IAEA-TECDOC-1557

Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety:

PWR Vessel Internals

2007 Update

IAEA
International Atomic Energy Agency

June 2007

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FOREWORD

At present, there are over four hundred operational nuclear power plants (NPPs) in IAEA Member States. Operating experience has shown that effective control of the ageing degradation of the major NPP components (e.g. caused by unanticipated phenomena and by operating, maintenance or manufacturing errors) is one of the most important issues for plant safety and also plant life. Ageing in these NPPs must be therefore effectively managed to ensure the availability of design functions throughout the plant service life. From the safety perspective, this means controlling within acceptable limits the ageing degradation and wear-out of plant components important to safety so that adequate safety margins remain, i.e. integrity and functional capability in excess of normal operating requirements.

IAEA-TECDOC-1119 documents ageing assessment and management practices for PWR Reactor Vessel Internals (RVIs) that were current at the time of its finalization in 1997–1998. Safety significant operating events have occurred since the finalization of the TECDOC, e.g. irradiation assisted stress corrosion cracking (IASCC) of baffle-former bolts, which threatened the integrity of the vessel internals. In addition, concern of fretting wear of control rod guide tubes has been raised in Japan. These events led to new ageing management actions by both NPP operators and regulators. Therefore it was recognized that IAEA-TECDOC-1119 should be updated by incorporating those new events and their countermeasures.

The objective of this report is to update relevant sections of the existing IAEA-TECDOC-1119 in order to provide current ageing management guidance for PWR RVIs to all involved in the operation and regulation of PWRs and thus to help ensure PWR safety in IAEA Member States throughout their entire service life.

The IAEA officer responsible for this publication was T. Inagaki of the Division of Nuclear Installation Safety.

EDITORIAL NOTE

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1. INTRODUCTION

1.1. BACKGROUND

IAEA-TECDOC-1119 documented ageing assessment and management practices for PWR reactor vessel internals (RVIs) that were current at the time of its finalization in 1998. It concluded that while irradiation assisted stress corrosion cracking (IASCC) of PWR internals had not been observed for structural components globally so far, it may be of concern with time. After the issuance of the TECDOC, an inspection of baffle bolts in the United States discovered cracking in two of the four plants inspected. As the baffle and former assembly provides an interface between the core and the core barrel region and is important to safety because it provides a high concentration of the reactor coolant flow in the core region, IASCC on the baffle-former bolts is of safety concern. In addition, swelling (void formation), which was not addressed in IAEA-TECDOC-1119, could become an issue in PWR internals because of low displacement rates and increased temperature due to gamma heating. Concern of fretting wear of control rod guide tubes has also been raised in Japan. These events led to new ageing management actions by both NPP operators and regulators. Therefore it was recognized that IAEA-TECDOC-1119 should be updated by incorporating those new events and their countermeasures.

Basic requirements on NPP activities relevant to the management of ageing (maintenance, testing, examination and inspection of systems, structures and components (SSCs)) have been also updated and included in the IAEA Safety Requirements on the Safety of Nuclear Power Plants: Operation [1] and the associated Safety Guides on maintenance, surveillance and in-service inspection [2]. In addition the IAEA is preparing a new Safety Guide on ageing management which will provide key recommendations on managing ageing of SSCs important to safety.

1.2. OBJECTIVE

The objective of this report is to update the IAEA-TECDOC-1119 in order to provide current ageing management guidance for PWR RVIs to all involved in the operation and regulation of PWRs. The IAEA is preparing a new Safety Guide on ageing management which provides key recommendations on ageing management of SSCs important to safety. This report is consistent with the new Safety Guide and supplements it by providing specific information and guidance for ageing management of PWR RVIs.

IAEA-TECDOC 1119 is superseded and replaced with this report.

1.3. SCOPE

This report deals with age-related degradation and ageing management of pressurized water reactor vessel internals. It presents and discusses the requirements and methodologies utilized for the assessment and management of ageing of PWR RVIs. Pressurized heavy water reactor internals are not addressed in this report.

This report provides the technical basis for managing the ageing of pressurized water reactor (PWR) reactor pressure vessel internals (RVI) to ensure that the required safety and operational margins are maintained throughout the remainder of plant life. The scope of the report includes the following RVI components: upper material assembly, core support assembly and lower internals assembly.

The scope of this report is same as that of IAEA-TECDOC-1119. Significant operating experience and research and development results since the finalization of the IAEA-TECDOC-1119 in 1998 that have uncovered new safety issues and resulted in improved understanding of PWR RVIs ageing and in implementation of more effective ageing management actions have been added.

1.4. STRUCTURE

This report describes the RVI in Section 2, including an overall characterization of the design, importance to safety, materials and physical features of the RVI. In Section 3, the applicable design basis, codes, standards and regulations are addressed. Section 4 deals with operating conditions, Section 5 identifies dominant degradation mechanisms, sites, consequences, and significance of degradation mechanisms. Section 6 addresses the application of inspection technology to assess the condition of the RVI. Section 7 summarizes the current knowledge on service experience and related maintenance. Section 8 describes an ageing management programme for PWR RVI utilizing a systematic ageing management process and outlines relevant national and international ageing research.

This report retains the section numbering of IAEA-TECDOC-1119.

2. DESCRIPTION OF REACTOR VESSEL INTERNALS

Section 2.1 provides the overall system description of Western type PWR RVI and includes design features, applicable material specifications and differences among the various RVI components. Today's operating PWR RVI were mainly designed and manufactured by Westinghouse, Combustion Engineering, Inc., Babcock & Wilcox Company, Mitsubishi Heavy Industries, Ltd., Framatome, and Siemens/KWU. Section 2.2 provides the overall system description of Eastern type PWR RVI (WWER) and includes the main design features, applicable material specifications and differences among the WWER 440- and WWER 1000-type RVI components, designed by OKB Gidropress and manufactured by Izhora Works or Skoda.

2.1. WESTERN TYPE PWR RVIs

Fig. 1 shows the structural assembly grouping of a PWR RVI system which is more or less the same for Westinghouse, Combustion Engineering, Babcock & Wilcox, Mitsubishi Heavy Industries, Ltd., and Framatome. Siemens/KWU used a similar design at earlier plants, except that there was no instrumentation, mounted from bottom of the RPV. In the most advanced Konvoi plants (Fig. 2), Siemens/KWU made use of a welded core shroud assembly.

The core barrel provides a boundary for the reactor coolant. The primary coolant enters the reactor vessel via the inlet nozzles; impinges on the side of the core barrel and is directed downward through the annulus formed by the gap between the outside diameter of the core barrel and the inside diameter of the reactor pressure vessel. The primary coolant flow then enters the lower plenum area between the bottom of the lower support plate and the reactor pressure vessel bottom head and is redirected upward through the core. After passing through the core, the coolant enters the upper core support region and then proceeds radially outward through the reactor pressure vessel outlet nozzles.

The reactor vessel internals consist of two structural assembly groupings, the upper and lower internals assemblies. The upper internals assembly consists of all the internals' components above the core. The lower internals assembly consists of all the remaining internals components. The fuel assemblies rest on the lower support structure of the lower internals assembly which transmits the resulting loads to the core barrel and, hence to the core barrel flange, which rests on the reactor pressure vessel flange. The upper internals assembly is attached under the reactor pressure vessel head flange. The vendors designation of reactor vessel internals subcomponents is not necessarily the same for all PWR nuclear steam supply systems (NSSS). The upper internals assembly is removed during each refuelling operation. The lower internals assembly is only removed during reactor pressure vessel in-service inspection (ISI) period (in most cases each 10 years).

