to constant (1.2 percent of the nominal value) in the subchannel and we did not take into account other heat transfer method (e.g. radiation). The enthalpy of the coolant at the entry was 167 kJ/kg.

Operational data for Unit 2 and cycles 16-19 were provided by the Paks NPP. They included:
- load map of archives for cycle 16 describing the burnup distribution at the beginning of cycle,
- power distribution and histories for cycles 16-19,
- refueling matrices between 16-19 cycles.

The burnup distribution in 10 axial nodes for each assembly was calculated using the GLOBUSKA module of the KARATE program system. The decay heat of each node of each assembly was calculated using power distribution and considering the storage time between reactor shutdown and the incident. The ORIGEN and TIBSO codes were applied for this purpose.

The fuel assemblies with similar power histories and of the same type (follower or working) were grouped into 6 groups and only one representative assembly was calculated for each group. The burnup dependent parameters of each assembly were determined using the reactorphysics data and fresh fuel parameters with the TRANSURANUS code. We got the average linear power, the
burnup, the pressure in the fuel rod and the axial distribution from the reactorphysics calculations. The calculations were made in the 10 evenly spaced axial part of the fuel rod.

2. Ballooning of the cladding

According to the analysis, after the decrease of the coolant stream the water boils in the subchannel and vapor formation starts. After the temperature reached the saturation point, the vapor formation finishes within 300-500 second at the whole length of the heated part of the fuel rod. The vapor stream is not able to cool the fuel rods therefore the rods starts to overheat.

Figure 1. The temperature and pressure of the most accused fuel rod

We made calculations for all group, the results are shown in the Table 1. The figure shows the temperature distribution of cladding of the most accused fuel rod (average linear power was 37.20 W/m, burnup was 26.7 MWd/kgU and inside pressure was 7.43 Bar). It is noticeable that the temperature of upper part of cladding exceeds 800 C° within 130 minutes. In this case, the fuel rods balloon up and burst, and accumulated fission gases release into the coolant. We can also see the inside pressure of fuel rods. The decrease of the curve shows the bursting of fuel rod when inside and outside temperature gets equal. In our calculations the bursting of fuel rod occurred about 2.7 hours after vapor formation. In this time the fission gases cause activity increase in the coolant and in the reactor hall. It is likely that this process was detected by activity detectors after 21.30.
The number of bursted fuel rods depends on burnup of fuel rods and temperature distribution of vessel. The ballooning and bursting of fuel rods with lower burnup and/or lower temperature occurred later, but based on our calculations, there were some fuel rods that did not bursted in this phase of incident. The characteristic of ballooning is known from previous AEKI experiments: the injury of cladding is only local, therefore at this time the fuel assemblies were definitely treatable/ manageable.

3. The oxidation and cracking of fuel assemblies

Most of the fuel assemblies were cracked by the oxidation and embrittlement induced by the zirconium-water steam reaction. After bursting, the temperature of fuel rods increased and the maximum cladding temperature exceeded 1200 °C. Support this temperature there are two observations:
- the steel part did not melt thus the temperature did not reach 1500 °C
- according to video records some parts of the zirconium cladding and wall of fuel assemblies were totally oxidized (this needs more hours of water steam – above 1000 °C – oxidation)

The thickness of zirconium-oxide layer on cladding grows quickly because of the intensive oxidation. Its locale value is 300 µm, after 5-6 hour oxidation. We did not model the wall of fuel assemblies, but it is evident that due to two-sided zirconium-water vapor reaction, its oxidation level was more intensive therefore it lost its mechanical strength totally. The cleaning tank cover locks were released after 2.20 and through the quenching and the mechanical impact most part of rigid but still hermetic fuel rods were cracked.

Figure 2. The rate of Hydrogen production
4. Hydrogen formation

Based on the amount of oxidized zirconium the hydrogen production was about 3 kg. This amount is about 70 m$^3$ (at a pressure of 2 bar and at temperature of 727 °C), which could not remain in the tank. The hydrogen was able to clear off through the gas transfer pipe: because the small density the rate of flow was high (about 100 m/s), but considering the parameters of the pipe and hydrogen, 0.3 bar of pressure difference was enough to create such flow conditions.

5. Summary of main results

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<thead>
<tr>
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<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Time of bursting (s)</td>
<td>12010</td>
<td>10100</td>
<td>8730</td>
<td>11130</td>
<td>9930</td>
<td>9850</td>
</tr>
<tr>
<td>Max. temperature of the cladding (°C)</td>
<td>1106.57</td>
<td>1155.60</td>
<td>1062.76</td>
<td>1151.83</td>
<td>1049.67</td>
<td>1088.39</td>
</tr>
<tr>
<td>Max. temperature of the fuel (°C)</td>
<td>1107.59</td>
<td>1156.74</td>
<td>1065.62</td>
<td>1154.11</td>
<td>1051.12</td>
<td>1089.78</td>
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<tr>
<td>Max. fuel rod internal pressure (bar)</td>
<td>14.21</td>
<td>15.42</td>
<td>12.24</td>
<td>12.74</td>
<td>14.73</td>
<td>14.72</td>
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<td>Max. cladding oxidation of a fuel rod (%)</td>
<td>10.13</td>
<td>10.11</td>
<td>10.08</td>
<td>10.06</td>
<td>10.13</td>
<td>10.13</td>
</tr>
</tbody>
</table>

Table 1. The main results of the calculations
APPENDIX B/VIII.

ACTIVITY RELEASE
CALCULATION OF THE PAKS-2 INCIDENT
Emese Szabó (Hungary, AEKI)

OECD-IAEA Paks Fuel Project
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1. **CALCULATION OF ACTIVITY RELEASE DURING THE PAKS-2 INCIDENT**

1.1. **Calculation method description**

In recent years there have been a number of theoretical and experimental studies dedicated to the analysis of the fission product release mechanisms during a loss of coolant accident. Recommendation was made by a group of a European specialist for the estimate of the activity release in the course of LOCA. This method has been used in the most of European countries for estimate of the activity release from damaged fuel rods in the safety analysis.

The calculated method estimates the release from the fuel rods in the percentage of isotope inventory. The method makes a distinction between release from the gap and fuel. First the gap inventory releases from the damaged fuel assemblies and then one part of the radioactive isotopes (that are located in the pellet) with different mechanisms, like fragmentation of pellet or leaching.

Parameters of the best-estimate method were derived from processing of different experimental data.

In the safety analysis the conservative values are used in general, so upper limit can be obtained for the release.

1.1.1. **Calculation of F + G %**

Based on the experimental data the following is proposed for the best estimate fraction of fuel inventory released as the result of a successfully terminated LOCA.

The total release from the fuel is:

\[(F+G)\% \text{ of core inventory},\]

where:

- F = the component arising from the fragmentation of fuel and
- G = the component arising from the gap inventory.

Both components are a function of the radioactive half-life and are as follows:

\[F(\%) = F_0 \times \lambda^{bf}\]
\[G(\%) = G_0 \times \lambda^{-0.5}\]

where:

- \(\lambda\) is the radioactive decay constant.

\(G_{max}\) or \(F_{max}\) values were used instead of the calculated G or F values when \(G > G_{max}\) or \(F > F_{max}\).
The equations are applied until the maximum values of $F_{\text{max}}$ and $G_{\text{max}}$ respectively are obtained for the long-lived nuclides. For nuclides of longer half-life, the stable nuclide values $F_{\text{max}}$ and $G_{\text{max}}$ are assumed. The values for each of the above parameters for each class of nuclide are shown in Table 1 and 2 for both the best estimate and conservative cases:

<table>
<thead>
<tr>
<th>Nuclide class</th>
<th>Best-estimate</th>
<th>Conservative</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$G_0$ %</td>
<td>$G_{\text{max}}$ %</td>
</tr>
<tr>
<td>Noble gases</td>
<td>2.50E-04</td>
<td>1.0</td>
</tr>
<tr>
<td>Volatiles</td>
<td>2.50E-04</td>
<td>1.0</td>
</tr>
</tbody>
</table>

Table 1: Parameters for the gap inventory release

<table>
<thead>
<tr>
<th>Nuclide class</th>
<th>Best-estimate</th>
<th>Conservative</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$F_0$ %</td>
<td>$F_{\text{max}}$ %</td>
</tr>
<tr>
<td>Noble gases</td>
<td>4.00E-02</td>
<td>-0.29</td>
</tr>
<tr>
<td>Volatiles</td>
<td>3.80E-02</td>
<td>-0.17</td>
</tr>
</tbody>
</table>

Table 2: Parameter for the fragmented fuel release

Apart from the noble gases 10 % of the total release is taken to be released in the dry phase and 90 % is released after the rupture is recovered. All the noble gas release is taken to occur in the dry phase.

In the description of this method it is not clear that which isotopes are volatiles (because it depends on conditions of the accident, mainly on the temperatures). In our work release of the all isotopes except the noble gases were calculated by values of volatiles.

1.2. Calculation of activity release of isotopes for assemblies

These calculations consisted of 4 steps. First of all we took the isotope inventory of assemblies with 13 days after the incident, after these data were multiplied by the $F+G$ % and were divided by 100. Since these calculated values were given in Curie these were changed to Bq (these data were multiplied by 3.7E+10).

1.3. Results of the calculation

For the evaluation of results the measured and the calculated data were compared. Time of failure of every assembly group was taken from the results of FRAPTRAN calculations.

1.3.1. Release through the chimney

There was significant release through the chimney during the incident. Among the released noble gases the $^{133}\text{Xe}$ was the most significant.

So the next figure represents the activity values of $^{133}\text{Xe}$ that was recorded each ten minutes.
1.3.2. Comparison of activities of main isotopes

All isotopes were considered as volatile material in our calculations. The measured activity concentrations in the coolant were integrated for the first two weeks after the incident. (PAKS2 database / ACTIVITY / Measured).

See in the calculations of activity releases were performed for every assembly. The fuel assemblies with similar power histories and of the same type (follower or working) were grouped into 6 groups. In the next table and figure it could be seen that good agreement was found between calculated and measured activities.

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Corrected total release during the first two weeks after the incident [Bq]</th>
<th>Best-estimate values [Bq]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ru-103</td>
<td>6.32E+11</td>
<td>1.47E+15</td>
</tr>
<tr>
<td>Ru-106</td>
<td>8.69E+12</td>
<td>5.50E+14</td>
</tr>
<tr>
<td>Cs-134</td>
<td>4.21E+13</td>
<td>1.06E+14</td>
</tr>
<tr>
<td>Cs-136</td>
<td>2.44E+13</td>
<td>1.16E+13</td>
</tr>
<tr>
<td>Cs-137</td>
<td>3.84E+13</td>
<td>1.44E+14</td>
</tr>
<tr>
<td>Ba-140</td>
<td>1.82E+14</td>
<td>8.68E+14</td>
</tr>
<tr>
<td>Ce-141</td>
<td>6.39E+13</td>
<td>1.67E+15</td>
</tr>
<tr>
<td>Ce-144</td>
<td>7.19E+13</td>
<td>2.08E+15</td>
</tr>
<tr>
<td>Np-239</td>
<td>3.54E+12</td>
<td>2.32E+14</td>
</tr>
<tr>
<td>I-131</td>
<td>5.89E+14</td>
<td>2.70E+14</td>
</tr>
<tr>
<td>Te-132</td>
<td>2.94E+13</td>
<td>5.63E+13</td>
</tr>
<tr>
<td>Xe-133</td>
<td>6.61E+14</td>
<td>1.20E+15</td>
</tr>
<tr>
<td>Cm-242</td>
<td>9.26E+08</td>
<td>1.49E+13</td>
</tr>
<tr>
<td>Cm-244</td>
<td>3.57E+07</td>
<td>1.05E+12</td>
</tr>
<tr>
<td>Pu-238</td>
<td>2.34E+07</td>
<td>1.47E+12</td>
</tr>
</tbody>
</table>

Table 3: Calculated release rates of different isotopes

Figure 1: History of $^{133}$Xe activity concentration through the chimney

Figure 2: Comparison of measured and activity data
1.4. Conclusion

Next figures represent the calculated and measured total activity release at burst as a function of time.

In case of volatile isotopes good agreement was found between the measured activity release and results of calculations (Fig.3-6). The most significant difference appeared for non-volatile elements, for example Cm-242 or Ce-141 (Fig.7-8).
APPENDIX B/IX.

ONE-PHASE 3D CFD INVESTIGATIONS ON THE DEVELOPMENT OF THE SERIOUS INCIDENT IN PAKS NPP, APRIL OF 2003

Gábor Légrádi, Ildikó Boros, Attila Aszódi (Hungary, BME)

OECD-IAEA Paks Fuel Project
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1. INTRODUCTION

On 10-11\textsuperscript{th} of April, 2003 a serious incident (INES-3 event) caused heavy damage of 30 fuel assemblies in a cleaning tank which was installed into the refueling pit of the 2\textsuperscript{nd} unit of the Paks NPP. The incident was investigated by several scientific institutes and with different analytical and numerical methods and tools. Among other methods, the early stage of the incident was investigated in the BME NTI with 3D CFD calculations in 2003 [1].

This earlier CFD model was developed with the CFX-5.5.1 code and it was capable for investigating the early stage of the incident between 16:40 and 19:00, 10\textsuperscript{th} of April, 2003 in a qualitative manner – concerning to the results. In this stage of the incident one-phase-flow was in progress in the cleaning tank and very stable temperature stratification was developed. Due to this temperature stratification the flow was blocked inside the fuel assemblies and the coolant was heated up to the saturation temperature in the top parts of them. The 3D CFD computational results clearly confirmed this phenomenon.

In 2006, a decision was made in the BME NTI to perform again 3D CFD calculations on this early stage of the incident in the framework of the OECD-IAEA Paks-2 Fuel Project. The new model was developed with the CFX-10.0 and the ICEM CFD-10.0 codes. Much more robust meshing, more detailed geometry and the best known real parameters were built into this model. With this model, the early stage of the incident was investigated till the temperature reached the saturation temperature.

By modifying the model’s geometry the effects of displacements of one or two working fuel assemblies were also investigated.
2. SUMMARY OF THE EARLIER CFD INVESTIGATION

In the same year when the serious incident occurred, the first 3D CFD analysis of the incident was performed in the BME NTI [1]. In this section, this earlier model, calculations and results are summarized briefly.

Although the 6th batch contained 11 working and 19 follower assemblies, this earlier developed model contained 30 working assemblies, since modeling working and follower assemblies together resulted in heavy numerical instabilities at that time. With regard to the limited computer capacities for the meshing, the modeled assemblies contained 6 perforations of 13 mm in diameter per each at the bottom part instead of the real 12 perforations.

With this model the one-phase flow field was calculated in the “B” operational mode which started at 16:40, 10th of April, 2003. With operating submersible pump of 20 t/h mass flow and without taking into account the heat power of the assemblies the 3D results showed that even without the effect of temperature stratification, the by-pass flow through the bottom perforations of the working fuel assemblies would have been 12%.

Then a 4000 s long transient calculation was performed with an estimation of 350 kW of the 6th batch’s heat source was specified in the model such a way that a uniform volume heat source was set inside the fuel assemblies. The later burn-up calculations showed that at the time of the incident the decay heat power of the 6th batch’s fuel assemblies was 241 kW.

According to the results of this calculation, the coolant warmed up in the fuel assemblies and it filled up the upper part of the tank. As the coolant became warmer inside and among the fuel assemblies, the up-flow of the cold water inside the assemblies became more difficult. Therefore, the ratio of the by-pass flow through the perforations increased. The results showed that the by-pass rate reached 71% and the maximum temperature increased from 36°C up to 81°C after 4000 seconds. Then, the run was become instable and divergent so it had to be stopped. But, by extrapolating the results, the authors stated that based on these results, the temperature might reach the saturation temperature within two hours without any displacements of the fuel assemblies. This result was in a very good agreement with the experienced process of the incident.

Another very important result was that the outlet temperature of the cooling tank operated in the “B” operational cooling mode was much lower than the maximum temperature inside the cleaning tank. At 4000 seconds the outlet temperature was only 8°C higher than it was at the inlet, while the temperature difference inside the cleaning tank was 50°C due to the temperature stratification.
3. THE RECENT 3D MODELS OF THE CLEANING TANK

In the BME NTI three new CFD models of the cleaning tank were developed with the ANSYS ICEM CFD-10.0 code in the end of 2006 and the beginning of 2007. All models contained 11 working and 19 follower fuel assemblies. The difference between the models was the number of displaced working fuel assemblies. Table 1 shows the alphanumeric of the different models.

Table 1. Alphanumeric of the different CFX models

<table>
<thead>
<tr>
<th>Model</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>W11_F19_DIS0</td>
<td>11 working, 19 follower assemblies, no displaced assembly</td>
</tr>
<tr>
<td>W11_F19_DIS1</td>
<td>11 working, 19 follower assemblies, 1 displaced assembly</td>
</tr>
<tr>
<td>W11_F19_DIS2</td>
<td>11 working, 19 follower assemblies, 2 displaced assemblies</td>
</tr>
</tbody>
</table>

3.1. Geometry of the models

In the following section, the main parts of the models are shown on Fig. 1 and Fig. 2 which are perspective views of the geometry. The models were developed on the basis of the IAEA-OECD Paks-2 Fuel Project’s database and our earlier works [1], [2].