During the design process, the importance to safety and not importance to safety of RVI components are determined by analysis and in some cases testing and are documented in the safety analysis report (SAR). RVI important to safety are those components that are needed for the performance of safety functions and those components whose failure could prevent any of the safety functions. The safety functions considered are those associated with supporting the core, maintaining reactivity control, assuring core cooling, and assuring instrumentation availability.

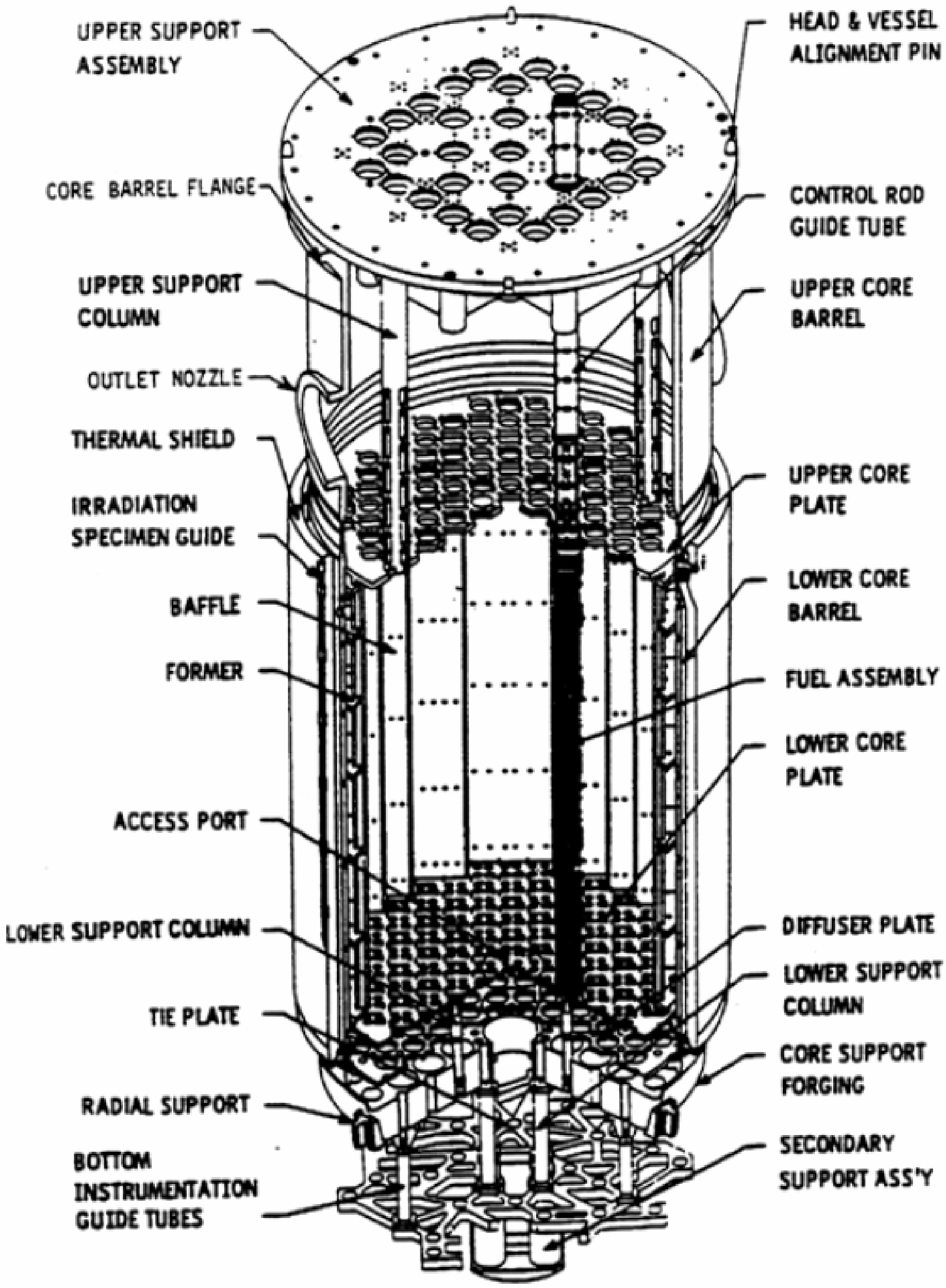


Fig. 1. Structural assembly grouping of PWR RVI.

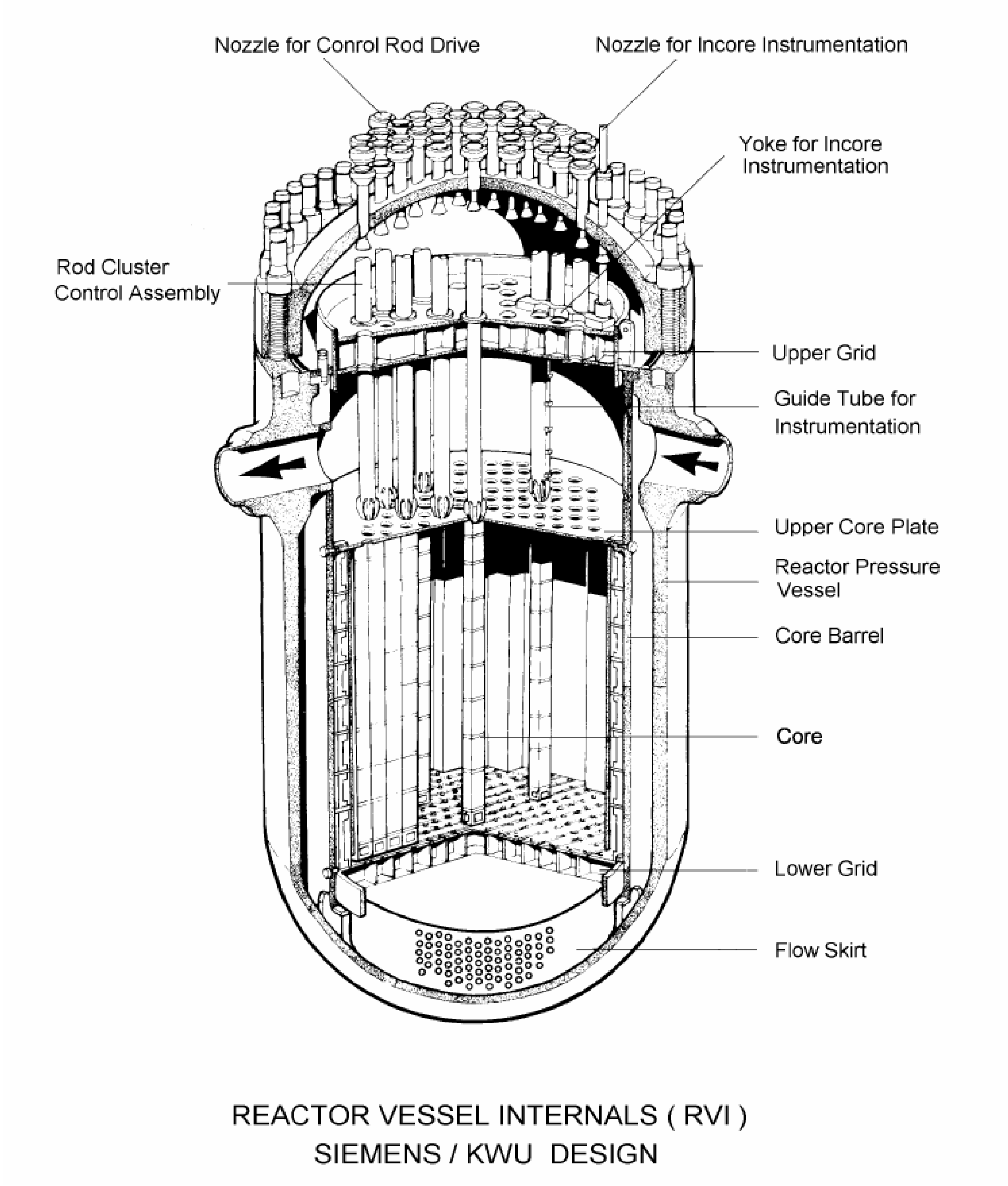


Fig. 2. Siemens/KWU RVI design (Konvoi plant).