*Fig. 1 The top part of the cleaning tank’s ANSYS ICEM CFD-10.0 geometry model*
The models contained the following components:

— The inner wall of the cleaning tank and the cover.
— The bottom elliptical part of the cleaning tank.
— The four inlet chunks and diverter baffles above them inside the elliptical bottom part.
— The bottom positioning and supporting plate with the 30 seats through which the coolant entered the fuel assemblies.
— Upper positioning plate.
— The two outlet chunks.
— 11 working assemblies. The assemblies were modeled as hexagonal prisms with a coaxial cylinder cut out from it. The cylinders’ cross section was equal with the flow cross section. The 12 perforations on the bottom part were replaced by 6 perforations per each assembly. The diameter of these perforations was 13 mm. The 12 perforations on the top part were replaced by 2 perforations. The leg parts of the working assemblies were modeled in a more detailed manner.
— 19 follower assemblies. The assemblies were modeled as hexagonal prisms with a cylinder cut out from it. The cylinder’s cross section was equal with the flow cross section.
— In the W11_F19_DIS1 and W11_F19_DIS2 modified models displacements of working fuel assemblies were taken into account such a way that apertures have been opened at
the corresponding assemblies’ very bottom end. The cross section of the apertures is 2030 \( \text{mm}^2 \) corresponding to [4].

The models’ origin of co-ordinate system was defined in the center point of the bottom surface of the bottom positioning and supporting plate. The main key dimensions of the geometry model are presented in Fig. 3.

![Fig. 3. Main axial dimensions of the ANSYS ICEM CFD-10.0 geometry model](image)

3.2. Meshing of the model

The model was meshed with combined unstructured tetrahedral mesh with hexahedral core and extruded pyramid mesh. The meshing strategy was the following (for the model dimensions see Fig. 3):

---

The upper part of the cleaning tank above the upper positioning plate (\( z=2.715 \text{ m} \)) and the fuel assembly head parts above the active parts (\( z=2.923 \text{ m} \)) were meshed with unstructured tetrahedral mesh with hexahedral core. It means that near to the complex surfaces the mesh generator made unstructured tetrahedral mesh, but far from the surfaces it was automatically replaced by hexahedral core elements.
The active part of the fuel assemblies and the middle part of the cleaning tank was meshed with extruded prism mesh. We call “middle part” the region of the cleaning tank bellow the level of the upper positioning plate (z=2.715 m) and above the fuel assemblies’ active parts’ bottom level (z=0.523 m). The triangle surface mesh of the tetrahedral mesh was extruded in the axial direction, this way developing a continuous mesh.

The bottom region of the cleaning tank under the active parts of the fuel assemblies (z=0.523 m) and the elliptical bottom part were meshed with unstructured tetrahedral mesh. The unstructured tetrahedral mesh was developed continuously from the triangle surface mesh in the bottom plane of the prismatic mesh.

The models contained 4.1 million control volumes. Table 2 shows the parameters which were used during the mesh generation.

Table 2. Maximum edge length values in different parts of the models

<table>
<thead>
<tr>
<th>Description</th>
<th>Edge Length (m)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Maximum global edge length</td>
<td>0.03125</td>
</tr>
<tr>
<td>Fuel assembly walls</td>
<td>0.015625</td>
</tr>
<tr>
<td>Top plates of the fuel assemblies</td>
<td>0.015625</td>
</tr>
<tr>
<td>Top perforations of the working fuel assemblies</td>
<td>0.0078125</td>
</tr>
<tr>
<td>Bottom perforations of the working assemblies</td>
<td>0.002</td>
</tr>
<tr>
<td>Inlet chunks’ walls, inlet face</td>
<td>0.015625</td>
</tr>
<tr>
<td>Outlet chunks’ walls, outlet face</td>
<td>0.015625</td>
</tr>
<tr>
<td>Holes on the bottom positioning plates’ seats</td>
<td>0.015625</td>
</tr>
<tr>
<td>Horizontal faces of the diverter baffles</td>
<td>0.015625</td>
</tr>
<tr>
<td>Holes on the diverter baffles</td>
<td>0.0078125</td>
</tr>
<tr>
<td>Volume of the displacements’ holes</td>
<td>0.0078125</td>
</tr>
<tr>
<td>Inner face of the displacements’ holes</td>
<td>0.004</td>
</tr>
</tbody>
</table>
3.3. Boundary conditions, parameters of the models

In this chapter the different parameters and the boundary conditions are summarized:

— For all calculations the laminar flow model was used. The buoyancy was taken into account with the Boussinesq-approximation.
— All walls were modeled as free-slip walls.
— For the most of the water parameters, constant literature data for 50°C temperature were used. For setting the volumetric thermal expansivity and the dynamic viscosity functions two polynomial functions were used:

\[
\text{HotCo} = 0.00004958372 \, [\text{K}^{-1}] + 0.00000860075 \times T_c \times 1 \, [\text{K}^{-2}]
- 1.42168 \times 10^{-8} \times T_c \times T_c \times 1 \, [\text{K}^{-3}]
\]

for the beta expansivity, and

\[
\text{DinVisz} = (1497.25 - 29.4225 \times T_c \times 1 \, [\text{K}^{-1}] + 0.2419 \times T_c \times T_c \times 1 \, [\text{K}^{-2}]
- 6.95 \times 10^{-4} \times T_c \times T_c \times T_c \times 1 \, [\text{K}^{-3}]) \times 0.000001 \,[\text{kg m}^{-1} \text{s}^{-1}]
\]

for the dynamic viscosity, where \( T_c \) was the temperature in Celsius (see Fig. 4).

![Figure 4. The Beta thermal expansivity (HotCo) and the dynamic viscosity (DinVisz) functions of the CFX models (plotted with CFX-Pre)](image)

— The fuel assemblies were divided into six groups by its decay heat power profile. The grouping and the decay heat power values of the different groups’ assemblies can be seen in Table 3.
Table 3. Sum decay heat power of the different groups’ assemblies

<table>
<thead>
<tr>
<th>Group Description</th>
<th>Power (W)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1st group (assembly 1-6)</td>
<td>7,159 W</td>
</tr>
<tr>
<td>2nd group (assembly 7-11)</td>
<td>9,006 W</td>
</tr>
<tr>
<td>3rd group (assembly 12)</td>
<td>8,154 W</td>
</tr>
<tr>
<td>4th group (assembly 13-18)</td>
<td>6,694 W</td>
</tr>
<tr>
<td>5th group (assembly 19-24)</td>
<td>8,837 W</td>
</tr>
<tr>
<td>6th group (assembly 25-30)</td>
<td>8,719 W</td>
</tr>
<tr>
<td><strong>Total power</strong></td>
<td><strong>241,638 W</strong></td>
</tr>
</tbody>
</table>

The decay heat power density profile was given in the model as point slopes function of the Z axial level. The data set for the heat power values in different nodes for different groups was given in the IAEA-OECD Paks-2 Fuel project’s database. Based on these data the decay power density was defined in the fuel assemblies (see Fig. 5).

---

Fig. 5. Power density in the fuel assemblies of the different groups
(ReMiCsop refers to the i\textsuperscript{th} group, plotted with CFX-Pre)
Different quadratic resistance values were set for the working and follower fuel assemblies. Based on literature data [3] 1363 $kg/m^4$ volumetric quadratic resistance inside the working and 2050 $kg/m^4$ inside the follower fuel assemblies were set. These values are correspond to $\xi = 9.4$ and $\xi = 14.2$ quadratic resistance factors.

The inlet parameters were constant static pressure and 36 °C inlet temperature. The outlet setting was constant mass flow rate of 5.555 $kg/s$ with uniform velocity profile.

The initial condition was uniform 36 °C temperature.

All calculations were run as transient calculations of 8,000 second. The time-step was set to be 2.5 s. The sensitivity for the time stepping was investigated.
4. CALCULATIONAL RESULTS

The calculations for the different model versions gave qualitatively very similar results. Therefore, most of the results’ main characteristics are summarized together in the following bullets and the most important parameters are summarized in Fig. 6 – Fig. 11:

![Graph of mass flow rate and by-pass rate through FAs](image)

Fig. 6. By-pass rate through the working FAs’ bottom perforations and the displacement orifices and mass flow rate through the active parts of the FAs for the three calculations

1. At the beginning of the transient, the warming up in the stagnating coolant inside the fuel assemblies drove the flow with a higher mass flow than the cooling system’s nominal mass flow of $5.555 \text{ kg/s}$. It means that the by-pass flow had a negative value (see Fig. 6). After 450-500 s, the mass flow through the FAs’ active parts decreased below $5.555 \text{ kg/s}$ and the by-pass turned to have a positive value.

2. The temperature stratification developed from the very beginning. After 1200 seconds such temperature fields developed that it kept its characteristics for the further part of the transients: The outlet temperature values froze at about 41-44 °C (5-8°C higher than the inlet temperature), and the temperature increased with similar rates in time at different levels in the flow domain above the level of the working FAs’ bottom perforations (see Fig. 7 – Fig. 9).

3. At 1800-2400 s, the sum by-pass flow become higher than the net flow through the FAs’ active parts and increased further up to 70% in case of the W11_F19_DIS0 calculation, up to 75% in case of the W11_F19_DIS1 calculation and up to 81% in case of the W11_F19_DIS2 calculation after 8000 s (see Fig. 6).
Fig. 7. Temperature in the cleaning tank’s centerline at different levels for the W11_F19_DIS0 calculation (see Fig. 3 for the axial dimensions)

Fig. 8. Temperature in the cleaning tank’s centerline at different levels for the W11_F19_DIS1 calculation (see Fig. 3 for the axial dimensions)
Fig. 9. Temperature in the cleaning tank’s centerline at different levels for the W11_F19_DIS2 calculation (see Fig. 3 for the axial dimensions)

(4) The average temperatures inside the fuel assemblies of groups 1, 2, 5 and 6 – working fuel assemblies and follower assemblies of the safety absorber elements – was very similar, and also the average temperatures inside the fuel assemblies of groups 3 and 4 – follower assemblies of the controller absorber elements – was also similar but it had lower values. On the other hand, the maximum temperature values developed in the follower FAs of groups 3 and 4 (see Fig. 10). Fig. 5 shows clearly the reason of this phenomenon: These assemblies had such power profile that the heat production was quite low at the bottom part and the maximum power developed in the upper parts of them.

(5) The calculations show that the maximum temperature did not approach the saturation temperature even after 8000 s. For the different calculations the increasing of the maximum temperature had a rate of about 3, 4 and 4,6 °C/1000 s (see Fig. 12). With these rates the saturation temperature (~120 °C) would have reached after 22900 s (6 hours 22 minutes), 17400 s (4 hours 50 minutes) and 14700 s (4 hours 5 minutes) for the W11_F19_DIS0-DIS1-DIS2 calculations respectively.
Fig. 10. Average temperature values in the different assembly groups’ active parts and the maximum temperature values in the FAs’ active parts for the W11_F19_DIS2 calculation (WFA: working fuel assembly, FFA: follower fuel assembly)

Evaluating the results’ characteristics on the basis of the above statements and figures the following conclusions may be deducted:

— Bullet (1) refers to the fact that the decay heat produced in the cleaning tank was not removed. This is true independently from the fact whether there were FA displacements or not. The $\Delta P$ removed heat power may be calculated by the following way:

$$\Delta P = \dot{m}_{\text{coolant}} \cdot c_{\text{water}} \cdot \Delta T = 5.555 \left[ \frac{\text{kg}}{\text{s}} \right] \cdot 4181.7 \left[ \frac{\text{J}}{\text{kg} \cdot ^\circ \text{C}} \right] \cdot \Delta T = 23 \left[ \frac{\text{KW}}{^\circ \text{C}} \right] \cdot \Delta T,$$
where \( \dot{m}_{\text{coolant}} \) is the mass flow of the cooling system, \( c_{\text{water}} \) is the specific heat capacity of the coolant, \( \Delta T \) is the difference between inlet and outlet temperatures. In case the \( \Delta T \) value had been a lower value than 10.5°C, the sum decay heat power of 241 kW would have been not removed. As it is mentioned in bullet (1), the \( \Delta T \) values were 5-8°C. This result is also in a very good agreement with the earlier investigations presented in section 2, where this value was given to be 8°C.

The results summarized in bullets (1) and (3) well correspond to the earlier calculation – 71% by-pass after 4000 s.

According to bullet (5), the warming up was not given to be as rapid that the maximum temperature could have reached the saturation temperature within 2 hours, as it was come from the earlier calculations. But this difference is evident, since the W11_F19 calculations were performed with 11 working fuel assemblies and 241 kW sum decay heat power taken into account, instead of the 350 kW power of the earlier calculation and 30 working fuel assemblies. On the other hand, it has to be emphasized that the valid initial temperature of the cleaning tank at 16:40, 10\(^{th}\) of April, 2003 is unknown. In some of the different analyses concerning to the incident, the initial temperature was set to be 57°C [1]. Fig. 7 – Fig. 9 show clearly and bullet (1) also refers to that independently from the initial temperature, a similar stratified temperature field could be developed after a while. It is likely that in case the initial temperature had been higher, the saturation temperature would have been reached earlier.

From the uncertainties mentioned above and the results evaluated in bullet (5), the most important conclusion may be drawn: Even with or without any fuel assembly displacements the temperature stratification could have been developed and blocked the flow through the fuel assemblies and the saturation temperature could have been reached after enough long period of time.

On the other hand, it is known that the maximum temperature reached the saturation temperature within 8400 s during the incident. Our results refer to that for reaching the saturation within 8400 s such initial temperature would needed for the W11_F19_DIS0 and W11_F19_DIS1 calculations that were unlikely in the cleaning tank after the cooling down program at the end of the “C” operational mode. Therefore, our calculations make it likely that beside the working fuel assembly’s bottom perforations more by-pass area was present in the bottom part of the cleaning tank.
Fig. 11. Maximum temperature values, whole flow domain volume average temperatures and mass flow weighted temperature averages for the outlet for the three calculations

Fig. 12. Fitted trend lines of the maximum temperature results from 1200 s for the three calculations
5. SUMMARY

In this report, a very sophisticated CFD model of the cleaning tank is presented. The results were compared with the results of an earlier CFD calculation. Both works were performed in the BME NTI in 2003 and 2006/2007. From this brand new work, very interesting and important conclusions were drawn.

Our new CFD model has three versions: in one of them all fuel assemblies are taken into account as placed into their seats precisely (W11_F19_DIS0), the other two versions contain one and two displaced working fuel assemblies (W11_F19_DIS1, W11_F19_DIS2). The starting temperature of the transients when the cooling system was set from “C” operational mode to “B” (16:40, 10th of April, 2003) is unknown. Also the coolant’s inlet temperature of the cleaning tank which was valid during the development of the incident is unknown. Therefore the initial temperature condition and the inlet temperature were set to be the same 36°C value.

On the basis of the three calculations’ results, the following important conclusions may be drawn:

— The temperature stratification developed from the very beginning. After 1200 seconds such temperature fields developed that it kept its characteristics for the further part of the transients: The outlet temperatures froze at about 41-44 °C temperature values which were 5-8°C higher than the inlet temperature (see Fig. 11). It means that the decay heat power was not removed from the cleaning tank for any of the calculations since the differences between inlet and outlet temperatures were less then 10.5°C. After 1200-1800 s, the temperatures increased with similar rates in time at different levels in the flow domain above the level of the working FAs’ bottom perforations.

— At 1800-2400 s, the sum by-pass flow became higher than the net flow through the FAs’ active parts and increased further up to 70% in case of the W11_F19_DIS0 calculation, up to 75% for the W11_F19_DIS1 calculation and up to 81% in case of the W11_F19_DIS2 calculation after 8000 s (see Fig. 6).

— The calculations show that the maximum temperature could not reach the saturation temperature even after 8000 s. The development of the maximum temperatures may be approached with trend lines as it can be seen in Fig. 12. It show that for the three calculations the increasing of the maximum temperatures had a rate of about 3, 4 and 4,6 °C/1000 s (see Fig. 12). With these rates the saturation temperature (~120 °C) would have been reached after 22900, 17400 and 14700 s for the W11_F19_DIS0-DIS1-DIS2 calculations respectively.
On the other hand, it has to be emphasized that the initial temperature of the cleaning tank which was valid at 16:40, 10\textsuperscript{th} of April, 2003 is unknown. The results show clearly that the temperatures increased with the same rate in time in the flow domain above the level of the working FAs’ bottom perforations after 1200-1800 s. It refers to that independently from the initial temperature a similar stratified temperature field could have developed after a while. It is likely that in case the initial temperature had been higher, the saturation temperature would have been reached earlier.

From the above explained details, the following most important conclusions may be drawn:

Our calculations showed that even with or without any fuel assembly displacements the temperature stratification could have been developed and blocked the flow through the fuel assemblies and the saturation temperature could have been reached after enough long period of time.

On the other hand, by taking into account the fact that the maximum temperature reached the saturation within 8400 s during the incident, our calculations make it likely that beside the working fuel assembly’s bottom perforations more by-pass area was present in the bottom part of the cleaning tank.
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APPENDIX B/X.

THERMAL-HYDRAULIC AND SOURCE TERM CALCULATIONS FOR THE OECD PAKS FUEL PROJECT

Gábor L. Horváth (Hungary, VEIKI)

OECD-IAEA Paks Fuel Project
SHORT SUMMARY

The present report has been prepared under the OECD-IAEA Paks Fuel project and contains the description of the MELCOR 1.8.5 calculations for the Cleaning Tank incident at Paks NPP.

Based on several calculations with different MELCOR models (PWR and BWR) it can be concluded that MELCOR reproduced the incident phenomenon well, and the experience can be utilized in plant calculations. There is no big difference between the MELCOR PWR and BWR models results in term of accident progression and activity release. However the MELCOR BWR physical model describes the heat transfer and oxidation phenomena of fuel assembly channel boxes special for this case better.

Calculations were done with the vent line closed. Some studies were also done with vent line on the tank open. Preliminary results suggest, that some core damage might have happened even in case of vent line open. However lack of precise data prevents us from drawing strong conclusions on this case.
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1. **INTRODUCTION**

On 10 April 2003 severe damage of fuel assemblies took place during an incident at Unit 2 of Paks Nuclear Power Plant in Hungary. The assemblies were being cleaned in a special tank (Fig. 2.2) below the water level of the spent fuel storage pool in order to remove crud buildup. That afternoon, the chemical cleaning of assemblies was completed and the fuel rods were being cooled by circulation of storage pool water. The first sign of fuel failure was the detection of some fission gases released from the cleaning tank during that evening. The cleaning tank cover locks were released after midnight and this operation was followed by a sudden increase in activity concentrations. The visual inspection revealed that all 30 fuel assemblies were severely damaged. The first evaluation of the event showed that the severe fuel damage happened due to inadequate coolant circulation within the cleaning tank.

In the framework of the OECD-IAEA Paks Fuel Project numerical simulation of the Paks-2 incident is to be carried out by project participants. This report summarizes the calculations performed with the MELCOR 1.8.5. version on IBM PC.

2. **MELCOR PWR INPUT MODEL OF THE Paks-2 Cleaning Tank**

Two basic versions have been elaborated for the Paks-2 cleaning tank, the MELCOR PWR and the MELCOR BWR model. This chapter describes the MELCOR PWR model. The following systems were included into the MELCOR [2] modelling scope for the cleaning tank (Fig. 2-1. and table 2-3.).

**Systems:**
- Inlet pipe with constant inlet flow (22.0 t/h) simulating submersible pump
- Cleaning tank with 30 fuel assemblies in it
- Outlet pipe leading to water pool of shaft No.1 containing the cleaning tank
- Surge tank connected to the cleaning tank by a vent pipe
- Pool of Shaft No.1

**Processes:**
- Constant inlet flow (22 t/h) to vessel via the inlet pipe
- Heat transfer from fuel elements to coolant
- Release of dissolved gases in boiling conditions (water assumed to be saturated by gases at pool conditions [6])
- Heat-up, oxidation and melting of fuel and structures
- Fission product release and transport.

2.1. **Cleaning tank volumes**

The location of the cleaning tank in shaft No. 1 is given in Fig. 2-1. The main elements of the cleaning tank real geometry is given in Fig. 2-2. The MELCOR interpretation of the tank as an input model is given in and Fig. 2-3.

The system was modelled with a total of 49 control volumes according to Fig. 2-1. and 2-3. illustrating the control volume numbers and flow-paths connecting them. The two inlet and the 2 outlet pipes were lumped together as two single pipes. The two tanks (inner and outer) and the double cover were modelled separately. The inner vessel was divided as follows:
- 1 inlet tube control volume (CV1)
- Single lower plenum (inlet to fuel assemblies, CV2)
5 vertical ring channels each divided into 7 control volumes (CV201-217 …CV602-617) along the height (1 lower unheated, 5 heated with UO2, 1 upper unheated,
1 Upper plenum (CV5)
4 CV outside of the fuel assemblies (FA, CV63, CV62, CV61 and CV6),
2 CV exit pipes divided in vertical direction (CV7 and CV71)
The surge tank (CV99) was connected to the cleaning tank at the top of the upper plenum (CV5). 11 fixed FA have been considered with 12 holes (each with d=9mm) at the bottom and 12 holes at the top. The 19 follower FA were without holes. The geometry of fixed FA and followers were considered to be the same using the fixed FA as base.

2.2. **Heat sink**
Heat loses from the inner vessel to the vacuum gap between the inner and outer vessel has been modelled both at vessel wall and bottom and at vessel cap. Subsequently the outer vessel communicated the heat to the pool of shaft No.1.
The pool of Shaft No.1 has been assumed very large so its temperature did not change after absorbing the decay heat of the fuel elements in the cleaning tank.

<table>
<thead>
<tr>
<th>Cvol. No.</th>
<th>Compartments</th>
<th>Total volume,m3</th>
<th>Water volume,m3</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Inlet tube</td>
<td>0,0925</td>
<td>Full</td>
</tr>
<tr>
<td>2</td>
<td>Lower plenum (elliptical bottom, inner vessel)</td>
<td>0,3755</td>
<td>Full</td>
</tr>
<tr>
<td>31</td>
<td>Core ring 1: Central hole of vessel (No fuel, closed from top)</td>
<td>0,026</td>
<td>Full</td>
</tr>
<tr>
<td>202-217</td>
<td>Core ring 2: Fuel assemblies Fixed No.1-6</td>
<td>0,1631</td>
<td>Full</td>
</tr>
<tr>
<td>302-317</td>
<td>Core ring 3: Fuel assemblies Fixed No.7-11,Follower No.12</td>
<td>0,1631</td>
<td>Full</td>
</tr>
<tr>
<td>402-417</td>
<td>Core ring 4: Fuel assemblies, Followers No.13-18</td>
<td>0,1631</td>
<td>Full</td>
</tr>
<tr>
<td>502-517</td>
<td>Core ring 5: Fuel assemblies, Followers No.19-24</td>
<td>0,1631</td>
<td>Full</td>
</tr>
<tr>
<td>602-617</td>
<td>Core ring 6: Fuel assemblies, Followers No.25-30</td>
<td>0,1631</td>
<td>Full</td>
</tr>
<tr>
<td>5</td>
<td>Upper plenum (inner vessel)</td>
<td>0,367</td>
<td>Full</td>
</tr>
<tr>
<td>63</td>
<td>Outside fuel elements upper part (inner vessel)</td>
<td>0,6862</td>
<td>Full</td>
</tr>
<tr>
<td>62</td>
<td>Outside fuel elements upper middle part (inner vessel)</td>
<td>1,157</td>
<td>Full</td>
</tr>
<tr>
<td>61</td>
<td>Outside fuel elements lower middle part (inner vessel)</td>
<td>0,9304</td>
<td>Full</td>
</tr>
<tr>
<td>6</td>
<td>Outside fuel elements lower most, exit part (inner vessel)</td>
<td>0,9304</td>
<td>Full</td>
</tr>
<tr>
<td>7</td>
<td>Exit (riser) tube lower part</td>
<td>0,0467</td>
<td>Full</td>
</tr>
<tr>
<td>71</td>
<td>Exit (riser) tube upper part</td>
<td>0,11</td>
<td>Full</td>
</tr>
<tr>
<td>8</td>
<td>Vacuum gap between the internal and external vessels</td>
<td>3,2761</td>
<td>Vacuum</td>
</tr>
<tr>
<td>9</td>
<td>Vacuum gap between internal and external vessel caps</td>
<td>0,048</td>
<td>Vacuum</td>
</tr>
<tr>
<td>99</td>
<td>Surge tank</td>
<td>Estimated 0,1</td>
<td></td>
</tr>
<tr>
<td>100</td>
<td>Shaft No.1 (pool)</td>
<td>Very large 1000 estimate</td>
<td></td>
</tr>
</tbody>
</table>

**Table 2-1. Control volumes and their main properties for the Paks-2 fuel cleaning tank**

2.3. **Heat structures**
All the walls of the pipes and tanks has been calculated for metal volumes and included as heat structures. Elliptical caps have been recalculated to be modelled as a hemisphere. In case of core boundary HS (HS002 to HS021) structure to structure radiation heat transfer has been taken into account through the vacuum gap between the two vessels. HS002 to HS0021 representing internal vessel wall radiate to HS801 representing outer vessel wall.
2.4. **Cleaning tank flow paths**

The flow-paths connecting the control volumes are summarised in table 2-2. and Fig. 2-3. The basic geometry of the tank given in [1] is reproduced in Fig. 2-2. The details of the connections were calculated using hydraulic handbooks of [3,4,5].

<table>
<thead>
<tr>
<th>Fl. No.</th>
<th>CV from</th>
<th>Compartment from</th>
<th>CV to</th>
<th>Compartment to</th>
<th>Area,m²</th>
</tr>
</thead>
<tbody>
<tr>
<td>11</td>
<td>100</td>
<td>Shaft No.1 Pool</td>
<td>1</td>
<td>Inlet tube (Downcomer)</td>
<td></td>
</tr>
<tr>
<td>12</td>
<td>1</td>
<td>Inlet tube (Downcomer)</td>
<td>2</td>
<td>Lower plenum (LP)</td>
<td></td>
</tr>
<tr>
<td>131</td>
<td>2</td>
<td>LP</td>
<td>31</td>
<td>Core ring1: Vessel central hole</td>
<td>Closed</td>
</tr>
<tr>
<td>141</td>
<td>31</td>
<td>Central hole of vessel</td>
<td>5</td>
<td>UP</td>
<td>Closed</td>
</tr>
<tr>
<td>202</td>
<td>2</td>
<td>LP</td>
<td>202</td>
<td>Core ring2: lower unheated</td>
<td>0,0425</td>
</tr>
<tr>
<td>302</td>
<td>2</td>
<td>LP</td>
<td>302</td>
<td>Core ring3: lower unheated</td>
<td>0,0305</td>
</tr>
<tr>
<td>402</td>
<td>2</td>
<td>LP</td>
<td>402</td>
<td>Core ring4: lower. unheated</td>
<td>0,0425</td>
</tr>
<tr>
<td>502</td>
<td>2</td>
<td>LP</td>
<td>502</td>
<td>Core ring5: lower unheated</td>
<td>0,0425</td>
</tr>
<tr>
<td>602</td>
<td>2</td>
<td>LP</td>
<td>602</td>
<td>Core ring6: lower unheated</td>
<td>0,0425</td>
</tr>
<tr>
<td>141</td>
<td>31</td>
<td>Central hole of vessel</td>
<td>5</td>
<td>UP</td>
<td>Closed</td>
</tr>
<tr>
<td>221</td>
<td>217</td>
<td>Core ring2: upper unheated</td>
<td>5</td>
<td>UP</td>
<td>0,0426</td>
</tr>
<tr>
<td>321</td>
<td>317</td>
<td>Core ring3: upper unheated</td>
<td>5</td>
<td>UP</td>
<td>0,0426</td>
</tr>
<tr>
<td>421</td>
<td>417</td>
<td>Core ring4: upper unheated</td>
<td>5</td>
<td>UP</td>
<td>0,0426</td>
</tr>
<tr>
<td>521</td>
<td>517</td>
<td>Core ring5: upper unheated</td>
<td>5</td>
<td>UP</td>
<td>0,0426</td>
</tr>
<tr>
<td>621</td>
<td>617</td>
<td>Core ring6: upper unheated</td>
<td>5</td>
<td>UP</td>
<td>0,0426</td>
</tr>
</tbody>
</table>

Core intermediate Fl see on Fig. 2-3.

<table>
<thead>
<tr>
<th>Fl. No.</th>
<th>CV from</th>
<th>Compartment from</th>
<th>CV to</th>
<th>Compartment to</th>
<th>Area,m²</th>
</tr>
</thead>
<tbody>
<tr>
<td>150</td>
<td>5</td>
<td>Upper Plenum</td>
<td>63</td>
<td>Outside FA Uppermost</td>
<td></td>
</tr>
<tr>
<td>151</td>
<td>63</td>
<td>Outside FA Uppermost</td>
<td>62</td>
<td>Outside FA Upper middle</td>
<td></td>
</tr>
<tr>
<td>152</td>
<td>62</td>
<td>Outside FA Upper middle</td>
<td>61</td>
<td>Outside FA Lower middle</td>
<td></td>
</tr>
<tr>
<td>153</td>
<td>61</td>
<td>Outside FA Lower middle</td>
<td>6</td>
<td>Outside FA Lowermost</td>
<td></td>
</tr>
<tr>
<td>16</td>
<td>6</td>
<td>Outside FA Lowermost</td>
<td>7</td>
<td>Riser tube</td>
<td></td>
</tr>
<tr>
<td>17</td>
<td>7</td>
<td>Riser tube</td>
<td>71</td>
<td>Exit tube</td>
<td></td>
</tr>
<tr>
<td>7</td>
<td>71</td>
<td>Exit tube</td>
<td>100</td>
<td>Shaft No.1 Pool</td>
<td></td>
</tr>
</tbody>
</table>

**Air vent**

<table>
<thead>
<tr>
<th>Fl. No.</th>
<th>CV from</th>
<th>Compartment from</th>
<th>CV to</th>
<th>Area,m²</th>
</tr>
</thead>
<tbody>
<tr>
<td>18</td>
<td>5</td>
<td>Upper Plenum</td>
<td>99</td>
<td>Surge tank</td>
</tr>
</tbody>
</table>

**By-passes: Bottom leak**

<table>
<thead>
<tr>
<th>Fl. No.</th>
<th>CV from</th>
<th>Compartment from</th>
<th>CV to</th>
<th>Area,m²</th>
</tr>
</thead>
<tbody>
<tr>
<td>101</td>
<td>2</td>
<td>LP</td>
<td>6</td>
<td>Outside FA Lowermost</td>
</tr>
</tbody>
</table>

**By-passes: Bottom holes**

<table>
<thead>
<tr>
<th>Fl. No.</th>
<th>CV from</th>
<th>Compartment from</th>
<th>CV to</th>
<th>Area,m²</th>
</tr>
</thead>
<tbody>
<tr>
<td>222</td>
<td>202</td>
<td>Ring2 lower unheated</td>
<td>6</td>
<td>Outside FA Lowermost 6 fix FA</td>
</tr>
<tr>
<td>223</td>
<td>302</td>
<td>Ring3 lower unheated</td>
<td>6</td>
<td>Outside FA Lowermost,5Fix+1Foll. FA</td>
</tr>
</tbody>
</table>

**By-passes: Top holes**

<table>
<thead>
<tr>
<th>Fl. No.</th>
<th>CV from</th>
<th>Compartment from</th>
<th>CV to</th>
<th>Area,m²</th>
</tr>
</thead>
<tbody>
<tr>
<td>223</td>
<td>217</td>
<td>Ring2 upper unheated</td>
<td>63</td>
<td>Outside FA Uppermost 6 fix FA</td>
</tr>
<tr>
<td>323</td>
<td>317</td>
<td>Ring3 upper unheated</td>
<td>63</td>
<td>Outside FA Uppermost, 5Fix+1Foll FA</td>
</tr>
</tbody>
</table>

Table 2-2. Flow-paths and their main properties for the Paks-2 fuel cleaning tank.
2.5. **Core model**

Degraded core calculations were performed in MELCOR by the COR package. The steady-state core power consists of two parts:

- in the COR package the fission power is defined as zero
- in DCH package the residual (decay) power has been specified as a value valid 13 days after the shutdown. MELCOR calculated it as 258.2 kW.

The core (Fig.2-4.) has been modelled with

- 6 radial rings (ring1 is empty representing the central hole) and
- 21 axial levels in the core.

In the COR package the core has been extended to the lower plate of the internal vessel.

The COR 6 radial rings and 21 axial levels have been grouped into several control volumes as follows (see Table 2-3., Fig 2-3. and Fig.2-4., Fig.2-5.).

All the walls and metal structures have been calculated for mass and surface and included in COR packages as support or non-support structures. The elliptical bottom have been recalculated to be modelled as a hemisphere as required by MELCOR.

<table>
<thead>
<tr>
<th>Control volume name</th>
<th>CV</th>
<th>CV</th>
<th>CV</th>
<th>CV</th>
<th>CV</th>
<th>CV</th>
<th>COR ax. level</th>
</tr>
</thead>
<tbody>
<tr>
<td>Upper unheated + head</td>
<td>31</td>
<td>217</td>
<td>317</td>
<td>417</td>
<td>517</td>
<td>617</td>
<td>17-21</td>
</tr>
<tr>
<td>UO2 Fuel 5</td>
<td>31</td>
<td>215</td>
<td>315</td>
<td>415</td>
<td>515</td>
<td>615</td>
<td>15-16</td>
</tr>
<tr>
<td>UO2 Fuel 4</td>
<td>31</td>
<td>213</td>
<td>313</td>
<td>413</td>
<td>513</td>
<td>613</td>
<td>13-14</td>
</tr>
<tr>
<td>UO2 Fuel 3</td>
<td>31</td>
<td>211</td>
<td>311</td>
<td>411</td>
<td>511</td>
<td>611</td>
<td>11-12</td>
</tr>
<tr>
<td>UO2 Fuel 2</td>
<td>31</td>
<td>209</td>
<td>309</td>
<td>409</td>
<td>509</td>
<td>609</td>
<td>9-10</td>
</tr>
<tr>
<td>UO2 Fuel 1</td>
<td>31</td>
<td>207</td>
<td>307</td>
<td>407</td>
<td>507</td>
<td>607</td>
<td>7-8</td>
</tr>
<tr>
<td>Lower unheated + end pieces</td>
<td>31</td>
<td>202</td>
<td>302</td>
<td>402</td>
<td>502</td>
<td>602</td>
<td>2-6</td>
</tr>
<tr>
<td>Lower plenum</td>
<td>2</td>
<td>2</td>
<td>2</td>
<td>2</td>
<td>2</td>
<td>2</td>
<td>1</td>
</tr>
</tbody>
</table>

**Table 2-3. Control volumes and COR package axial levels of fuel assemblies of cleaning tank**

Core modelling parameters:

- Radiative exchange factors (COR0003)
  - The default values of 0.25 are used

- Candling Heat Transfer Coefficients (COR0005)
  - The default values of 1000 are reduced to 300 W/m²K. The reason is to delay the refreezing process, i.e. to delay the forming of core blockage which causes steam starvation and leads to less hydrogen generation.