The following RVI components are needed in support of the above safety functions:

(a) Maintaining the core support structure and/or cooling geometry

- Lower core plate
- Lower support forging or casting plate
- Lower support columns
- Core barrel
- Radial keys and clevis inserts
- Baffle and former assembly
- Core barrel outlet nozzle
- Secondary core support
- Diffuser plate
- Upper support plate assembly
- Upper core plate
- Upper support columns and guide tubes
- Internals holddown spring
- Head/vessel alignment pins
- Clevis inserts.

(b) Maintaining reactivity control and control rod insertion time

- Rod cluster control assembly (RCCA) control rods
- RCCA guide tubes
- Upper core plate alignment pins and clevis inserts
- Driverods.

(c) Assuring instrumentation availability

- Bottom Mounted Instrumentation (BMI) columns and flux thimbles
- Upper instrumentation column.

Failure of the following RVI components could prevent the above safety functions:

- Neutron panels/thermal shield
- Head cooling spray nozzles
- Mixing device.

2.1.1. RVI consistent parts

The reactor core is positioned and supported by the lower internals and upper internals assembly. The individual fuel assemblies are positioned by fuel pins in the lower and upper core plates. These pins control the orientation of the core with respect to the lower internals and upper internals. The lower internals are aligned with the upper internals by the upper core plate alignment pins and secondarily by the head/vessel alignment pins. The lower internals are orientated to the vessel by the lower radial keys and by the head/vessel alignment pins. Thus, the core is aligned with the vessel by a number of interfacing components.

RVI constituent parts are classified as either core support structures (shown as CS in the following paragraph) or internals structures (shown as IS in the following paragraphs). A core support structure provides support and restraint of the core. The internals structures are all other structures within the reactor pressure vessel that are not core support structures, fuel assemblies, blanket assemblies, control assemblies, or instrumentation.

Core Support Structure (CS)

Lower core plate and fuel alignment pins

The lower core plate (LCP) is important to safety because it positions and supports the core and provides a metered control of reactor coolant flow into each fuel assembly.

The LCP is located near the bottom of the lower support assembly, inside the core barrel, and above the lower support forging. There are fuel pins, typically two per fuel assembly, attached to the core plate, that position the fuel assemblies. The fuel assemblies are positioned over the four flow holes per assembly which control the amount of flow entering each fuel assembly. The AISI Type 304 stainless steel perforated plate is circular and is bolted at the periphery to a ring welded to the ID of the core barrel. The span of the plate is supported by lower support columns that are attached at their lower end to the lower support plate. At the core plate centre, a removable plate is provided for access to the lower head region of the vessel.

Lower support forging or casting

The lower support forging or casting is important to safety because it provides support for the core by reacting against LCP loads transmitted through the lower support columns. The plate must direct coolant flow from the lower head plenum to the core region. Also, access to the lower head region of the vessel during field assembly and ISI is provided via a removable plate.

Lower support columns

The lower support columns are important to safety because they support the LCP and transmit the loads from the LCP to the much thicker and stiffer lower support forging. Some lower support columns also serve as a guide for the neutron flux thimbles.

The lower support columns separate the LCP and the lower support. The columns react against the core loads acting on the LCP and transmit these loads to the lower support. The columns are attached with threaded fasteners to the LCP and a threaded joint to the lower support.

Core barrel

The core barrel is important to safety, because its primary function is to support the core. Lateral support for the core is provided at the upper and lower core plate locations and at intermediate positions during a seismic and LOCA event. During a seismic and LOCA event, the core may impact the baffle/former assembly that is supported by the core barrel. In addition to the support requirement, the core barrel needs to provide a passageway for the reactor coolant flow. It directs the reactor coolant flow to the core, and after leaving the core it directs the flow to the outlet nozzles.

The core rests directly on the LCP that is ultimately supported by the core barrel. The LCP is attached at its periphery to the core barrel ID and supported by lower support columns that are attached to the lower support forging. The lower support forging is welded at its edge to the bottom end of the core barrel.

Radial keys and clevis inserts

The radial keys and clevis inserts are important to safety because they restrain large transverse motions of the core barrel but at the same time allow unrestricted radial and axial thermal expansions.

The lower core barrel is restrained laterally and torsionally by these uniformly spaced keys. The radial keys, along with the matching clevis inserts, are designed to limit the tangential motion between the lower end of the core barrel and the vessel. At assembly, as the internals are lowered into the vessel, the keys engage the keyways of the inserts in the axial direction. With this design, the core barrel is provided with a support at the farthest extremity and may be viewed as a beam fixed at the top and guided at the bottom. With the radial key and inserts, the radial and axial expansions of the core barrel are accommodated but circumferential movement (i.e. rotation) of the core barrel is restricted. The radial keys are attached to the core barrel at the lower support forging level.

Baffle and former assembly

The baffle and former assembly (Fig. 3) is important to safety because it provides a high concentration of the reactor coolant flow in the core region. The baffle and former assembly is made up of vertical plates called baffles and horizontal support plates called formers. The baffle plates are bolted to the formers by the baffle/former bolts, and the formers are attached to the core barrel ID by the barrel/former bolts.

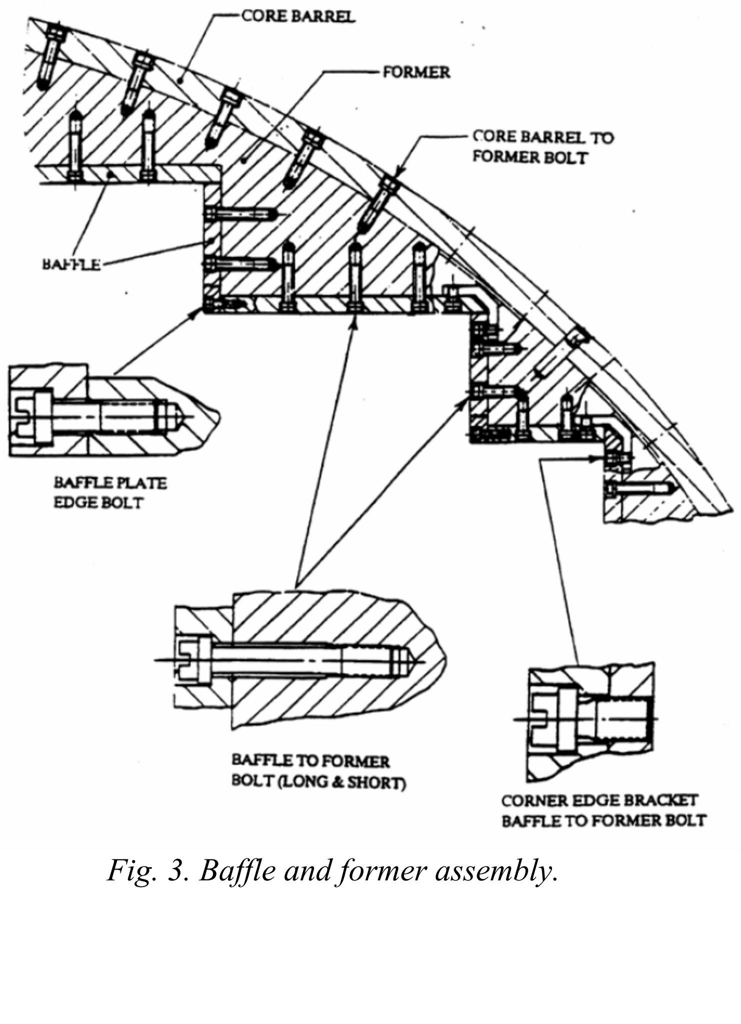


Fig. 3. Baffle and former assembly.