- Component support flags (CORZjjNS and SS)
  - All the structures have been modelled as non support structures (NS) steel or zirconium except vessel lower support plate which was a support structure (SS). The later can support particulate debris until its failure temperature is reached. In the core no supporting structure is modelled, i.e. in this region if particulate debris is formed, it can be relocated to the next lower level.

- Vessel failure
  - The vessel fails due to creep rupture but no debris ejection has been modelled.

The fuel assemblies were divided into 5 groups and placed into rings 2-6. (Ring1 was empty).
Decay distribution and locations of assemblies is given in Fig. 2-6. and 2-7 and also in the above table.

<table>
<thead>
<tr>
<th>COR ax. level</th>
<th>Material</th>
<th>Name</th>
</tr>
</thead>
<tbody>
<tr>
<td>21NS</td>
<td>STEEL</td>
<td>Fuel Assemblies head exit area</td>
</tr>
<tr>
<td>20NS</td>
<td>STEEL</td>
<td>Upper protecting plate above Fuel Pins</td>
</tr>
<tr>
<td>19NS</td>
<td>ZIRC</td>
<td>Above fuel exit up to protecting plate</td>
</tr>
<tr>
<td>18NS</td>
<td>ZIRC</td>
<td>Upper fuel plugs</td>
</tr>
<tr>
<td>17NS</td>
<td>ZIRC</td>
<td>Fuel pin upper unheated (gas+upp.grid)</td>
</tr>
<tr>
<td>16NS</td>
<td>ZIRC</td>
<td>Core_10</td>
</tr>
<tr>
<td>15NS</td>
<td>ZIRC</td>
<td>Core_9</td>
</tr>
<tr>
<td>14NS</td>
<td>ZIRC</td>
<td>Core_8</td>
</tr>
<tr>
<td>13NS</td>
<td>ZIRC</td>
<td>Core_7</td>
</tr>
<tr>
<td>12NS</td>
<td>ZIRC</td>
<td>Core_6</td>
</tr>
<tr>
<td>11NS</td>
<td>ZIRC</td>
<td>Core_5</td>
</tr>
<tr>
<td>10NS</td>
<td>ZIRC</td>
<td>Core_4</td>
</tr>
<tr>
<td>9NS</td>
<td>ZIRC</td>
<td>Core_3</td>
</tr>
<tr>
<td>8NS</td>
<td>ZIRC</td>
<td>Core_2</td>
</tr>
<tr>
<td>7NS</td>
<td>ZIRC</td>
<td>Core_1</td>
</tr>
<tr>
<td>6NS</td>
<td>ZIRC</td>
<td>Fuel pin lower unheated</td>
</tr>
<tr>
<td>5NS</td>
<td>STEEL</td>
<td>Fuel rod lower support plate</td>
</tr>
<tr>
<td>4NS</td>
<td>ZIRC</td>
<td>Below plate - Ass. side</td>
</tr>
<tr>
<td>3NS</td>
<td>STEEL</td>
<td>Fuel Assemblies end pieces</td>
</tr>
<tr>
<td>2NS</td>
<td>STEEL</td>
<td>Tail pieces in holes of Vessel Lower Support Plate</td>
</tr>
</tbody>
</table>

Table 2-4. Material of core non support structures of Paks-2 fuel cleaning tank core.

Initial conditions were the flowing:
- Pool temperature 30 C
- Tank coolant temperature 57 C
- Fuel assembly temperatures Coolant+1-2 C
- Pump flow rate 22 t/h
- Vacuum in the gap between the internal and external tanks 0.1 and 0.01 bar (in PWR and BWR models resp.)

3. **MELCOR BWR INPUT MODEL OF THE Paks-2 Cleaning Tank**

The MELCOR BWR model (Fig. 3-1.) – different from PWR model mainly by the presence of by-pass CVs (CV702 … CV717) and corresponding flow paths - can model the channel boxes of the VVER-440/213. It allows to simulate heat transfer to space outside of the fuel assemblies modelled as by-pass region of a BWR core. In this way it was possible to avoid the modelling of the oxidation of VVER-440 Zr channel boxes by a double surface area.

The space outside of the fuel assemblies (by-pass) has been divided along the vertical in the same way as the CVs along the core axial levels (Fig. 3-1.). In radial direction no division has been done,
a by-pass CV on an axial level covered the whole cross section outside of fuel assemblies. Opposite to the PWR model where the fuel-containing outer-most core ring (No6) was transferring heat to internal vessel wall simulated as core wall (and the CVs outside of the fuel assemblies (CV63,62,61,6) had no heat loss via HS), in the BWR model space (CVs) outside of the fuel assemblies (by-pass) transferred heat the to wall of the inner vessel modelled as core wall. These by-pass CVs also communicated with the space inside the fuel assemblies by heat conduction via the assembly walls (channel boxes). However the radiative heat from the outermost core ring (Ring6) gets absorbed directly by the core wall and not by the gas in bypass CVs.

4. SCOPING STUDIES

4.1. Identification of safe envelope of operation

Simple hand calculations were done to identify the safe margins of inlet flow-rates at assembly bottom without and with steam generation.

Fig. 4-1. Absolute flow-rate ensuring steady-state heat removal (without steam accumulation) from groups of FA of Paks-2 cleaning tank.
The flow-rate that would fully remove the decay heat of a group of 5 FA with 30 C entrance temperature and \( f_{\text{steam}} \) fraction of inlet flow evaporating (without steam accumulation).

\[
G = \frac{N}{(C_p \cdot dT + f_{\text{steam}} \cdot r)}
\]

where:

- \( N \) - decay heat of 30 fuel elements, 258.2/5 kW
- \( C_p \) - specific heat of water 4.181 kJ/(kg C)
- \( dT \) - water heat-up from 30 C to saturation at \( p=2\text{bar} \) 120-30=90C
- \( r \) - latent heat of evaporation, 2202.2 kJ/kg

Note that there is a small difference of MELCOR calculated value of 258.2 kW and OECD-IAEA Paks Fuel project Data Base value of 242kW in the total decay heat.

The figures 4-1. and 4-2. show that about 12% of nominal flow-rate would have been enough to remove the decay without boiling and an even smaller flow if the vent line could remove the steam generated to avoid level depression and uncovery.

4.2.  Identification of time range of accident development

Knowing the amount of steam generated one can obtain the time available until this steam fills the UP (vessel cap) initiating the uncovery provided, that the vent line to surge tank (and subsequently to atmosphere) is closed.
\[
\begin{align*}
\frac{dt}{v^n} = \frac{V_{UP}}{v^n \cdot G_{steam}} = \frac{V_{UP}}{v^n \cdot G \cdot f_{steam}}
\end{align*}
\]
where:
- \(V_{UP}\) - Volume of upper plenum, 0.367 m³
- \(v^n\) - specific volume of steam, m³/kg
- \(G_{steam}\) - steam generation rate, kg/s
- \(G\) - total flow rate, kg/s
- \(f_{steam}\) - steam flow fraction

Calculating the time needed to fill the UP by steam using the above formulae and doing a cross plotting vs. steady state flow for heat removal with steam generation one can obtain the time range for accident development. (Fig. 4-3.)

Figure 4-3. shows a very steep change around 12\% of nominal flow. Fig. 4-3. also contains the flow-rates experienced with MELCOR (see ch. 5.) before the strong two-phase flow, which shows that some channels are generating steam and some are not both in MELCOR and in simple hand calculations. The figure show also the time needed to fill the UP with released dissolved gases. It shows that uncovering by dissolved gases in not a real danger.

Fig. 4-3. Time needed to fill the upper plenum (vessel cap) by steam with relative flow-rate ensuring steady-state heat removal with steam generation from groups of FA of Paks-2 cleaning tank. (Relative to nominal flow to group assuming 22 t/h for the whole vessel.)
5. TRANSIENT CALCULATIONS BY MELCOR PWR MODEL

5.1. Definition of initiating events and scenarios
In all the studies the bottom and top holes in fixed fuel assemblies were present. Several runs were done with different values of bypass due to missposition of some of the fuel assemblies. Runs with no disposition of any assembly did not reach boiling in any place of the assemblies – so no transient happened. Also if vent line was suddenly open at moderate steam generation rate, most of the steam got discharged from the tank and the FA were re-flooded. The final calculations were done with vent line closed.

The flows at inlet were specified as one-way so no internal recirculation via the core rings was assumed. Ring No. 3 was assumed to be miss-positioned (area=0.0120m2) and inlet flow area to ring 3 lowermost CV (FA unheated end tail containing the bottom holes ) has been reduced by the same amount.

5.2. Reconstruction of the incident by MELCOR analysis
The objective of the analysis was to determine the thermal-hydraulic behaviour and the source term of the incident at Paks-2 cleaning tank as it happened. Computer code used for the analysis was MELCOR 1.8.5. The calculation was started from t= 0 s which corresponded to switching to low flow-rate pump – operation “AMDA C”. The timing of the main events is shown in Table 5-1.

<table>
<thead>
<tr>
<th>Run 9nA, PWR model, vent line closed</th>
<th>Accident phase</th>
<th>Mark</th>
<th>Time, s</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Initial heat-up</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Initiating event</td>
<td></td>
<td>0.0</td>
<td></td>
</tr>
<tr>
<td>Core starts to boil</td>
<td>Boil-off</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ring4 elev. 415 B 5600</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ring3 elev. 315 7101</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ring5 elev. 515 7310</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ring6 elev. 615 7366</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ring2 elev. 215 7383</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Core heat-up</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Core uncovery</td>
<td>Ring3-4 U 9600</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Core exit temperature above 550 C</td>
<td>Ring4 5 15921</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Core damage</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Start of Zr-H2O reaction (1100K=827C)</td>
<td>Ring4 Zr 17804</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Gap release (1173K = 900C)</td>
<td>Ring4 17977</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ring5 21030</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ring3 22123</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Failed steel (T&gt;1273 K)</td>
<td>Not reached</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Molten steel (T&gt;1700 K)</td>
<td>Not reached</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Table 5-1. Main events during the Paks-2 cleaning tank incident reconstruction calculations using MELCOR 1.8.5 PWR model.

5.3. Thermal-hydraulics
The sequence can be divided into the following phases:
- initial heat-up
- boil off
- core heat up and damage
The initiating event was a switch to submersible pump - of intermediate cooling mode with low water flowrate (10/APR/2003 at 16.40) of 22t/h which is 6.11kg/s. It can be seen on Fig. 5-1. that this flow is available in inlet pipe and inlet to lower plenum. The flow in outlet (exit) pipe is the same until the start of boiling. The temperatures were also stable and around 40 C in the exit pipe (Fig. 5-2.). Pressures (Fig. 5-3.) and tank levels (Fig.5-4.,5., 6.) were also stable. However cladding temperatures (Fig. 5-7., 8.) were slowly rising to saturation, where boiling started (B Fig. 5-8.).

This was the result of diverting part of the flow to miss-position bypass and FA bottom holes instead of entering the fuel part of FA. Fig. 10. shows that a large flow is entering Ring2 and Ring3 (containing fixed FA with bottom holes) but most of this flow is lost via the holes. Flow to ring 3 is smaller (Fig. 5-10.) because the miss-positioning also reduces the inlet to FA. The real flow to UO2 fuel part – after the holes – is much smaller (curve Fix-In307 marked 7 on Fig. 5-10.). The dominating flow is the bypass via the miss-positions (Fig.5-10. Fix+FollBcross marked B, 0.0120m2).

It took 5600s until the core first started to boil near the exit in ring 4, which contained followers 13-18 with the highest power (Fig. 2-6. Ring4) near the exit (except FA12 which has been lumped to ring3). However the decay heat could be removed from FA in steady boiling until the steam depressed the level to the top of FA or UO2 fuel (U – 9-10000s, Fig.5-4., 5-5.). Soon after the uncovery U the clad and coolant FA exit temperatures started to rise (Fig. 5-7., 5-8.). The delay is attributed to the fact that there is an unheated part at the top of FA. Coolant FA exit temperatures reached 550 C at 15921s marked as 5. on figures.

Tank water levels outside of the assemblies got also depressed. The level decreased below the top of bottom opening of the exit tube entrance opening height (Fig. 5-6.) during intensive steam generation periods. However before the start of Zr-H2O reaction apart from some steam mostly bypass water was discharged with temperature slightly declining after a brief peak (Fig. 5-2.). The strong depression and some steam discharge is explained by the fact that in these calculations vent line was assumed to be closed.

The clad maximal temperature reached 1100K=827C at time 17804s marker Zr. This means the beginning of Zr-H2O endothermic reaction. At this late stage most of water is lost to miss-position by pass (Fig. 5-10.).

The total heat loss to Shaft No1 pools via the gap between the internal and external vessels is mostly below 1kW. This low heat loss is explained by the 0.1 bar vacuum assumed between the internal and external vessels allowing internal circulation and convective heat transfer in the gap and by the low temperatures in outermost core ring (Ring6) resulting in low radiation from the core to internal vessel wall selected as core wall (Fig. 5-7).

No vessel failure was achieved because the calculations stopped with water still in the vessel and it was not intended to calculate further.

Studies with the vent line open were also done. Preliminary results suggest, that core damage might have happened even in case of vent line open. However lack of precise data prevent us from drawing strong conclusions on this case.

5.4. Radioactivity release

The radioactive release (Fig. 5-12), was started soon after the Zr-H2O reaction at 17977s with gap release in ring 4. The release continued from fuel during further heat-up and failure of cladding in 2 more rings about 1 hour later.
6. TRANSIENT CALCULATIONS BY MELCOR BWR MODEL

All the conditions in the BWR model were the same as in PWR model except the by-pass geometry described in Ch. 3, which meant mainly the existence of separate channel boxes and by-pass volume representing the space outside of the fuel assemblies. In addition BWR calculations were done with 0.01 bar vacuum between the two tanks opposite to PWR calculations of 0.1 bar. The radiation emissivities of core wall were 0.8 instead of 0.1 in PWR case. However these changes did not make significant differences between the PWR and BWR because the PWR case did not reach the time and state when the radiation becomes really significant.

The results given in Table 6-1 and Fig. 6-1. – 6-11. were similar to that of the PWR model with the difference that larger heat loss through the walls of channel boxes to space outside of FA (by-pass) needed a larger bottom by-pass (by 20-35%) to obtain the phenomena experienced. This extra by-pass at the bottom of the fuel assemblies has been attributed to the follower assemblies.

The heat loss to pool is much higher than in PWR case first of all because temperatures in outermost ring (Rin6) radiating to core wall reach the state when they are much higher.

The activity release (Fig. 6-12.) tendencies were very similar taking into account the shift in time of cladding failure.

| Table 6-1. Main events during the Paks-2 cleaning tank incident reconstruction calculations using MELCOR 1.8.5 BWR model. |

<table>
<thead>
<tr>
<th>BWR: Run H, vent line closed</th>
<th>Accident phase</th>
<th>Mark</th>
<th>BWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initial heat-up</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Initiating event</td>
<td></td>
<td>0.0</td>
<td></td>
</tr>
<tr>
<td>Boil-off</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Core starts to boil</td>
<td>Ring4 elev. 415</td>
<td>B</td>
<td>7190</td>
</tr>
<tr>
<td>Core heat-up</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Core uncovery</td>
<td>Ring3-4</td>
<td>U</td>
<td>9800</td>
</tr>
<tr>
<td>Core exit temperature above 550 C</td>
<td>Ring4</td>
<td>5</td>
<td>15706</td>
</tr>
<tr>
<td>Core damage</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Start of Zr-H2O reaction (823 C)</td>
<td>Ring4</td>
<td>Zr</td>
<td>17664</td>
</tr>
<tr>
<td>Gap release</td>
<td>Ring4</td>
<td></td>
<td>17999</td>
</tr>
<tr>
<td></td>
<td>Ring3</td>
<td></td>
<td>21711</td>
</tr>
<tr>
<td></td>
<td>Ring5</td>
<td></td>
<td>23600</td>
</tr>
<tr>
<td></td>
<td>Ring6</td>
<td></td>
<td>23141</td>
</tr>
<tr>
<td>Molten Zr (T&gt;1825 C)</td>
<td></td>
<td></td>
<td>19901</td>
</tr>
</tbody>
</table>

7. COMPARISON TO MEASURED VALUES
Table 7-1. shows the timing of calculated and observed values.

<table>
<thead>
<tr>
<th>Accident phase</th>
<th>Mark</th>
<th>Time, s PWR model</th>
<th>Time, s BWR model</th>
<th>Time, s Observed</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Initial heat-up</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Initiating event</td>
<td></td>
<td>0.0</td>
<td>0.0</td>
<td></td>
</tr>
<tr>
<td>Core starts to boil</td>
<td><strong>Boil-off</strong></td>
<td>Ring4 elev. 415</td>
<td>B</td>
<td>5600</td>
</tr>
<tr>
<td><strong>Core heat-up</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Core uncovery</td>
<td>Ring3-4</td>
<td>U</td>
<td>9600</td>
<td>9800</td>
</tr>
<tr>
<td>Core exit temp. above 550 C</td>
<td>Ring4</td>
<td>$S$</td>
<td>15921</td>
<td>15706</td>
</tr>
<tr>
<td><strong>Core damage</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Zr-H$_2$O reaction start (823 C)</td>
<td>Ring4</td>
<td>Zr</td>
<td>17804</td>
<td>17667</td>
</tr>
<tr>
<td>Gap release (900C)</td>
<td>Ring4</td>
<td></td>
<td>17977</td>
<td>17999</td>
</tr>
<tr>
<td></td>
<td>Ring3</td>
<td></td>
<td>22123</td>
<td>21711</td>
</tr>
<tr>
<td></td>
<td>Ring5</td>
<td></td>
<td>21030</td>
<td>23600</td>
</tr>
<tr>
<td></td>
<td>Ring6</td>
<td></td>
<td>23141</td>
<td></td>
</tr>
<tr>
<td>Molten Zr (T&gt;1825 C)</td>
<td></td>
<td>Not reached</td>
<td></td>
<td>19901</td>
</tr>
</tbody>
</table>

1. 18600-24600s corresponds to time interval of April 10, 16:40-23:30h

Table 7-1. Comparison of calculated and observed main events during the Paks-2 cleaning tank incident using MELCOR 1.8.5 PWR and BWR models.

Characteristic values of clad maximal temperatures were reached at times given below.

<table>
<thead>
<tr>
<th>PWR: Run9nA</th>
<th>Time, s PWR model</th>
<th>Time, s BWR model</th>
</tr>
</thead>
<tbody>
<tr>
<td>BWR: Run H</td>
<td>800 C</td>
<td>17500</td>
</tr>
<tr>
<td></td>
<td>1200 C</td>
<td>18688</td>
</tr>
</tbody>
</table>

Uncovery of the top of fuel has been attributed to the level increase observed in pressurizer and pool level. Gap release has been identified as the activity increase on the AMDA by-pass tank. The observed water temperature exiting the tank was about 37 C, which was reproduced by the calculations well (Fig. 5-2. and 6-2.).

The results were sensitive to size a vacuum (0.01 bar in BWR model) between the two tanks and surface emissivity (0.8 in BWR model) of the vessel walls. These influenced first of all the radiative heat loss to water pool in shaft No1 at later stages. However the heat-up times in PWR and BWR model were not very different because radiation starts to be important only at elevated temperatures not reached in outermost ring in PWR model.

8. CONCLUSIONS
Based on several calculations with different MELCOR models (PWR and BWR) it can be concluded that MELCOR reproduced the incident phenomenon well, and the experience can be utilized in plant calculations. There is no big difference between the MELCOR PWR and BWR models in term of accident progression and activity release. However the MELCOR BWR physical model describes the heat transfer and oxidation phenomena of fuel assembly channel boxes special for this case better.

A coarse nodalisation in the core could not achieve conditions when steam generation stated for the present case with low heat generation.

Studies were done with vent line on the tank closed. Some studies were also done with vent line on the tank open. Preliminary results suggest, that core damage might have happened even in case of vent line open. However lack of precise data prevents us from drawing strong conclusions on this case.

9. REFERENCES

10. FIGURES

Fig. 2-1. Location of the cleaning tank in Shaft No1 of Paks-2 NPP
Fig. 2-2. Geometry of the Paks-2 cleaning tank in Shaft No1 of Paks-2 NPP
Fig. 2-3. MELCOR 1.8.5 PWR nodalization of cleaning tank in Shaft No1 of Paks-2 NPP
Note: Core boundary HS2 …21 transfers heat between outermost core ring CV602…617 and gap between the two vessels.
Fig. 2-4. MELCOR 1.8.5 PWR core nodalization of cleaning tank in Shaft No1 of Paks-2 NPP
Fig. 2-5. Fixed fuel assembly of Paks-2 NPP and MELCOR COR package interpretation in terms of axial control volume distribution.
Fig. 2-6. Radial and axial distribution of FA decay power in the Paks-2 cleaning tank.

Fig. 2-7. Location of FA in the Paks-2 cleaning tank and approximate distribution according to MELCOR model radial rings. (The hexagonal form FAs are at their exact locations.)
Fig. 3-1. MELCOR 1.8.5 BWR core nodalization of cleaning tank in Shaft No1 of Paks-2 NPP
APPENDIX B/XI.

RESULTS OF THE PAKS-2 CLEANING TANK INCIDENT SIMULATION WITH ICARE/CATHARE CODE
Yu. Zvonarev, V. Kobzar, A. Volchek (Russia, KI)

OECD-IAEA Paks Fuel Project
FOREWORD

The presented results of the Paks-2 cleaning tank incident simulation with ICARE/CATHARE code are obtained in the frame of technical collaboration between RRC “Kurchatov Institute” (Russia) and IRSN (France), and should be regarded as a common RRC KI and IRSN contribution to OECD-IAEA Paks fuel project.

The current work is carried out in accordance with recommendations from OECD-IAEA Paks fuel project meetings and input transient and modeling parameters are set following current database of Paks incident. An analysis of the calculation results is done accounting the requirements for the participants of the project.

The ICARE/CATHARE code, developed in IRSN (France), is devoted to calculate in detailed mechanistic way core degradation during severe accidents in LWRs. Both parts of the code, ICARE2 and CATHARE, were developed separately as for analyses of wide range of high temperature core phenomena as for realistic simulations of thermalhydraulics in different reactor components. Recent wide validation of coupled ICARE/CATHARE code against numerous integral experiments and SFD reactor scenarios showed its applicability and high prediction power, in particular, to VVER reactors and VVER type experimental facilities.

The first part of the current work included simulation of thermalhydraulic phase of the incident using CATHARE2 V1.3L_1 code. The second part concerned phenomena during SFD phase, which appeared after dryout of the tank, heat-up and the high temperature evolution of the fuel assemblies. The SFD phase was simulated using the updated version of the ICARE/CATHARE code with ICARE2 V3.2 code version.

1. THERMALHYDRAULIC PHASE OF THE TRANSIENT

It was assumed that 3 of 30 fuel assemblies have additional flow bypass at their entry. This bypass is formed due to incorrect positioning of these assemblies on the support plate of the tank. Cross-flow area of the bypass 120 cm² was chosen in accordance with decisions of the 1st Meeting on OECD-IAEA Paks fuel project. Moreover ordinary bypass due to perforation in the lower part of 11 fixed assemblies was taken into account. Geometry and hydraulic resistance values of the fuel assemblies were reproduced in the input data.

Water inlet mass flow 5.5 kg/s with temperature about 30°C was supplied with a pump to the tank from the cooling pond to cool the fuel assemblies.

The input deck for CATHARE2 code used in that work to simulate thermalhydraulic phase of the incident was developed on the base of information from the AEKI Paks database.

Simplified approach to the modelling of coolant flow through the assemblies in the tank was used in the simulation. Three fixed assemblies were assumed to be wrongly positioned and they were separated to special group. The other 27 correctly positioned assemblies were modelled as one representative assembly. This group of assemblies is divided in current nodalization scheme by two groups. Correctly positioned fixed assemblies and the followers are modelled separately now. It allows more precise simulation of coolant flow thought the assemblies.
The nodalization scheme used for current simulation is presented in Fig. 1.1. Three representative assemblies model thirty fuel assemblies, installed into the tank. The first of them (VOL3 and UP3AX) simulates 3 fixed assemblies with wrong positioning, the second one (VOL8 and UP8AX) models 8 correctly positioned fixed assemblies and the third representative assembly (VOL19 and UP19AX) corresponds to 19 followers. Components VOLDOWN and DOWNAX are used to simulate the space in the tank between fuel assemblies.

Each representative fuel assembly consists of two components. First component of VOLUME type simulates lower part of the assembly from its lower end up to the beginning of fuel column in the fuel rods. Junctions CAS3HL and CAS8HL are used for description of water leakage through the perforation in the lower part of fixed assembly shroud. Coolant within the assemblies (at elevation of fuel column and the assembly top head) is modelled by AXIAL component, while a bundle of fuel rods inside this coolant flow is represented by CATHARE component of FUEL type. All three AXIAL components concerning fuel assemblies are divided into 11 nodes of different lengths. The boundaries between the nodes (vector points) correlate with location of such elements of assemblies as spacer grids, support grids and protective grids. Location of grids is different for the fixed assemblies and the followers. This is the reason of different nodalization for different representative assemblies.
The parts of the tank located below the support plate (lower plenum) and above the assemblies (upper plenum) are modelled by separate VOLUME components. The space of lower plenum in the vicinity of fuel assembly inlet is modelled separately (components IN3VOL, IN8VOL and IN19VOL) to provide a possibility to account pressure gradient inside the lower plenum, which can exist, in particular, due to the leakage from the bypass.

Modelling hydraulic system include one inlet and three outlets. The inlet corresponds to water supply line and is modelled by SOURCE element LPLIN. Junction DOWNOUT and boundary condition BCDOWN simulate the first water outlet line. A pair of components (UPLOUT and BCUP) represents the second outlet through the gas relief pipe. And, at last, the third outlet for the bypass is set by components LEAK and BCLEAK in a manner similar to the first one. Cross section area defined for the junction LEAK corresponds to area of flow bypass formed due to wrong positioning of fuel assemblies.

The set of WALL type components is used to model fuel assemblies shrouds and internal walls of the tank. It allows simulation of the shrouds and the tank walls heat capacity and heat losses from the tank walls.

In accordance with decisions of the First Meeting on OECD-IAEA PAKS Fuel Project the cross-flow area of the bypass at the bottom of the fuel assemblies is assumed to be equal 120 cm². This bypass exists due to coolant leakage between the assembly bottom and the support plate. The reasons of this leakage can be both wrong positioning of some assemblies and fouling in the place of contact between the assemblies and the support plate. Chosen value of bypass area corresponds to bypass flow ~50% of total flow at the tank inlet. Bypass area is assumed to be equally distributed between three fixed assemblies, which are wrongly positioned.

Two axial power profiles assumed for the fixed assemblies and the followers. Value of total decay heat power at the time of the incident was also updated to be equal to 241 kW.

The flow rate of water at the tank inlet was set in the input deck in accordance with Paks database. Flow rate value at cleaning mode is calculated in the simulation by polynomial formula versus time. Polynomial coefficients were chosen to make the flow rate value close to the data of measurements.

The value of heat exchange coefficients equal to 8 W/m²K is used in current simulation to calculate the heat losses through the tank vessel. This value coincides with corresponding parameter used for modelling of late stage of the incident.

The results of CATHARE2 simulation of Paks-2 cleaning tank incident are presented below. The simulation predicts fuel assemblies dryout and beginning of their heatup ~2.5 hours after the initiation of cooling mode of tank operation (inlet water flow rate reduction). The last is used as zero point on time axis of the plots. Performed simulation covers period up to dryout of considerable part of the tank (approximately up to 10⁴ second).

Modelling of the hydraulics is a key point of simulation of initial stage of Paks-2 cleaning tank incident and intensity of fuel rods cooling is mainly determined by the rate of coolant flow through the assemblies. Inlet water supply is distributed among 30 fuel assemblies installed in the tank and partly is bypassed through the wrongly positioned assemblies (3 in this simulation). Certain amount of water can also bypass the fuel rods leaving the assemblies through the holes in lower part. The ratio of inlet flow distribution among heated part of the assemblies, bypass and the holes, which is calculated by the code is shown in Fig. 1.2. One can see that only 10-20% of supplied water takes part in cooling process, while 80-90% of water leaves the tank being cold.
According to our results such amount of water flow through the heated part of fuel assemblies cannot provide effective cooling and outlet water temperature permanently goes up (see Fig 1.3).

The only factor slowing down the heating process is heat capacity of the fuel assemblies. Outlet coolant temperature reaches the saturation value approximately 7000 s after initial inlet flow rate reduction and water boiling started. Water begins to evaporate a little earlier in the followers than in the fixed assemblies because lower flow rate value. Steam void appears under the tank cover and rather quickly expands to lower elevations providing dryout of fuel assemblies. Evolution of boiled-up water level in the tank is presented in Fig 1.4. This value is calculated on the basis of void fraction values predicted by CATHARE2 for four hydraulic channels: three representative
assemblies and the space between the assemblies. Here numerical meshes are considered free, if void fraction exceeded value of 0.99. Water level moves down everywhere in the tank during 8400-9600 seconds (2 hr 20 min - 2 hr 40 min) after the incident beginning. These time values correspond to the time interval, when water level increase was detected in the pressurizer of Paks-2 unit. Downward movement of coolant is predicted for this period in the fuel assemblies along with water mass decrease and flow rate increase at the tank outlet. In our opinion this additional amount of water flow out from the tank provides observable rise of water level in the refuelling pit.

Simulated steam void expansion lasts ~20 minutes. Water level position approximately stabilizes after dryout at elevation 0.5-0.9 m above the support plate (Fig. 1.4), remaining lower parts of the fuel assemblies permanently cooled by water. This stable position of water level correlates with the data of visual observations, which found that the most of the assemblies were destroyed above this elevation. Some uncertainty of predicted level position is connected with the axial nodalization of the tank region occupied by the assemblies. Approximate water level stabilization at later stage of the incident appeared due to stabilization of steam amount in the tank. The steam, generated in the assemblies below the water level, leaves the tank through the gas relief pipe.

Considerable water amount, which is supplied to the tank in the cooling mode, flows through the bypass and shroud perforation and leaves the assemblies being cold. This water is mixed with small amount of hot coolant flowing between the assemblies from upper part of the tank. Predicted resulting temperature at the tank outlet is shown in Fig. 1.5. It is possible to compare these calculated values with the data of measurements made during the incident. Good agreement takes place at times before steam void formation. Measured data are absent for the period of water level lowering and the code predicts increase of coolant temperature at this time. The reason is the reverse of flow in the assemblies. After water level position stabilisation the temperature reduces again.

Predicted evolution of fuel rod cladding temperature is illustrated in Fig. 1.6. The temperatures in all spatial zones above water level rise almost linearly. Maximum value of temperature in assemblies of groups 1 and 2 (fixed assemblies) is predicted at elevation ~1.75 m above the support
plate. Maximum position for followers is higher – 2.0 m. Cladding temperatures near water level are less or near saturation temperature (~120°C).

![Graph of Water Temperature at the Tank Inlet and Outlet](image1.png)

**Fig. 1.5. Base Case: Water Temperature at the Tank Inlet and Outlet.**

![Graph of Fuel Cladding Temperature Axial Profile](image2.png)

**Fig. 1.6. Base Case: Fuel Cladding Temperature Axial Profile (fixed assemblies).**

An uncertainty analysis was performed to estimate the variation range of physical parameters at the beginning of SFD phase. Four series of variant calculations were carried out, including bypass flow area, gas relief pipe flow capacity and decay heat power. It was shown that duration of the period...
from inlet flow reduction until the onset of the assemblies’ dryout varied in wide range (from 6% to 15%) due to these parameter uncertainties. Pressure loss coefficients of the assemblies located in the tank were found to be less important parameters for the incident simulation.

All known events and parameters of initial (thermalhydraulic) phase of the cleaning tank incident were successfully reproduced in CATHARE2 calculation. The results of this calculation formed initial and boundary conditions for subsequent ICARE2 calculations.

2. SFD PHASE OF THE TRANSIENT

According to Paks database all 30 assemblies inside cleaning tank are divided on 6 groups, basing on burn up history and type of the assembly:

1. Fix assemblies 1-6 (burn up 10.9 MW day/kgU, power 7.159 KW);
2. Fix assemblies 7-11 (burn up 27.0 MW day/kgU, power 9.006 KW);
3. Follower 12 (burn up 9.2 MW day/kgU, power 8.154 KW);
4. Followers 13-18 (burn up 21.3 MW day/kgU, power 6.694 KW);
5. Followers 19-24 (burn up 13.9 MW day/kgU, power 8.837 KW);
6. Followers 25-30 (burn up 13.7 MW day/kgU, power 8.719 KW).

Total decay power at time of incident is 241 KW.

Axial meshing in ICARE2 simulation included 20 nodes along heated region of the fuel rods. A nodalization scheme with 3 parallel channels was used. Additionally, recently developed best-fitted correlations for Zr+1% Nb alloy oxidation have been applied.

Architecture of the stand alone ICARE2 code, which was applied for this work, have clearly defined module structure with orientation of main modules on description of separate physical phenomena taking place with components – anagoges of physical objects of the core. Table 1 presents short description of main modules implemented into ICARE2 code. Fig.2.1 illustrates the main structures realized in the input deck.