The baffle and former assembly forms the interface between the core and the core barrel. The baffles provide a barrier between the core and the former region so that a high concentration of flow in the core region can be maintained.

Upper support plate assembly

The upper support plate assembly is safety important because it supports the guide tubes and the core. The upper support plate assembly is a rigid base that positions and supports the guide tubes and the upper support columns that, in turn, position and support the UCP. The upper support plate also positions and supports the thermocouple columns and guides. There are three models of upper support plate assemblies: (1) a deep beam, (2) top hat, and (3) an inverted top hat.

Upper core plate

The upper core plate (UCP) is safety important because it interacts with the core by positioning the fuel assemblies and the guide tubes. The UCP positions the upper ends of the fuel assemblies and the lower ends of the control rod guide tubes, thus serving as the transition member for the control rods during entry and retraction from the fuel assemblies. It also controls coolant flow when it exits from the fuel assemblies and serves as a boundary between the core and upper plenum.

Upper support column

The upper support columns are safety important because they interact with the core (fuel assemblies). They perform the following functions:

– Preload fuel assembly and react fuel assembly forces

– Serve as separation members for the upper support plate and UCP in formation of the core outlet plenum
– Position, guide, and support the thermocouples for core outlet water temperature measurement including housing flow-mixing devices.

Internals Structure (IS)

Core barrel outlet nozzle

The core barrel outlet nozzles are safety important because they direct the reactor coolant after it leaves the core, radially outward through the reactor vessel outlet nozzles. The core barrel outlet nozzles are located in the upper portion of the core barrel directly below the flange and are attached to the core barrel, each in line with a vessel outlet nozzle face and the vessel outlet nozzle land. The nozzles extend radially from the core barrel to the ID of the vessel and are customized during manufacture to minimize this gap. The size of the gap reduces during heat-up and may go to a small interference at operating temperatures. This component is classified as an internal structure, since it does not provide support for the core.

Neutron panels/thermal shield

Neutron panels or thermal shields are not considered to be safety important because they do not support or interact with the core. Additional neutron shielding of the reactor vessel is provided in the active core region by neutron panels or thermal shields that are attached to the outside of the core barrel. Neutron panels are attached to the OD of the core barrel at strategically located positions to reduce the fluence on the reactor vessel welds. The thermal shield design provides shielding for the complete 360-degree circumferential sector. It is fastened with bolts and dowels below the outlet nozzles and also near the lower portion of the core barrel with flexures. At some plants, the thermal shield has been removed.

Secondary core support

The secondary core support is considered safety important because it maintains integrity of the core following a postulated failure of the core barrel. The function of the secondary core support, following a postulated failure and downward displacement of the core barrel subassembly, is to:

– Absorb a portion of the energy generated by the displacement and limit the force imposed on the vessel
– Transmit and distribute the vertical load of the core to the reactor vessel
– Limit the displacement to prevent withdrawal of the control rods from the core
– Limit the displacement to prevent loss of alignment of the core with the upper core support to allow the control rods to scram.

Bottom-mounted in-core instrumentation columns and flux thimbles

Bottom-mounted in-core instrumentation columns and flux thimbles are not considered to be safety important because they do not support or interact with the core. The functions of these columns are to provide a path for the flux thimbles into the core from the bottom of the vessel and to protect the flux thimbles during the operation of the reactor. There are two types of bottom mounted in-core instrumentation columns. The cruciform columns extend through the flow holes of the lower support forging and attach to the bottom of the LCP. The standard guide columns line up with the lower support columns and are bolted to the bottom side of the lower support. These are line drilled to provide a flux thimble path, and the lower end of the column is counterbored to fit over the vessel conduit penetration. This provides an uninterrupted, protected path for flux thimbles entering the reactor core.

The flux thimble is a long, slender stainless steel sealed tube that passes through the vessel penetration, through the lower internals assembly, and finally extends to the top of the fuel assembly. The flux thimble provides a path for the neutron flux detector into the core and is subjected to reactor coolant pressure and temperature on the outside surface and to atmospheric conditions on the inside.

The flux thimbles remain stationary during reactor operation, with the bullet end of the thimbles positioned slightly above the top of the active fuel. For refuelling, the thimbles are retracted to a point where the bullet tip is below the LCP. For the removal of the lower internals assembly, the flux thimbles are pulled out further until the bullet tip is outside of the reactor vessel.

Diffuser plate

The diffuser plate is not considered to be safety important because it is not the primary source of flow uniformity. To enhance flow uniformity entering the LCP, some plants employ an additional orifice plate called a diffuser plate. This plate is clamped in place by the lower support columns between the LCP and lower support plate.

Head cooling spray nozzles

Head cooling spray nozzles are not considered to be safety important because they do not support or interact with the core. Head cooling spray nozzles are used to adjust the upper plenum coolant temperature by allowing bypass flow at the vessel inlet temperature from the vessel/core barrel downcomer region to flow directly into the upper head plenum. Different designs evolved, so the exact configuration would depend on the production date.

Guide tube

The guide tubes (GTs) are safety important because they control the path of the control rods in and out of the core. Guide tubes are bolted from the top of the upper support plate and are supported at their lower end to the UCP with spring-type pins. They perform the following functions:

– Provide a straight low-friction path for the control rods into or out of the fuel assemblies.
– Provide sufficient protection for the control rods when they are withdrawn from the fuel elements to prevent damage due to parallel and lateral coolant flow.
– Provide a convenient, safe storage place for the control rod drive lines during refuelling.

Upper instrumentation column

The upper instrumentation columns are not considered safety important. The upper instrumentation columns provide a passageway and cross-flow protection to the conduits that, in turn, house the thermocouples. The thermocouples are inserted into the top of the upper instrumentation columns and are routed down through the inside of various support columns. The ends of the thermocouples protrude below the upper support columns so that the temperature of the coolant exiting the fuel assemblies can be measured.

Mixing device

The mixing device is not considered safety important. Mixing devices are used with thermocouples to enhance the temperature reading at the core outlet just above the UCP. Mixing devices are not used in all plant designs.

The mixing devices are cast cylinders with four vanes cast on the inside. They are located individually on the UCP or full penetration-welded to the upper support columns at all thermocouple locations. They sustain the same loads as the upper support columns except when individually attached to the UCP.

Interfacing components

The interfacing components listed in this section and following sections are considered safety important because they basically interact with components that support the core. The general requirements of the interfacing components are to orient adjacent components with respect to each other and/or provide support for an adjacent component. These components are the lower internals assembly, the upper internals assembly, the fuel and driveline, or the reactor vessel.

The UCP alignment pins position the UCP with respect to the lower internals assembly and provide lateral support to the lower end of the upper internals assembly. The holddown spring supports the upper internals assembly and holds the lower internals assembly down. The head and vessel alignment pins align the lower internals assembly and the upper internals assembly with the vessel. The radial support inserts provide a support surface for the radial support keys.