**Table 1. Brief Description of ICARE2 Models.**

<table>
<thead>
<tr>
<th>Module</th>
<th>Separate models</th>
<th>Brief description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermalhydraulics</td>
<td>Two-phase</td>
<td>Thermalhydraulics of steam-water media mixed with non condensable gases</td>
</tr>
<tr>
<td>Heat transfer</td>
<td>Convection</td>
<td>Exchange between coolant and solid structures</td>
</tr>
<tr>
<td></td>
<td>Thermal Conductivity</td>
<td>Heat transfer inside and between solid structures</td>
</tr>
<tr>
<td></td>
<td>Radiation</td>
<td>Radiation heat transfer between solid structures through the coolant</td>
</tr>
<tr>
<td></td>
<td>Horizontal plates</td>
<td>Heat transfer between horizontal plates and other solid structures</td>
</tr>
<tr>
<td>Power release</td>
<td>Residual power</td>
<td>Is set by the use or calculated on the basis of fission products available</td>
</tr>
<tr>
<td>-------------------------------</td>
<td>--------------------------------------------------------------------------------</td>
<td>--------------------------------------------------------------------------------</td>
</tr>
<tr>
<td>Mechanical behavior</td>
<td>Thermal expansion</td>
<td>Calculation of thermal expansion basing on known materials properties</td>
</tr>
<tr>
<td>Fuel cladding behavior (IRSN)</td>
<td></td>
<td>Calculation of ballooning and burst on the basis on experimental correlations (Zircaloy)</td>
</tr>
<tr>
<td>Chemical interactions</td>
<td>Zr oxidation (IRSN)</td>
<td>Modeling of energy release and oxide scale on the basis of experimental correlations. Recent update for Zr1%Nb claddings</td>
</tr>
<tr>
<td></td>
<td>Fuel-cladding interaction (IRSN)</td>
<td>Modeling of UO₂ dissolution on the basis of experimental correlations</td>
</tr>
<tr>
<td></td>
<td>Steel oxidation (IRSN)</td>
<td>Modeling of energy release on the basis of experimental correlations</td>
</tr>
</tbody>
</table>

Fig. 2.1. Modeling of Main Processes with ICARE2 Input Deck.

The specific of ICARE2 application to simulation of Paks incident includes next items:

- General level common RADB and CONV structures are used to deal with radiation and convective heat transfer from the assemblies (6 groups, defined separately as RODs in input deck) to the internal wall of cleaning tank;
- Similar phenomena, internal to the assembly, are described with separate RADC and CONV structures, applied to the fuel rods surrounded by internal assembly wall (one ROD per the assembly with weight 126);
- All ROD structures are embedded in three-channel two-phase fluid component (type FLUID2). Internal assemblies from 1 to 6 belongs to the first channel, middle assemblies from 11 to 18 – to the second one and peripheral from 19 to 30 – the third;
- Meshing in ICARE2 simulations included 20 nodes along heated region of the fuel rods.
Modeling approach includes application of ZROX, RADB/RADC, GAP, CONV, CREEP, UZRS modules. For most of them default parameters were used except the ones, which are intended to follow the specific of the VVER assembly or findings obtained in course of whole investigation. Recently developed (at KRRC KI) best-fitted correlations for Zr+1%Nb oxidation have been applied in current work for base case instead SOKOLOV ones used in the ICARE2 code.

Among most important characteristics of the regarded transient is temperature behavior of the bundle. Next figures present the results obtained for fuel rod pellets and claddings in a form requested from the OECD-IAEA Paks fuel project participants.

Fig. 2.2 shows axial cladding temperature profile for three assembly groups at the instant, when maximum cladding temperature exceeds 800 °C that happens in simulation approximately at 14360 second. One can see that assembly of the 3rd group (single assembly № 12) first reaches this temperature. The event totally corresponds to the peculiarities of the local decay power, which according to Paks database was the highest namely for this assembly at the same elevation. Later increase of radiation heat transfer smoothes this dependence and hottest zone is shifted to central region.

![Fig. 2.2. Axial Cladding Temperature Profile of the Assemblies.](image-url)
According to simulation results presented in Figures 2.3 and 2.4 maximum fuel rod temperatures does not exceed 1500 °C, which are realized approximately after 24000 seconds of the transient. Our analyses show that growth of maximum temperatures is limited by oxygen starvation. This phenomenon starts in simulation at 2.2 m elevation and leads to decreasing of oxidation rate at levels higher than 1.6 m and stopping of temperature increase. The temperature between fuel and cladding generally is rather small and differs by 5-10 °C.

Fig. 2.5, presented below, gives the evolution of internal fuel rod pressure, which was obtained during the simulation. One can see that first failure of fuel rod occurs at ≈ 16700 second in the central assembly. The latest failure occurs about 500 seconds later on the periphery. It should be noted that these results should be regarded only as a preliminary ones, as they were obtained with application of mechanical behaviour model CREEP, which was developed for Zircaloy claddings.
An update of CREEP module with mechanical properties of Zr1%Nb claddings is foreseen in the nearest future with application to Paks incident at the second half of this year.

The evolution of two values, characterizing oxidation state of the assemblies is presented in Fig. 2.6. One can see that following the simulation results total extent of Zr oxidation is not very high increasing to approximately 15% at the end of the transient. At the same time local oxidation of the fuel rod cladding appears to be substantial at narrow region in central region of the cleaning tank.

Specific Zr oxidation behaviour of the regarded transient is realized in simulation of hydrogen release. Fig. 2.7 presents the evolution of total hydrogen mass, resulted from oxidation of all structures. Following the computation the entire transient can be subdivided in three periods:
- Firstly, when time was less than 20000 s, simulated temperatures are relatively low with no limitations on oxygen supply;
- Afterwards the hydrogen release rate is limited by oxygen starvation and blanketing and remains constant;
- At the end of the transient the primary Zr in major part of hot zone is converted into $\alpha$-Zr(O), most limitations of oxygen uptake disappeared and total hydrogen release rate decreased.

Total hydrogen release reaches in this simulation the value about 11.5 kg.

**Fig. 2.7.** Evolution of Hydrogen Release.

Fig. 2.8 presents axial distribution of simulated external diameter of fuel rods at the instant before quenching, while Fig. 2.9 gives zirconia layer thickness at the same instant.

Axial elevation of each value maximum correlates one with another. Therefore it can be supposed that 25% of cladding radial increase during ballooning lead to subsequent increase in Zr oxidation rate and is one of the reasons of heavy oxidation at these elevations.

The comparison of simulation results for different assembly groups show certain dependence on radial position of the group that is presented in Fig 2.8. Namely, the central group (number 1 in the database list) is simulated with highest temperatures (see Fig. 2.4) and oxide scale thickness (Fig. 2.9), while these values decreases from the center to the periphery (here for middle assembly are taken the results for fourth group). The latter effect is mainly the consequence of substantial role of radiation transfer inside the cleaning tank.
3. CONCLUSIONS

The ICARE/CATHARE code input decks for the thermalhydraulic and SFD phases simulation was developed using data issued from AEKI Paks database.

The code predicted that ~80-90% of the water, supplied into the tank in the cooling mode, was lost through the bypass and the perforation. It led to insufficient cooling of the fuel assemblies, their heatup, formation of steam void in the tank and assembly dryout.
Predicted dynamic of steam void formation, final water level position and the values of water temperature at the tank outlet are in good agreement with the data of measurements.

The code predicted that major part of the water, supplied into the tank in the cooling mode, was lost through the bypass and the perforation. Only 10-20% of supplied water took part in the cooling process. It led to insufficient cooling of the fuel assemblies and their gradual heatup. The temperatures reached the values leading to water boiling ~2 hr after the moment of flow reduction at the beginning of cooling mode. Assembly dryout began at ~2 hr 20 min and continued ~20 min. Water level stabilization was predicted at elevation ~0.7 m, timing of steam void formation and final level position correlated with known plant data.

As a whole the results of thermalhydraulic calculations of Paks cleaning tank incident performed with the ICARE/CATHARE code adequately agreed with currently known features of incident scenario. The obtained HD results will be used as initial and boundary conditions for subsequent simulation of assembly degradation during SFD phase.

The first characteristic event of SFD phase appeared at 14360 second, when maximum cladding temperature exceeds 800 °C. The assembly of the 3rd group (single assembly № 12) first reaches this temperature that totally corresponds to the peculiarities of the local decay power, which according to Paks database is the highest for this assembly at the same elevation. Later increase of radiation heat transfer smoothes this dependence and hottest zone is shifted to central region of the cleaning tank. The assemblies of the 1st group (assemblies № 1-6) first reach temperature of 1200 °C at time 19700 s. According to simulation results maximum fuel rod temperatures does not exceed 1500 °C, which is realized approximately after 24000 seconds of the transient. The analyses show that growth of maximum temperatures is limited by oxygen starvation. The temperature between fuel and cladding generally is rather small and differs by 5-10 °C.

The evolution of internal fuel rod pressure, which is obtained during the simulation, indicates that first failure of fuel rod occurred at ≈ 16700 second in the central assembly. The latest failure occurs about 500 seconds later on the periphery. It should be noted that these results should be regarded only as a preliminary ones, as they were obtained with application of mechanical behaviour model CREEP, which was developed for Zircaloy claddings.

The analyses of evolution of oxidation state of the assemblies show that total extent of Zr oxidation is not very high increasing to approximately 15% at the end of the transient. At the same time local oxidation of the fuel rod cladding appears to be substantial (about 80%) at narrow region in central part of the cleaning tank.

Specific Zr oxidation behaviour of the regarded transient is realized in simulation of hydrogen release. Following the computation the entire transient can be subdivided in three periods. Firstly, when time is less than 20000 s, simulated temperatures are relatively low with no limitations on oxygen supply. Afterwards the hydrogen release rate is limited by oxygen starvation and blanketing and remained constant. At the end of the transient the primary Zr in major part of hot zone is converted into α-Zr(O) and total hydrogen release rate decreased. Total hydrogen release reaches in this simulation the value about 11.5 kg.

Axial distribution of simulated external diameter of fuel rods at the instant before quenching and zirconia layer thickness at the same instant correlates one with another. Therefore it can be supposed that 25% of cladding radial increase during ballooning led to subsequent increase in Zr oxidation rate and is one of the reasons of heavy oxidation at these elevations. The comparison of
simulation results for different assembly groups shows certain dependence on radial position of the group. Namely, the 1st group is simulated with highest temperatures and oxide scale thickness, while these values decrease from the center to the periphery. The latter effect is mainly the consequence of substantial role of radiation transfer inside the cleaning tank.

A current analysis is based on recently developed by RRC KI best-fitted correlations for Zr+1%Nb alloy oxidation, which were obtained from results of several experimental groups. The difference between this kinetic and previously used so-called Sokolov correlations falls mainly in the temperature regions T > 1500 °C and T < 1300 °C, where best-fitted correlations predicts higher rate of oxygen mass gain.

Generally, simulations of current studies confirmed behavior of accident scenario with respect to the uncertainties of the models and incident settings, which was outlined as a result of previous investigations.

Further examinations are foreseen (outside of the OECD-IAEA Paks fuel project) with the improved modelling of mechanical properties of Zr+1%Nb alloys. An update of CREEP module of the ICARE2 code with mechanical properties of Zr1%Nb claddings is foreseen in the nearest future with application to Paks incident at the second half of 2007 year. This work will be performed in the frame of technical collaboration between RRC KI and IRSN.
APPENDIX B/XII.

ANALYSES OF THE PAKS INCIDENT WITH THE ASTEC CODE

Peter Matejovic (Slovak Republic, IVS)

OECD-IAEA Paks Fuel Project
Introduction

Analysis was performed using the integral code ASTEC (Accident Source Term Evaluation Code), version V1.3 rev0. The ASTEC V1 series has been developed jointly by IRSN and GRS since 1998 with the aim to get a fast running code for the simulation of the total sequences of severe accidents in LWR from the initiating event up to the possible fission product release to the environment. In the frame of SARNET Project (6th EU Framework Programme) the code was released to European partners and is widely assessed through validation against number of experiments and benchmarks with integral and/or mechanistic codes. The code applicability to VVER is an important objective, too. The code version V1.3 rev0 was released in December 2006.

The integral code ASTEC consists of several modules. Only several of them were used in the analysis of Paks cleaning tank accident:

- CESAR for RCS two-phase thermal-hydraulics during the front-end phase and the degradation phase;
- DIVA for core degradation including late phase phenomena (molten pool, corium slump to lower head, corium in lower head) and vessel failure;
- ELSA for release of FP from fuel rods and debris and of materials from control rods, using a semi-empirical approach;
- SOPHAEROS for FP vapour and aerosol transport in RCS;

Normally, in typical reactor applications, the ASTEC modules are working in coupled mode exchanging the relevant data through common database. During the front-end thermal-hydraulic phase of the accident whole calculation (i.e. core, primary and secondary system) is performed with CESAR module and the other modules are switched on gradually when the relevant physical conditions appear. When the core heatup takes place, the CESAR thermal-hydraulics in the downcomer, lower plenum and core is replaced by DIVA module and only rest of primary (and secondary) system is henceforth analyzed by CESAR module. Pre-defined basic configuration of the core and lower part of reactor vessel, which represents arrangement of typical PWR, is assumed in DIVA. Typically, the core region is split into arbitrary number of radial rings and axial layers. The same axial power profile has to be considered in all rings. Contrary to DIVA the CESAR nodalisation is very flexible and thus applicable to “arbitrary” thermal-hydraulic systems. All above mentioned points imply certain limitations in modelling of Paks cleaning tank arrangement. The two most important are as follows:

1) Uniform axial power profile has to be considered in all fuel assemblies (at least during the core heatup and degradation phase);
2) After start of DIVA module (i.e. after start of core heatup and replacement of CESAR core nodalisation by DIVA nodalisation), it was not possible to model direct bypass from the lower holes of fuel assemblies to outlet nozzle from cleaning tank.

To cope with this, two successive calculations performed:

- CESAR stand alone up to start of fuel heatup (analysis focused on precise modelling of the front-end thermal hydraulics of the cleaning tank);
- All modules in coupled mode since the beginning of the fuel heatup (450 °C, start of DIVA module). Since it was further not possible to model flow bypass through lower holes in shroud to outlet from cleaning tank, the original fine CESAR nodalisation was
replaced by simplified one and “proper” inlet boundary conditions was used instead (analysis focused on fuel degradation).

The 30 fuel assemblies were split into 5 groups (6 assemblies in each group) based on their location from the centre of cleaning tank:

1\textsuperscript{st} (central) ring: assemblies No. 1 to 6,
2\textsuperscript{nd} ring: assemblies No. 7 to 12,
3\textsuperscript{rd} ring: assemblies No. 13 to 18,
4\textsuperscript{th} ring: assemblies No. 19 to 25,
5\textsuperscript{th} (outer) ring: assemblies No. 26 to 30.

The burnup of fuel assemblies within one group was relatively close each to other. The only one exception was assembly No. 12 in 2\textsuperscript{nd} group. Average power within each group was proportional to average burnup. This defines radial power distribution in 5 parallel core channels (CESAR) and in 5 core rings, respectively. The total “core” power was 241 kW.

**TABLE I:** Radial power distribution (from centre to outer part)

<table>
<thead>
<tr>
<th>ring No.</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
</tr>
</thead>
<tbody>
<tr>
<td>average power of one FA [W]</td>
<td>7159</td>
<td>9006</td>
<td>6694</td>
<td>8837</td>
<td>8719</td>
</tr>
</tbody>
</table>

Uniform axial distribution was used for both, CESAR and DIVA modules. This distribution represents an average calculated from all fuel assemblies.

**TABLE II:** Axial power distribution (from bottom to top)

<table>
<thead>
<tr>
<th>axial mesh</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
<th>7</th>
<th>8</th>
<th>9</th>
<th>10</th>
</tr>
</thead>
<tbody>
<tr>
<td>relative power</td>
<td>$6.372 \times 10^{-2}$</td>
<td>$9.126 \times 10^{-2}$</td>
<td>$1.001 \times 10^{-1}$</td>
<td>$1.151 \times 10^{-1}$</td>
<td>$1.200 \times 10^{-1}$</td>
<td>$1.199 \times 10^{-1}$</td>
<td>$1.159 \times 10^{-1}$</td>
<td>$1.081 \times 10^{-1}$</td>
<td>$9.575 \times 10^{-2}$</td>
<td>$7.028 \times 10^{-2}$</td>
</tr>
</tbody>
</table>

Only simplified approach was used in modelling of fission product release and transport. Generic data were used for initial FP inventory instead of specific and the same inventory was considered in all fuel assemblies. Individual isotopes were not modelled, only chemical elements.

**Cesar nodalisation**

The CESAR nodalisation is shown on Fig. 1. Main features are as follows:

- Core (heated volume) represents only part inside fuel shrouds;
- Bypass represents part outside shrouds;
- 5 parallel core channels was modelled;
- 14 axial nodes used in total (2 lower, 2 upper and 10 heated nodes containing fuel);
- Lower head with BC (representing inlet from cooling pump) and water inlet to fuel assemblies;
- Upper head – outputs from fuel assemblies and bypass; degassing line on the top was modelled too;
CESAR nodalisation of cleaning tank

DIVA nodalisation of cleaning tank
- 10 horizontal junctions (holes in the shrouds, improperly positioned FA was not considered);
- Heat transfer through shrouds modelled;
- Outlet pipe and pressure BC.

The pressure head losses were tuned in order to get sufficient agreement (inlet/outlet temperature, mass-flow-rate) for cleaning and cooling regime.

**DIVA nodalisation:**

- Basic components: Lower plenum, core (5 radial rings, 19 axial meshes – 15 of them heated) and bypass;
- Urbanic-Heidrick model was used for cladding oxidation;
- DIVA is not able to model forced circulation cooling of lower part of tank (i.e. direct bypass through lower holes to outlet from tank; “once through” model have to be used instead;
- Because of once-through model, the inlet BC was chosen to keep constant water level ~ 1 m from the bottom of cleaning tank (i.e. applied inlet flow rate was chosen just to compensate water boil off from flooded part of FAs; real inlet flow rate was higher due to direct flow bypass through the lower holes of FAs to outlet nozzle from cleaning tank. Heat losses from the tank walls were artificially increased to compensate this effect.

**Results:**

The analysis started at t = 0 s with the transition from cleaning to cooling regime. This was modelled by lowering the prescribed inlet mass-flow-rate to cleaning tank (BC). This event triggered slow process of steady increase of coolant temperature in cleaning tank towards saturation (and later superheated) conditions. Since about 2000 s of the transient the coolant flow-rate through lower holes exceeded the flow through the fuel assemblies (Fig 2). The first, local boiling in FA occurred at about 9 500 s. Later, when the steam production exceeded the degassing flow, a steam bubble started to form bellow the CT upper head. At about 11 000 s (~ 3 h) the steadily expanded steam bubble reached the elevation of upper holes and the circulation through fuel assemblies was interrupted. The whole coolant flow from the cooling pump is realised through the lower holes with only minor effect on “core” cooling. Expanded steam bubble in upper part of cleaning tank (Fig. 5, 6) gradually depressed the water level in CT and displaced hot water from the stagnated volume of cleaning tank, what temporarily caused increase of water temperature at the outlet from CT (Fig. 3).

At about 14 000 s (~ 3 h, 45 min) the maximum cladding temperature reached 450 °C (start of DIVA) and later, at 19 311 s (~ 5 h, 22 min) the first cladding burst occurred in the 2nd ring with highest power (and thus highest temperature and gap pressure). Until t ~ 22000 s (~ 6 h) the clad burst gradually occurred in 3rd, 1th and finally in 4th ring. The fuel rods in last 5th outer ring remained intact till the end of the analysis and survived successfully also the quenching after opening of cleaning tank (the sudden drop of gap pressure on Fig. 10 is due to cool down). This prediction is in accordance with the “experimental” results.

The first clad burst was followed with almost immediate release of volatile fission products from the gap to coolant (Fig. 16) and hence to environment (Fig. 18). The code prediction is delayed about 10 minutes against “experiment” (Kr activity recorded at 5 h, 13 min). The thermally driven release of semi-volatile and especially low-volatile elements (Fig. 17) was significantly lower and delayed.
The time development of cladding oxidation profile is shown on Figs. 11-15. Maximum cladding oxidation was predicted in central part of upper, uncovered “core” region and reached 100%. In total there was produced 4.68 kg of hydrogen during the analysis (total oxidation represents 10.93 % of all Zr structures). Maximum cladding temperature during the accident was lower than 1400 °C (Fig. 9).

Finally it can be stated that the overall system behaviour was reasonably well predicted by the ASTEC code. However, certain code limitations were identified especially in the field of applicability to geometrical arrangements which differ significantly from typical LWR (PWR) configuration.

**TABLE III:** Chronology of main events in analysis

<table>
<thead>
<tr>
<th>event</th>
<th>time [s]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Transition from cleaning to cooling regime (16:40 h. real time)</td>
<td>0.</td>
</tr>
<tr>
<td>Local saturation in the fuel assembly</td>
<td>~ 9 500</td>
</tr>
<tr>
<td>Increase of PRZ water level 7 cm</td>
<td>10 700 – 12 100</td>
</tr>
<tr>
<td>Max. cladding temperature 450 °C (start of DIVA module)</td>
<td>14 014</td>
</tr>
<tr>
<td>Max. cladding temperature 800 °C</td>
<td>17 872</td>
</tr>
<tr>
<td>failure of the 1st fuel rod due to balooning</td>
<td>19 311</td>
</tr>
<tr>
<td>Start of intense Zr oxidation (production &gt; 2 g/s)</td>
<td>22 000</td>
</tr>
<tr>
<td>Failure of the last fuel rod before quenching*</td>
<td>22 019</td>
</tr>
<tr>
<td>Max. cladding temperature 1200 °C</td>
<td>24 930</td>
</tr>
<tr>
<td>Opening of cleaning tank and quenching the fuel assemblies</td>
<td>34 200</td>
</tr>
<tr>
<td>End of analysis</td>
<td>34 410</td>
</tr>
</tbody>
</table>

* the fuel assemblies in outer ring remained intact
Fig. 1. Pressures.

Fig. 2. Water massflows.
Fig. 3. Temperatures (1).

Fig. 4. Temperatures (2).
Fig. 5. Levels.

Fig. 6. Cleaning tank water mass.
Fig. 7. Hydrogen production.

Fig. 8. Hydrogen production rate.
Fig. 9. Maximum temperature in the core.

Fig. 10. Gap pressure.
Fig. 11. Oxid layer thickness (1st channel).

Fig. 12. Oxid layer thickness (2nd channel).
Fig. 13. Oxid layer thickness (3rd channel).

Fig. 14. Oxid layer thickness (4th channel).
Fig. 15. Oxid layer thickness (5th channel).

Fig. 16. Volatile and semi-volatile FP released from fuel.
**Fig. 17.** Low-volatile FP released from fuel.

**Fig. 18.** Volatile and semi-volatile FP released to environment.
Fig. 19. Low-volatile FP released to environment.

Fig. 20. Maximum cladding oxidation.
Fig. 21. Temperature field in cleaning tank
Fig. 22. Gas temperature in cleaning tank
Fig. 23. Material composition in cleaning tank
APPENDIX B/XIII.

THERMAL HYDRAULIC CALCULATIONS OF PAKS INCIDENT USING RELAP5/MOD3.2.2

Martin Vogel (Slovak Republic, VUJE)

OECD-IAEA Paks Fuel Project
I.1. Description of RELAP5/Mod3.2.2 computer code

For thermal hydraulic calculations of Paks incident RELAP5/Mod3.2.2 code was used. The RELAP5/Mod3.2.2 code has been developed for the analyses of light water reactor coolant systems during transients and postulated accidents. Code RELAP5/Mod3.2.2 code is based on a non-homogeneous and non-equilibrium model for the two-phase system that is solved by a fast, partially implicit numerical scheme to permit economical calculation of system transients. The code includes many generic component models from which general systems can be simulated. The component models include pumps, valves, pipes, heat releasing or absorbing structures, reactor point kinetics, electric heaters, jet pumps, turbines, separators, accumulators, and control system components. In addition, special process models are included for effects such as form loss, flow at an abrupt area change, branching, choked flow, boron tracking, and non-condensable gas transport. The detailed description of the code is listed in [1].

RELAP5 code validation is carried out within the framework of CAMP (Code Application and Maintenance Program) whose Slovakia (and also VUJE, a.s.) is a member. The sponsor and coordinator of this program is US NRC while the RELAP5 development and validation is performed by ISL company (Information Systems Laboratories, Inc.) and Pennstate University. Users of RELAP5 from more than 25 countries are further CAMP member and they are effectively participate on the code development and code application.

I.2. Description of the input model

Calculation model of cleaning tank with 30 pieces of fuel assemblies was developed. Nodalisation of the model is shown in Fig. 1.

The model is composed from interconnected simple volumes. Coolant inflow into tank is realized by connection no. 30. Type of this connection is ‘TMDPJUN’. By this connection, during the whole analyzed period, constant coolant mass flow 21 t/h with constant temperature 30°C is added into tank.

Volume under lower plate is created by one volume no.31 (‘BRANCH’ type). This component consists from one volume and 7 junctions. Six junctions connect volume under lower plate with assemblies and 7th junction represents by pass area created by incorrectly seating assemblies.
Fuel assemblies are lumped into 6 groups according to [2]. Assemblies groups are presented in Table 1. Each group of fuel assemblies is modelled by ‘PIPE’ type component (volumes 1 - 6). These volumes are split in axial direction onto 13 sections. Fuel part of assemblies is divided onto 10 axial sections. These fuel sections are represented by heat structures 1001 - 1006. In each group of fuel assemblies, decay heat and axial profile of decay heat is defined according to [2]. Total decay heat for each assemblies group is presented in Table 1. Graphical form of axial profile of decay heat distribution for each assemblies group is presented in Fig. 2.

Table 1 Description of assemblies groups

<table>
<thead>
<tr>
<th>Group number</th>
<th>Assembly number</th>
<th>Assembly kind</th>
<th>Total decay heat [kW]</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1 - 6</td>
<td>working FA</td>
<td>42.952</td>
</tr>
<tr>
<td>2</td>
<td>7 - 11</td>
<td>working FA</td>
<td>45.028</td>
</tr>
<tr>
<td>3</td>
<td>12</td>
<td>follower FA</td>
<td>8.154</td>
</tr>
<tr>
<td>4</td>
<td>13 - 18</td>
<td>follower FA</td>
<td>40.164</td>
</tr>
<tr>
<td>5</td>
<td>19 - 24</td>
<td>follower FA</td>
<td>53.021</td>
</tr>
</tbody>
</table>
Heat structures 1001 - 1006 represent fuel rods. In these structures there is assumed water-metal reaction. The shroud of assemblies is not modelled there. This assumption was done based on sensitivity studies. In case when these shrouds were modelled, strong instabilities occurred and it was not possible to reach decrease of water level. Based on this simplification, for radiation heat transfer, assemblies shroud were substituted with fuel rods and radiation heat transfer was assumed directly between fuel rods and inner tank wall.

Assemblies in groups 1 and 2 are working assemblies. There are perforations in lower and upper parts of assemblies. Perforations in lower part are modelled by junctions no. 41 and 42. Flow areas of these junctions are proportional to areas of real perforations. Loss coefficients for these perforations were set to 2.7 in both directions of flow.

Volume of cleaning tank above fuel assemblies is modelled by single volume no. 34 (‘BRANCH’ type). This component consists from one volume and 7 junctions. Six junctions connect outlet from assemblies to volume above assemblies and 7th junction connects volume no. 34 with volume no. 7, which represents volume between fuel assemblies and inner wall of cleaning tank.

Component no.35, connected into upper part of volume no. 34, represents air letdown valve, which is connected into volume no. 36 (‘TMDPVOL’ type). Volume no. 36 represents boundary condition, where is during whole analyzed period constant pressure 0.2 MPa. Flow area of the air letdown valve is proportional to diameter 10 mm. This valve opens when tank pressure exceeds 0.25 MPa.
Coolant flows from tank through volume no. 38 (‘PIPE’ type) into ‘TMDPVOL’ component no. 40, which represents boundary condition. There is constant pressure during whole analyzed period.

Tank heat losses are modeled by heat structures no. 1020 and 1021. Heat transfer coefficient at the inner side of these structures is calculated by RELAP code and depends on flow conditions. Heat transfer coefficient at the outer side is assumed to 1800 W/m²K. External pool temperature (30°C) is constant during the whole analyzed period.

I.3. The reference case

I.3.1. Initial and boundary conditions

Initial and boundary conditions for analysed cases were set in order to reach as good agreement with the real conditions during incident as possible.

In this chapter the reference case is showed. Initial and boundary conditions for this case were set as follows:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>By-pass flow area</td>
<td>120 cm²</td>
</tr>
<tr>
<td>Decay power of assemblies No. 1 - 6</td>
<td>42.952 kW</td>
</tr>
<tr>
<td>Decay power of assemblies No. 7 - 11</td>
<td>45.028 kW</td>
</tr>
<tr>
<td>Decay power of assembly No. 12</td>
<td>8.154 kW</td>
</tr>
<tr>
<td>Decay power of assemblies No. 13 - 18</td>
<td>40.164 kW</td>
</tr>
<tr>
<td>Decay power of assemblies No. 19 - 24</td>
<td>53.021 kW</td>
</tr>
<tr>
<td>Decay power of assemblies No. 25 - 30</td>
<td>52.312 kW</td>
</tr>
<tr>
<td>Total decay power</td>
<td>241.7 kW</td>
</tr>
<tr>
<td>Initial coolant temperature in container</td>
<td>56 °C</td>
</tr>
<tr>
<td>Initial container pressure</td>
<td>0.23 MPa</td>
</tr>
<tr>
<td>External pool temperature</td>
<td>30 °C</td>
</tr>
<tr>
<td>Temperature of inlet coolant</td>
<td>30 °C</td>
</tr>
<tr>
<td>Inlet mass flow</td>
<td>21 t/h</td>
</tr>
</tbody>
</table>

Axial profile of decay heat distribution for each assembly group was defined according to [2]. Graphical form of axial profile of decay heat distribution is presented in Fig. 2.

I.3.2. Results of the reference case

Process analysis begins at the moment of start of the intermediate cooling with coolant mass flow through tank of 21 t/h. Due to the lower flow rate of the intermediate cooling pump, the by-pass flow through the perforations in the assembly shroud and at the bottom of incorrectly seated assemblies became much more significant, than it was during the cleaning operation with high flow rate. Up to 90 % of water added into cleaning tank flows through these
perforations and through the by-pass. Cooling of fuel assemblies is not sufficient and heating up of the water leads to saturation state at the top of the cleaning tank at the 4500 s.
Water level in tank begins to decrease after 2 hours since start of intermediate cooling regime. Water level is stabilised at level 1.8 m from lower plate. This decreased water level leads to the temperature increase of the fuel assemblies. Due to its highest and peaked axial decay heat profile, maximum cladding temperature 1470 °C is found in assembly 12 at time 29700 s.
The air letdown valve opens at 9440 s, when tank pressure exceeds 0.25 MPa. Flow rate through this valve is approximately 5 g/s.
Chronological sequence of important events is presented in Table 2. The results in graphical form are presented in Fig. 3 to Fig. 9.

Table 2 Chronological sequence of important events

<table>
<thead>
<tr>
<th>Event</th>
<th>Time [s]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Start of cooling mode (mass flow 21 t/h)</td>
<td>0</td>
</tr>
<tr>
<td>Saturation in the cleaning tank</td>
<td>4500</td>
</tr>
<tr>
<td>Water level drop in the cleaning tank</td>
<td>7260</td>
</tr>
<tr>
<td>Air letdown valve opening</td>
<td>9440</td>
</tr>
<tr>
<td>Maximum cladding temperature exceeds 800°C</td>
<td>13900</td>
</tr>
<tr>
<td>Start of intense Zr oxidation</td>
<td>16400</td>
</tr>
<tr>
<td>Maximum cladding temperature exceeds 1200°C</td>
<td>20600</td>
</tr>
<tr>
<td>Maximum cladding temperature (1470 °C in assembly no.12)</td>
<td>29700</td>
</tr>
<tr>
<td>End of calculation</td>
<td>36000</td>
</tr>
</tbody>
</table>

**Fig. 3. Cleaning tank coolant outlet flow rate**
Fig. 4. Cleaning tank coolant outlet temperature

Fig. 5. Water level in the cleaning tank
Fig. 6. Max. cladding temperature in assemblies

Fig. 7. Inlet flow rate of assemblies
Fig. 8. By-pass flow rate due to incorrect seating assemblies

Fig. 9. Flow rate through the air letdown valve
I.4. Results of sensitivity studies

With developed model of cleaning tank, following sensitivity studies were performed.

In 1st case, by-pass area of incorrectly seated assemblies has been changed in range from 110 cm$^2$ to 130 cm$^2$. Other initial and boundary conditions were not changed. Smaller by-pass area leads to earlier beginning of water level decrease and vice versa. Water level and maximal cladding temperature behaviour as a result of by-pass area change is shown in Fig. 10 and Fig. 11.

Next sensitivity studies were performed with heat losses and radiation heat exchange variations. In case when heat losses and radiation heat exchange inside cleaning tank is assumed, lower maximal cladding temperature is reached and differences of cladding temperatures of single assemblies groups are smaller than in case when heat losses and radiation heat exchange is not assumed (Fig. 12, Fig. 13, Fig. 14).

Performed sensitivity study shows that such input parameters as bypass flow, heat losses and radiation heat exchange have relative strong influence on the results.

![Graph showing water level in the cleaning tank - by-pass area variation](image-url)
**Fig. 11. Maximal cladding temperature in assemblies - by-pass area variation**

**Fig. 12. Maximal cladding temperature in assemblies - heat losses and radiation exchange variation**
Fig. 13. Axial cladding temperature profile - radiation exchange and heat losses are assumed

Fig. 14. Axial cladding temperature profile - radiation exchange and heat losses are not assumed
REFERENCES

APPENDIX B/XIV.

ANALYSIS OF THE PAKS EVENT USING MELCOR
K.C. Wagner (USA, SNL)

OECD-IAEA Paks Fuel Project
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1. ANALYSIS OF THE PAKS EVENT USING MELCOR

This summary report describes the analysis of the Paks event using the MELCOR severe accident analysis code. The Paks event was a fuel degradation accident in a cleaning tank on the main floor of the reactor at the Hungarian Paks Nuclear Power Station. In the event, 30 assemblies in the cleaning tank overheated due to inadequate heat removal.