Upper core plate alignment pin

The UCP alignment pins locate the UCP laterally with respect to the lower internals assembly. The pins must laterally support the UCP so that the plate is free to expand radially and move axially during differential thermal expansions between the upper internals and the core barrel.

The UCP alignment pins are the interfacing components between the UCP and the core barrel. The UCP alignment pins are shrunk-fit and welded into the core barrel and the core barrel bearing pad. The gap sizes between the alignment pins and the matching inserts are customized.

Hold down spring

The hold down spring provides a preload to limit the axial motion of the upper and lower internals assemblies and to prevent lift-off of the core barrel flange from the vessel ledge. The spring preload also reduces the lateral motion of the upper support plate flange and the core barrel flange. The hold down spring is required to be designed for operating condition loads.

The hold down spring, which is a circumferential spring with an essentially rectangular cross-section, is located between the flanges of the upper support plate and the core barrel. The hold down spring is preloaded by a compressive force when the reactor vessel head is clamped in place with the reactor vessel closure studs and nuts. Therefore, the hold down spring is an interfacing component between the upper internals assembly and the lower internals assembly.

Head and vessel alignment pins

The head and vessel alignment pins align the upper and lower internals assemblies with respect to the vessel. The head-vessel alignment pins are located at the outside periphery of the core barrel flange at the four major axes. A portion of the pin extends below the core barrel flange and engages pockets in the reactor vessel to provide alignment of the lower internals assembly with respect to the vessel.

Similarly, a portion of the pin extends above the flange and aligns the upper internals assembly with respect to the vessel. This portion of the pin engages pockets in the reactor vessel head, thus establishing an alignment of lower internals, reactor vessel, upper internals, and reactor vessel head. Minimal clearance is maintained between the pins and the engagement pockets to ensure functional alignment and to allow ease of assembly. The clearances are designed to prevent thermal loads in the pins during temperature excursions and to reduce the stress in the pins during horizontal loading of the upper internals.

Radial keys and clevis inserts

The radial keys and clevis inserts provide the interface between the lower internals and the vessel.

Driveline components

The driveline components are the drive rod and the control rods. The control rods are identified as the RCCA (rod cluster control assembly). The drive rod and RCCA make up the interface between the drive mechanism on the reactor pressure vessel head and the guide tubes and fuel.

2.1.2. Materials

Various product forms are used in the manufacture of reactor vessel internals assemblies subcomponents. These various product forms include plates, forgings, rolled rings, and castings of austenitic stainless steel. The reactor vessel internals assemblies subcomponents are joined by either welding or bolting the subcomponents together to form a complete assembly. Stainless steels have been used in the manufacture of reactor vessel internals because of their corrosion resistance, toughness, ductility, strength and fatigue characteristics in pressurized water reactor environment. In western type reactor vessel internals, AISI Type 304 and 347 stainless steel or of corresponding designations are used in various product forms in all of the larger internals components, as for example, core barrels, support columns, core barrel flange, core plates, core support plates, hold down springs, guide tubes and core baffle-former assemblies. Fastener or bolts are fabricated from 304, 347, 316 cold worked, 316Ti cold worked stainless steel or Alloy X-750 material. In some western type reactor vessel internals, Alloy 600 may be used. All materials employed in reactor vessel internals have established fabrication and service histories.

In France, core barrel, core baffles and formers are fabricated with the plates of Type 304L steel with a controlled nitrogen content in solution annealed condition. The former plates have a final thickness of some 40 mm. The core barrel is made with plates of a thickness close to 50 mm, rolled and welded.

In the UK, core support structures are constructed from Type 304 austenitic stainless steel solution annealed where required core support structure plates and forgings are welded together using a mechanized tungsten inert gas (TIG) process. The filler metal used is of type SFA 5.9 Class ER 308L.

In Germany, the materials which are employed are niobium stabilized austenitic stainless steels, solution annealed. Materials most relevant to the internal structures are steels X6CrNiNb 18-10 (1.4550) and X6CrNiMoTi 17-12-2 (1.4571).

Threaded structural fasteners and bolts from cold worked Type 316 austenitic stainless steel are used in French and British reactors internals. In Germany, a cold worked type 1.4571 austenitic stainless steel (similar to AISI Type 316 but titanium stabilized) is employed for fasteners.

In French reactor internals, several other materials are also used. They are very similar to those of the US reactor internals (forged, cast stainless steel, etc.). Their specifications are identical to those of similar materials employed in other parts of the reactor, but there is a specific requirement for the cobalt content, which has to be limited to 0.1% for large enough pieces (RCCM G2400).

It is however worthwhile to mention alloy Inconel X-750, which is used for the support pins of the control rods mechanisms and in former German design for baffle bolts and fuel alignment pins as well. This alloy is solution annealed at 1080°C for 1 hour and is exposed to a precipitation hardening heat treatment for 20 h at 700°C.

Typical reactor vessel internals materials and their chemical composition are given in Tables 1–4.

TABLE 1. TYPICAL RVI MATERIALS

Component	Standards and specifications		
	US type reactors (ASTM, ASME)	French type reactors (RCC-M, AFNOR)	German type reactors (KTA 3204)
Upper support forging	SA-182 Grade F304	Z2CN 19-10 N controlled (M 3302)	X6CrNiNb18-10 (1.4550)
Hold down spring	SA-182 Grade F403 (mod) or SA-182, Grade F304	Z2CN 19-10 N controlled (M 3301)	Inconel X-750
Core barrel nozzles	SA-182, Grade F304	Z2CN 19-10 N controlled (M 3301)	X6CrNiNb18-10 (1.4550)
Lower support forging	SA-182, Grade F304	Z2CN 19-10 N controlled (M 3302)	X6CrNiNb18-10 (1.4550)
Radial keys	SA-182, Grade F304	Z2CN 19-10 N controlled (M 3301)	X6CrNiNb18-10 (1.4550)
Radial keys/hard facing	SA-182, Grade F304/Stellite	Z2CN 19-10 N controlled (M 3301)	Alloy 600/Stellite 6 or 1.4550/ hard faced
Core barrel	SA-240, Type 304	Z2CN 19-10 N controlled (M 3310)	X6CrNiNb18-10 (1.4550)
Baffles & formers	SA-240, Type 304	Z2CN 19-10 N controlled (M 3310)	X6CrNiNb18-10 (1.4550)
Lower core plate	SA-240, Type 304	Z2CN 19-10 N controlled (M 3310)	X6CrNiNb18-10 (1.4550)
Neutron panels/thermal shield	SA-240, Type 304	Z2CN 19-10 N controlled (M 3310)	X6CrNiNb18-10 (1.4550)
BMI tie plates	SA-240, Type 304	Z2CN 19-10 N controlled (M 3310)	–
Flow distribution plate	–	–	X6CrNiNb18-10 (1.4550)
Upper support column	SA-479, Type 304	Z2CN 19-10 N controlled (M 3301)	X6CrNiNb18-10 Tube (1.4550)
Ucp alignment pins/hardfacing	SA-479, Type 304 / Stellite	Z2CN 19-10 N controlled (M 3301)	X6CrNiMoTi 17-12-2 (1.4571) hard faced
Guide tube cards	SA-479, Type 304	Z2CN 19-10 N controlled (M 3301)	–
Irradiation specimen guide	SA-479, Type 304	Z2CN 19-10 N controlled (M 3301)	X6CrNiNb18-10 (1.4550)
Head cooling nozzles	SA-479, Type 304	Z2CN 19-10 N controlled (M 3301)	X6CrNiNb18-10 (1.4550)
BMI columns	SA-479, Grade F304	Z2CN 19-10 N controlled (M 3301)	–
Secondary core support	SA-479, Type 304	Z2CN 19-10 N controlled (M 3301)	–
Upper instrumentation column	SA-213, Grade TP304LN	Z2CN 19-10 N controlled (M 3304)	X6CrNiNb18-10 (1.4550)
Upper & lower core plate fuel pins	SA-193, Type 316 (cold worked)	Z6CND17-12 or Z2CND17-12 or Z2CND17-12 N controlled, all cold worked (M3308)	X6CrNiMoTi 17-12-2 (1.4571) cold worked and hard faced Inconel X-750
Upper support column fastener	SA-193, Type 316 (cold worked)	Z6CND17-12 or Z2CND17-12 or Z2CND17-12 N controlled, all cold worked (M3308)	X6CrNiMoTi 17-12-2 cold worked (1.4571)