A MELCOR model was developed to represent the inner cleaning tank, the outer tank, the 30 assemblies within the tank, and the surrounding pool. The specified boundary conditions for the model included the tank inlet flow rate, the assembly decay heat, the hydrostatic pressure at the tank exit and degassing line, and the operation of the hydraulic lid locking system of cleaning tank. There were 11 standard assemblies in the tank with burn-ups of 10.8 MWd/kgU (6) and 26.7 MWd/kgU (5). 19 follower assemblies were also in the tank with burn-ups of 10.1 MWd/kgU (1), 21.1 MWd/kgU (6), 13.3 MWd/kgU (6) and 13.1 MWd/kgU (6). The total decay power of the 30 assemblies was 241.6 kW.

Some of the inlet flow leaked around misaligned assemblies at the base plate of the inner tank (i.e., FL-101) and bypassed the fuel assemblies. The leakage area was specified to be 120 cm$^2$ based on an analysis by Hermann Plank of Areva that was presented at the first workshop [Plank]. The leakage was due to incorrect seating of the assemblies into the lower support plate. The perforations at the top and bottom of the shroud of the standard assemblies were also modeled as leakage paths.

At the top of the inner tank, there was a 10 mm OD degassing line. In normal operation, the degassing line is full of water. It was assumed that water within the degassing line precluded vapor flow until the inner tank steam bubble was established and the tank started to pressurize. From the Paks database, the degassing line resistance corresponded to a length of 15 m and its elevation change was 13.1 m [Szabó].

The MELCOR model was subdivided into 5 rings to represent the 30 assemblies (see Figure ES-1).

- **Ring 1** – Assemblies 1-6, standard type assemblies with the lowest decay power but geometrically located at the center of cleaning tank.

- **Ring 2** – Assemblies 7-11, standard type assemblies with the high decay power and geometrically located in the middle ring of assemblies in the tank.

- **Ring 3** – Assembly 12, a follower type assembly with the highest decay power at the top of assembly (i.e., the location exposed to steam as the accident progresses) and geometrically located in the middle ring of assemblies in the tank.

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1 It was assumed that a 0.1 m$^2$ leakage hole opened at 9.5 hours when the tank lid was unlocked.

B/XIV-3
• Ring 4 – Assemblies 13-18, follower type assemblies with a top-skewed decay power profile, and geometrically located in the middle ring of assemblies in the tank.

• Ring 5 – Assemblies 19-30, follower type assemblies with a normal decay power profile, and geometrically located in the outer ring of assemblies in the tank.

The VVER fuel assembly is represented by 17 axial levels as shown on the left hand-side of Figure ES-1. The inner tank used the same axial nodalization as the assemblies (see Figure ES-2).

The decay heat profiles were specified to follow the axial profile of each group of assemblies. In particular, the follower assemblies in Rings 3 and 4 had a top-skewed power profile that represented their calculated decay heat profile (i.e., these follower assemblies were partially retracted into the lower plenum in the last cycle). The released activity was calculated using the non-dimensional release fractions for each nuclide class multiplied by the activity in each assembly.

MELCOR includes thermal radiation models for exchange between the fuel rods and the shroud as well as a small amount of axial radiative exchange between axial levels within an assembly. The outward facing shroud surfaces of the assemblies in Ring 5 radiate to the tank wall. A radiation network was specified for radiative exchange between outer shroud walls of the assemblies in the five rings. The exit nozzle of the assemblies also radiated upward to the tank lid. An axially dependent thermal radiation network was modeled between the outer wall of the inner tank to the inner wall of the outer tank.

The MELCOR CORSOR-Booth fission product release model was used and the volatile specie release coefficients were modified to match measurements from recent French experiments. The vapor pressure, compound form, and initial inventory default inputs were modified to represent Cs$_2$MoO$_4$ and CsI as the dominant compounds for the cesium and iodine releases. The initial fission product inventory in the gap between the fuel in cladding was estimated to be <0.1%, based on the calculated released fission product gas [Szabó].

1.1. Thermal-hydraulic Response

The simulation of the accident was initiated with the termination of the “B” cleaning process (i.e., the conditions at 4:40 pm on April 10, 2003). The main cooling pumps were stopped and the resulting residual cooling flow of fuel assemblies in the cleaning tank was provided by a low-flow submersible pump located in the decay pool. The inlet flow to the cleaning tank was initially 5.69 kg/s and 30°C. From the previous operation, the water temperature in the tank was 57°C (330 K). The calculation was terminated 0.5 hours after the tank lid was unlocked, which flooded the tank (i.e., 10 hours).

Following the shift to the low-flow submersible pump and the storage pool as the water source, relatively cold water (30°C) was introduced into the initially warm tank (57°C). 65% (i.e., the initial maximum) of the cold water flowed upward into the assemblies and was approximately
evenly distributed across the 30 assemblies (see Figure ES-3). The calculation assumed some of
the water leaked between baseplate holes and the assemblies (initially ~25%) and some leaked
through the perforations on the standard assembly inlet nozzles (initially ~10%). The cold water
cooled the bottom of the assemblies and increased the hydrostatic pressure drop through those
pathways relative to the leakage pathways. Hence, the calculated leakage flow across the
baseplate and through the standard assembly perforations steadily increased while the assembly
flow decreased. As the accident progressed, the calculated assembly flow dropped to 12% of the
inlet flow by 2 hours while the leakage flow increased to 88% (52% across the baseplate and
36% through the perforations in the standard assemblies).

As a result of the flow reduction into the assemblies, the calculated fluid temperature at the upper
portion of the assemblies steadily increased towards saturation conditions. At 2 hours into the
simulation, the water at the exit of Assembly 12 in Ring 3 started to boil. By 2.3 hours, the
assemblies in all 5 rings were boiling. As steam exited the assemblies, it created a steam bubble,
which filled the top of the tank and depressed the water level in the tank and in the assemblies
(see Figure ES-4). One of the few measurements available during the accident was the primary
circuit pressurizer level, which was hydraulically connected to the storage pool with the cleaning
tank. Between 2.3 to 2.6 hours into the accident (7:00-7:20 pm on April 10, 2003), the
pressurizer tank showed a 70 mm water level increase (i.e., an undiagnosed indication of water
displacement from the cleaning tank). As shown in Figure ES-4, the timing of the calculated
drop in the cleaning tank water level closely corresponded to the measured increase in the
pressurizer tank level.

The calculated level response shows an initial level depression to 1.2 m followed by a recovery
to 1.3-1.4 m (see Figure ES-4). Although the rate of steam production exceeded the degassing
flow, condensation of steam against the relatively cool inner tank wall reversed the level trend.
After 3.2 hours, non-condensable hydrogen from the zirconium-steam oxidation reaction started
to replace steam in the upper tank and helped stabilize the water level. The water level remained
between 1.3-1.4 m until 9.5 hours when the hydraulic latch on the lid was opened.

At a water level of 1.4 m, the top 60% of the fuel was uncovered. The steam flow due to boiling
below the 1.4 m water level was inadequate to cool the fuel. The top portions of all the
assemblies heated from 2.5 hours to 9.5 hours. The peak temperature response in each ring is
shown in Figure ES-5. Due to thermal radiative heat transfer, there was a temperature gradient
from the assemblies closest to the inner tank wall (#19 -#30) versus the other assemblies at the
center of the tank (#1 - #6). The peak temperatures of the assemblies in the middle ring (#7
through #18) lied between the inner and outer rings of fuel assemblies.

The peak cladding temperature was calculated to be 1743 K, which occurred in Ring 1
(Assemblies 1 through 6) just prior to the tank reflood. The corresponding peak temperatures
calculated for Rings 2 through 5 were 1540 K, 1672 K, 1667 K, and 1403 K, respectively. The
temperature response of the fuel was a complex function of the magnitude of the decay heat and
oxidation powers, the axial power profile, the magnitude of steam generation below the water
level in each assembly to drive oxidation, the radiative exchange to surrounding assemblies or
the tank wall, and the convective heat removal rate. The radiative heat losses to the tank wall
had a significant impact of the peak temperatures. For example, the heat losses to the tank wall
exceeded the decay power of the assemblies in Ring 5 after 6.1 hr. However, the fuel
temperatures in Ring 5 continued to increase due to radiative heating from the assemblies in the inner rings. Because the assemblies in Ring 1 were the furthest from the tank wall, they reached the highest temperature. In the middle ring of assemblies, Rings 3 and 4 had higher decay heat power in the region above the water level and therefore reached a higher temperature than the assemblies in Ring 2. Although Ring 3 (Assembly 12) had the highest decay power above the water level and initially heated the fastest, radiative exchange to the surrounding assemblies and low steam production below the water level (i.e., for continued oxidation) limited its peak temperature to below Ring 1.

1.2. Fission Product Release

The overall volatile fission product releases are shown in Figure ES-5 as a function of the peak cladding temperature (i.e., the instantaneous peak across all assemblies). The first failure of the fuel cladding and release of the gap fission products was calculated to occur at 4.0 hr in Assembly 12. The other gaps subsequently failed from 4.2 hr to 4.9 hr. The thermally driven fission product releases continued following the gap failures in each group of assemblies. After about 6 hr, the release increased more rapidly. Approximately half of the volatile xenon, cesium, and iodine releases occurred in the last 1.5 hours (i.e., between 8 hr to 9.5 hr).

Table ES-1 shows the comparison of the calculated and measured releases of a few key nuclides. Four groups of results are shown: (1) the nuclide initial inventory and half-life, (2) the measured tank release, (3) the analysis of the measured data by Hermann Plank of Areva Corporation, and (4) the calculated results using MELCOR. The calculated releases were 1.1% of the noble gases, 1.1% of the cesium, 0.8% of the iodine, 0.02% of the barium, and 0.03% of the cerium and lanthanum.

It is important to point out a key difference between the measurements and the MELCOR calculation. The plant measured the released activity from which the percentage released could be calculated as the release measurement divided by the calculated initial inventory. The released inventory measurement included a correction for decay from the measurement time to the start of the accident (i.e., the timing basis for the initial inventory calculation). In contrast, a fractional release was calculated by MELCOR for each assembly from which the released activity was calculated as the product of the release fraction times the calculated initial inventory. There was no attempt in the MELCOR activity calculations to account for daughter products, which likely has led to errors in the prediction of the activity of some nuclides. The change in inventory due to daughter products would require additional calculations of the key nuclide decay chains and their response relative to the measurements.

The data from the measurements of 8 representative nuclides were integrated for the following time periods after the accident: 1 day, 3 days, 7 days, 14 days, and 2 years. In general, the amount of released activity for the various nuclides increased with time. In the case of the noble gases, they were highly mobile and were all released to the chimney over the first three days. However, the nuclides that condensed in the water took additional time to flow out of the cleaning tank and into the pool (see Figure ES-7). In addition, some other elements (e.g. Cs, Ce,

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2 Figure ES-5 shows the overall fractional release before multiplying by the specific assembly activities. See Table ES-1 for calculated activity release.
and Pu) showed continuous release/dissolution from the fuel following the accident. Consequently, AEKI recommended integrating the release from the first two weeks of the incident to compare the calculated and measured release from the fuel during the incident. The MELCOR calculations only simulated the high-temperature thermally driven fission product releases from the fuel. MELCOR does not include models for the continued long-term release of fission products due to dissolution of the failed fuel, which was measured from many nuclides including cesium, cerium, and plutonium.

In general, the comparisons of the MELCOR results to the 14-day integrated tank measurements were relatively good. The temperature profile from the inner assemblies to the outer assemblies meant the calculated fission product releases were higher in Ring 1 versus Ring 5. The overall calculated and measured results show that approximately 1% of the volatile inventory (i.e., the noble gases, cesium, and iodine) was released during the high temperature portion of the accident, which is in good agreement with the 14-day data. The measured time-evolving release of some nuclides raises some uncertainty in the comparison of MELCOR to some nuclides (i.e., what is attributed to the transport delay from the tank from the high temperature accident versus continued low-temperature dissolution release). The comparison of the calculated and 14-day barium release data suggests higher releases than calculated. The cerium and lanthanum release models had good agreement with the 14-day integrated data.
Table ES-1. Comparison of the Calculated Releases to Plant Measurements and the Plank Analysis.

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>$^{133}$Xe</th>
<th>$^{131}$I</th>
<th>$^{134}$Cs</th>
<th>$^{137}$Cs</th>
<th>$^{140}$Ba</th>
<th>$^{144}$Ce</th>
<th>$^{141}$Ce</th>
</tr>
</thead>
<tbody>
<tr>
<td>Calculated Inventory (Bq)</td>
<td>5.560E+16</td>
<td>4.150E+16</td>
<td>5.690E+15</td>
<td>7.220E+15</td>
<td>1.160E+17</td>
<td>1.200E+17</td>
<td>1.650E+17</td>
</tr>
<tr>
<td>Nuclide Half-life</td>
<td>5.25 d</td>
<td>8.04 d</td>
<td>2.91 h</td>
<td>30.2 y</td>
<td>12.76 d</td>
<td>284.4 d</td>
<td>32.5 d</td>
</tr>
<tr>
<td>First 14 days $^{\text{Note A, C}}$ (Bq, %)</td>
<td>1.6% $^{\text{Note D}}$</td>
<td>1.0%</td>
<td>0.57%</td>
<td>0.35%</td>
<td>0.10%</td>
<td>0.060%</td>
<td>0.039%</td>
</tr>
<tr>
<td>Database file ($^{\text{Note B}}$)</td>
<td>\gamma\gamma\gamma\gamma\gamma\gamma\gamma (\PAKS2M2\activity\calculat) \gamma\gamma\gamma\gamma\gamma\gamma\gamma \gamma\gamma\gamma\gamma\gamma\gamma \gamma\gamma\gamma\gamma\gamma\gamma \gamma\gamma\gamma\gamma\gamma\gamma</td>
<td>cXe133c.txt</td>
<td>c131.txt</td>
<td>cCs134.txt</td>
<td>cCs137.txt</td>
<td>cBa140.txt</td>
<td>cCe144.txt</td>
</tr>
<tr>
<td>H. Plank Analysis $^{\text{Note B}}$</td>
<td>4.52E+14</td>
<td>1.52E+14</td>
<td>3.72E+13</td>
<td>3.34E+13</td>
<td>1.35E+14</td>
<td>3.62E+12</td>
<td>3.98E+12</td>
</tr>
<tr>
<td>Release, Inventory, and % Released</td>
<td>5.44E+16</td>
<td>3.84E+16</td>
<td>6.97E+15</td>
<td>7.15E+15</td>
<td>1.07E+17</td>
<td>1.60E+16</td>
<td>1.58E+17</td>
</tr>
<tr>
<td>% Released</td>
<td>0.83% $^{\text{Note E}}$</td>
<td>0.40%</td>
<td>0.53%</td>
<td>0.47%</td>
<td>0.13%</td>
<td>0.003%</td>
<td>0.0025%</td>
</tr>
<tr>
<td>MELCOR New Base Case</td>
<td>6.17E+14</td>
<td>3.14E+14</td>
<td>6.42E+13</td>
<td>8.60E+14</td>
<td>2.46E+13</td>
<td>3.97E+13</td>
<td>5.08E+13</td>
</tr>
<tr>
<td>Calculated Release (Bq, %)</td>
<td>1.1%</td>
<td>0.76%</td>
<td>1.1%</td>
<td>1.2%</td>
<td>0.021%</td>
<td>0.029%</td>
<td>0.031%</td>
</tr>
</tbody>
</table>

Notes:

A. Measured tank releases: E. Szabó, Z. Hózer, Cs. Győri, Gy. Hegyi, Database for the OECD-IAEA Paks Fuel Project, Version 1.0, AEK1-FRL-2006-408-01/02-M2, IAEA Contract No. RER9076-90930J, September 2006. The measured values represent the summation of the “corrected integrated releases” using the formula and linear methods in the database file. The 14-day measurements used summations over the first 3.2 time periods in the database files (14.2 days). The measurements started at 2:21 am, 6 minutes after the hydraulic lid locking system was opened.

C. The integrated release data cover not only the time of the incident, but several months of the wet storage in the pool. In some cases (e.g. noble gases), there was release only during the incident, but some other elements (e.g. Cs, Ce, Pu) showed continuous release/dissolution from the fuel during the wet storage period. AEKI recommended an integration of releases from the first two weeks of the incident to compare the calculated and measured release from the fuel during the incident.

D. The noble gas releases were based on chimney measurements through 6 days after the accident. The release of nearly all the noble gases was done by 3 days. The 14-day measurement corresponds to the integrated result through 6 days.

E. Plank’s analysis derived the Xe-133 activity from the coolant activity concentration. The approach probably did not include the whole noble gas release from the fuel (see Appendix B, Response #1).
Figure ES-1  Schematic of the MELCOR Model of the Paks-2 Fuel Assemblies.
Figure ES-2  Schematic of the MELCOR Model of the Paks-2 Cleaning Tank.
Figure ES-3  Comparison of the Calculated Mass Flowrate per Assembly for the 5 Rings in the MELCOR Model.

Figure ES-4  Comparison of the Calculated Cleaning Tank Liquid Level to the Measured Pressurizer Level Response.
**Figure ES-5** Calculated Peak Cladding Temperature in the 5 Rings.

**Figure ES-6** Calculated Overall Volatile Fission Product Releases and Peak Temperature.
Comparison of Measured Releases Over Time for Some Key Nuclides

Figure ES-7  Comparison of the Measured Releases for Some Representative Nuclides Over Time.