TABLE 1. TYPICAL RVI MATERIALS (CONT'D)

Component	Standards and specifications		
	US type reactors (ASTM, ASME)	French type reactors (RCC-M, AFNOR)	German type reactors (KTA 3204)
Baffle-barrel-former fastener	SA-193, Type 316 (cold worked)	Z6CND17-12 or Z2CND17-12 or Z2CND17-12 N controlled, all cold worked (M3308)	X6CrNiMoTi 17-12-2 cold worked (1.4571)
Neutron panel bolts	SA-193, Type 316 (cold worked)	Z6CND17-12 or Z2CND17-12 or Z2CND17-12 N controlled, all cold worked (M3308)	X6CrNiMoTi 17-12-2 cold worked (1.4571)
Guide tube holddown bolts	SA-193, Type 316 (cold worked)	Z6CND17-12 or Z2CND17-12 or Z2CND17-12 N controlled, all cold worked (M3308)	X6CrNiMoTi 17-12-2 cold worked (1.4571)
BMI bolts	SA-193, Type 316 (cold worked)	Z6CND17-12 or Z2CND17-12 or Z2CND17-12 N controlled, all cold worked (M3308)	–
BMI thimble tubes	SA-213, Type 316 (cold worked), SB-167 (Alloy 600)	NC15Fe Alloy 600 (M 4102)	–
Guide tube support pins & nuts	SA-193, Type 316 (cold worked) or W Spec A637C01 (X-750)	Inconel X 750 (M 4104)	X6CrNiNb18-10 (1.4550) hard faced
Irradiation specimen guide spring	SB-637, UNS N07750 (X-750)	Inconel X 750 (M 4104)	–
Clevis insert fastener	SB-637, UNS N07750 (X-750)	Inconel X 750 (M 4104)	–
Flow mixing device	SA-351 GR CF8	Z3CN20-09M (M 3405)	X6CrNiNb18-10 (1.4550)
Lower support casting	SA-351 GR CF8	Z3CN20-09M (M 3405)	–
Clevis insert hardfacing	SB-166 (Ni-Cr-Fe) Annealed/Stellite	–	–
Weld metal 308 & 3081	ASME SCII Part C SFA 5.9	SFA 5.9 Class ER 308L	SFA 5.9 Class ER 308L

TABLE 2. CHEMICAL COMPOSITION OF RVI MATERIALS — AUSTENITIC STAINLESS STEELS

Material	C	Mn	Si	S	P	Ni	Cr	Mo	Nb	Ti	Co	Cu	N	B
SA-182 Grade F 304	= 0.08	≤2.0	≤1.0	≤ 0.03	≤0.040	8.0–11.0	18.0–20.0	–	–	–	≤0.05	–	–	
SA-240 Type 304	= 0.08	≤2.0	≤0.75	≤0.03	≤0.045	8.0–10.5	18.0–20.0	–	–		≤0.05		≤0.10	
SA-479 Type 304	= 0.08	≤2.0	≤1.0	≤0.03	≤0.045	8.0–10.5	18.0–20.0	–	–		≤0.05		≤0.10	
SA-213 Grade TP304LN	= 0.035	≤2.0	≤0.75	≤0.03	≤0.040	8.0–11.0	18.0–20.0	–	–		≤0.05		0.10–0.16	
Z2CN 19-10 N controlled (M 3301 – M 3303 –M 3304– M 3307–M 3310) 304L	≤0.035	≤2.0	≤1.0	≤0.03	≤0.040	9.0–10.0	18.5–20.0	–	–	–	≤0.10	≤100	≤0.08	≤0.0018
Z3CN 18-10 N controlled (M 3302) 304L	≤0.040	≤2.0	≤1.0	≤0.03	≤0.040	9.0–11.0	18.5–20.0	–	–	–	≤ 0.10	≤100	≤0.08	≤0.0018
X6CrNiNb 18-10 (1.4550)	≤0.040	≤2.0	≤1.0	≤0.02	≤0.035	9.0–12.0	17.0–19.0	–	≤ 0.65	–	≤ 0.20	–	≤0.08	
SA-213 Grade TP316 cold worked	= 0.08	≤2.0	≤1.0	≤0.03	≤0.040	11.0–14.0	16.0–18.0	–	–					
SA-193 cold worked Type 316	0.04–0.080	≤2.0	≤0.75	≤0.03	≤0.045	10.0–14.0	16.0–18.0	2.0–3.0	–	–	≤ 0.25		≤0.10	–
Z6CND17-12 cold worked (M 3308) 316	0.03–0.080	≤2.0	≤1.0	≤0.03	≤0.035	10.0–14.0	16.0–18.0	2.25–3.00	–	–	≤ 0.20	≤1.0	–	–
Z2CND17-12 cold worked N controlled (M 3308) 316	≤0.035	≤2.0	≤1.0	≤0.03	≤0.035	11.5–12.5	17.0–18.2	2.25–2.75	–	–	≤ 0.10	≤1.0	≤0.08	–
Z2CND17-12 cold worked (M 3308) 316	≤0.030	≤2.0	≤1.0	≤0.03	≤0.040	10.0–14.0	16.0–19.0	2.25–2.75	–	–	≤ 0.10	≤1.0	≤0.08	–
X6CrNiMoTi 17-12-2 cold worked (1.4571)	≤0.060	≤2.0	≤1.0	≤0.02	≤0.035	10.5–13.5	16.5–18.5	2.0 – 2.5	–	≤ 0.7	≤ 0.20	–	–	–
Z3CN 20-09M N controlled (M 3302) 304L	≤0.040	≤2.0	≤1.0	≤0.03	≤0.040	9.0–11.0	18.5–20.0	–	–	–	≤ 0.10	≤100	≤0.08	≤0.0018
SA-351 Grade CF8	≤0.080	≤1.5	≤2.0	≤0.04	≤0.040	8.0–12.0	17.0–21.0	= 0.50	–	–	–	–	–	–

TABLE 3. CHEMICAL COMPOSITION OF RVI MATERIALS — WELDS

Material	C	Mn	Si	S	P	Ni	Cr	Mo	Nb	Ti	Co	Cu	Others
Weld SFA 5.4 /5.9 Class ER 308L	≤0.030	≤2.5	≤1.0	≤0.03	≤0.030	9.0–12.0	18.0–20.0	–	–	–	≤0.20	≤1.0	Ferrite content 5–15%
Weld (1.4576)	≤0.035	≤2.0	≤1.0	≤0.03	≤0.035	11.5–12.5	17.0–18.2	2.252.75	–	–	≤0.20	≤1.0	

TABLE 4. CHEMICAL COMPOSITION OF RVI MATERIALS — NI BASED ALLOYS

Material	C	Mn	Si	S	P	Ni	Cr	Fe	Nb	Ti	Co	Cu	Al
SB-166 Alloy N06600	≤0.15	≤1.0	≤0.5	≤0.015		≥72	14.0–17.0	6.0–10				≤0.5	
NC15Fe Alloy 600 (M-4102)	≤0.10	≤1.0	≤0.5	≤0.015	≤0.025	≥72	14.0–17.0	6.0–10		≤ 0.5	≤0.2	≤0.5	≤0.5
SB-637 Alloy N07750 X-750	≤0.080	≤1.0	≤0.5	≤0.01		≥70	14.0–17.0	5.0–9.0	0.7–1.2	2.25–2.75	≤1.0	≤0.5	0.4–1.0
Inconel X-750 (M-4104)	≤0.080	≤1.0	≤0.5	≤0.01	≤0.01	≥70	14.0–17.0	5.0–9.0	0.7–1.2	2.25–2.75	≤0.2	≤0.3	

2.2. WWER TYPE RVI

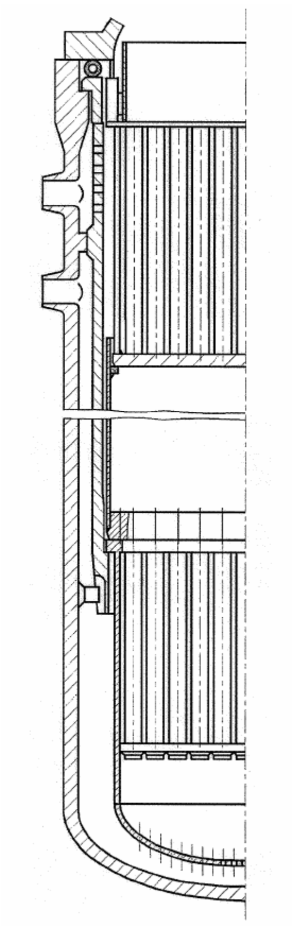
The reactor internals function is to support the core, to hold the fuel assemblies in place, to direct coolant flow, to hold and protect control rods in normal operation conditions and accident conditions. The reactor internals are designed to ensure cooling of the fuel, to ensure the movement of control rods under all operating conditions including accidents (up to maximum DBA with superimposed safe shutdown earthquake loads) and facilitate removal of the fuel and of the internals proper following an accident.

Hot and cold leg RPV nozzles are located in WWER reactors at two different elevations; therefore a horizontal seal separates the cold and hot legs. After passing through the core, the hot coolant enters the hot leg nozzles through a perforated part in the top section of the core barrel.

An overall view of the design of WWER-440 and WWER-1000 RVI is given in Fig. 4. For more details, see [3].

The main components of the WWER RVIs are the core barrel, the core shroud (core basket for WWER-440) at the level of the core and the block of guide tubes. These components are fixed together and to the reactor vessel in a way that allows their withdrawal, inspection, and partial repair as well as inspection of the reactor pressure vessel inner surface.

The WWER RVIs are manufactured, assembled and installed in line with requirements of the respective standards and quality control and assurance procedures. RVIs are tested at the manufacturer using vessel and core mock-up as well as during operation following an inspection (control rod movement).



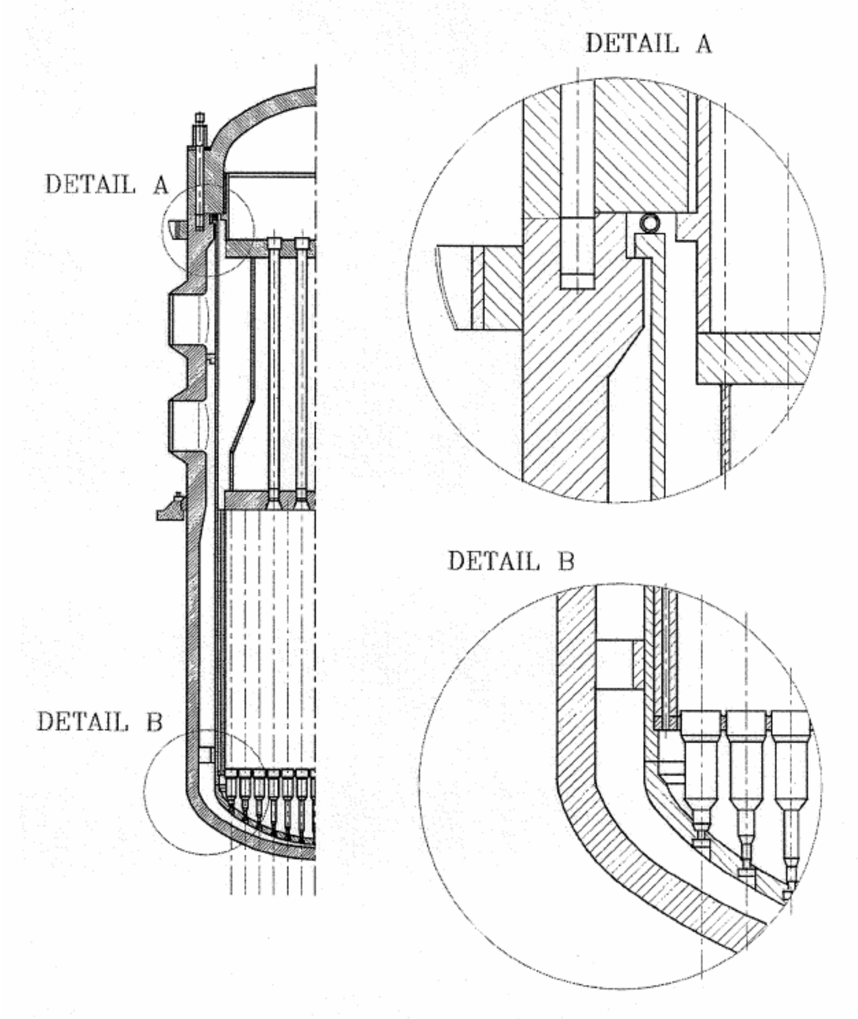


Fig. 4(a). WWER-440 RVI. Fig. 4 (b). WWER-1000 RVI.

2.2.1. RVI constituent parts

Core barrel

The core barrel and its support structure (the core barrel bottom for WWER-440 and support tubes and distance grid for WWER-1000) are important to safety, because their primary functions are to confine, fix and support the shroud (basket) and provide a passageway for the reactor coolant flow. They direct the reactor coolant flow to the core, and after leaving the core, to the outlet nozzles. In addition, the WWER-1000 core barrel support structure provides proper horizontal and vertical positioning of the fuel assembly end and supports the core.

The core barrel keys are important to safety because they restrain large transverse motions of the core barrel while allowing unrestricted radial and axial thermal expansions.

The core barrel supports the core shroud, the block of guide tubes for the drive rods of the control and protection system and separates the cold leg from the hot leg.

The core barrel of WWER-440 is a vertical cylinder with maximum diameter of 3,370 mm and height of 10,960 mm made of seven cylindrical rings welded together. The wall thickness of the cylindrical rings is between 50 and 80 mm.

The bottom of the core barrel consists of an upper forged lattice 150 mm thick and a lower lattice spacer 50 mm thick. 37 tubes and a vertical cylinder connect both lattices.

Guide tubes of these emergency control assemblies, which are welded to the upper lattice, are located inside the tubes at the periphery. They contain the fuel containing part of the emergency control assemblies when they are in the bottom position.

The vertical cylinder has a maximum diameter of 3,085 mm and a height of about 2,945 mm. The wall thickness of the three cylindrical rings is as a maximum 52.5 mm.

Orifice plates in upper lattice of core barrel bottom create additional resistance of the channels redistributing coolant flow. This resistance improves hydraulic characteristics of assemblies, thus making the core less susceptible to any changes (decrease) of coolant flow during transients.

The orifice plate which rests on positioners along the periphery of core barrel cylindrical part is fixed by 12 studs against uplift (another 16 studs were added to these vessels in 1989).

On the bottom of each emergency control assembly opening are installed the guide tubes and over them the protection tubes. The tubes are fitted into the openings and welded along their periphery.

The guide and guide tubes on the bottom are protected at their side and from below by a perforated stainless steel sheet shaped as a truncated cone with its broad part up (thickness of approx. 10 mm).

The core barrel of WWER-1000 is a vertical cylinder with elliptic perforated bottom (close to the WWER-440 core barrel does not have this item). The core barrel height is 10,510 mm, its outer diameter is 3,670 mm, the wall thickness of the cylindrical part is 65 mm. The lower part of the core barrel has supports installed inside as perforated tubes (with narrow 3 mm wide slits). The upper ends of these support tubes are fixed into a distance grid. Their lower ends are fixed into the perforated bottom of the core barrel.

In its upper part, the core barrel is fixed by three elastic tube elements placed between the RPV cover and the barrel flange and by keys at the barrel flanges. During reactor heatup, the middle part of the barrel locks against the RPV separating belt due to thermal expansion.

In its lower part, the core barrel is fixed by keys welded to the RPV cylindrical shell.

Core basket (WWER-440)

The core basket is important to safety because it provides a reduction of neutron flux to the RPV, and protects the integrity of the fuel assemblies in the event of pressure differences inside the RVI.

The core basket support plate is important to safety because it provides proper horizontal and vertical positioning of the fuel assembly end and supports the core.

The core basket, intended to contain the core, consists of a 300 mm thick plate at the bottom and a cylindrical part consisting of three rings with a wall thickness of 30 - 40 mm. The rings are welded one to another to the bottom plate. The inside surface of the basket is shaped to match the hexagonal boundary of the core. The basket bottom is in fact a support plate of the fuel assemblies. It has 312 openings for the fuel assemblies and 37 hexagonal holes for the fuel assemblies of the emergency control assemblies. At the upper part of the basket a limiting ring is welded which covers the heads of the peripheral row of operating fuel assemblies and serves as a support of the block of guide tubes. In addition, along the entire height of the core periphery there is a reinforced ring made of shaped partitions which is intended to decrease neutron flux deformation in the fuel assemblies of the external row (the peripheral fuel assemblies) by narrowing the water reflector.

The core basket is erected on core barrel bottom and fixed to it by a plug with three studs of a diameter of 120 mm.

Core shroud (WWER-1000)

The core shroud is considered to be safety important because it provides a reduction of neutron flux to the RPV, and ensures integrity of the fuel assemblies in case pressure differences inside the RVI.

The core shroud in WWER-1000, acting simultaneously as a thermal shield, consists of forged cylindrical shells. The shells are bolted together and their relative position fixed by pins. There are vertical channels inside the shell walls and circumferential grooves on the outside surface to facilitate cooling of the shroud metal. The height of the WWER-1000 core shroud is 4,070 mm and its external diameter 3,485 mm. The inside surface of the shroud is shaped to match the hexagonal boundary of the core. In the original design, irradiation surveillance containers with RPV materials are placed on the upper edge of the top shell.

In its upper part, the shroud is fastened by keys on the inside surface of core barrel. In its lower part it is fastened by three pins on the core barrel distance grid and held down to it by six threaded tubes. The upper and lower fixing of the shroud restrains motions due to coolant flow while allowing unrestricted radial and axial thermal expansions with respect to the core barrel.

Block of guide tubes (WWER-440)

The block of guide tubes is important to safety because it protects fuel assemblies from lift off, and the emergency control assemblies, the drive rods of the control and protection system, and the small diameter tubes of the reactor instrumentation system against coolant flow dynamic effects.

The block of guide tubes of the WWER-440 reactors is held down by the RPV head (the so-called upper block), leans against the upper part (the heads) of the fuel assemblies and therefore prevents displacement of the core, the core basket and the core barrel bottom in all operating conditions (spring load, spring blocks in the upper part, fuel assemblies heads springs). The block of guide tubes protects the fuel assemblies, the emergency control assemblies, the drive rods of the control and protection system and the small diameter tubes of the reactor instrumentation system against coolant flow effects.

The WWER-440 block of guide tubes consists of upper and lower round plates with penetrations for coolant and for the guide tubes of the emergency control assemblies. The top and bottom plates are welded together with nine cylindrical rings and interconnected by 37 protective tubes. The bottom one serves as a guide for connection with the basket, while the top one serves as a support structure for the spring blocks.

The bottom part of the block of guide tubes presses against springs of the fuel assemblies. The upper part of the block of guide tubes consists of a top screen, rings and a girder to which the bundles of the cladding temperature monitoring detectors are attached.

Block of guide tubes (WWER-1000)

Guide tubes are important to safety because they protect the control rods from the dynamic effects of the coolant flow and consequently from becoming stuck.

The lower support plate is important to safety because it provides positioning for the fuel assemblies and hence, is one of the components assuring core integrity.

The block of guide tubes of the WWER-1000 reactors is a welded structure consisting of three plates connected by guide tubes, tubes for in-reactor instrumentation and shells. Control rods move inside guide tubes. The block of guide tubes structure maintains the fuel assemblies in the required position (horizontally and vertically), holds them down and protects the control rods from the dynamic effects of the coolant flow and consequently from becoming stuck.

The lower support plate is in contact with top heads of the fuel assemblies (spring loaded). It is perforated to assure coolant flow to the upper plenum. Perforated upper and middle plates and slots in the upper shell support flange provide for coolant circulation under the reactor head. The lower and middle plates are connected by a perforated shell.

Azimuthal positioning is assured with the help of three keys welded to the upper shell and corresponding slots in the core barrel, as well as with the help of six keys welded to the barrel and corresponding slots in the lower plate of the block of guide tubes. Positioning of the block of guide tubes allows unrestricted radial and axial thermal expansions with respect to the core barrel and reactor head.

The overall dimensions of the block of guide tubes are: height, 8,292 mm, external diameter of lower plate, 3,550 mm.

In the original design the surveillance specimens of RPV materials to monitor thermal ageing are attached to the inside surface of the upper shell.

2.2.2. Material

The general information for western type PWR RVIs given in Section 2.1.2 also applies to WWER-type RVIs.

The main structural material used for RVIs in both WWER-440 and WWER-1000 reactors is titanium stabilized austenitic stainless steel 08Ch18N10T (equivalent to A-321). In addition to this austenitic stainless steel and its niobium stabilized welds, precipitation hardened nickel based alloy ChN35VT is also used. This material is used for studs and is tungsten alloyed.

Regarding structural RVI materials, the corresponding code is PNAE G-7-002-87 [4], and national Russian standards GOST 5632, GOST 23304, and GOST 2246.

The specifications for WWER RVI materials are given in Tables 5–7.

TABLE 5. TYPICAL RVI MATERIALS

Component	Specification	Note
Core barrel		
Core barrel vessel, spacing grid, core barrel bottom, caps of supports, displacer	08Ch18N10T	Sheets, sheet stamped blanks
Keys, nuts, washers, plugs, pins	08Ch18N10T	Bar and forgings
Supports, tail pieces of supports, tubes	08Ch18N10T	Seamless tubes of improved quality
Studs	ChN35VT-VD	Bar, vacuum arc refined Ni based alloy
Core shroud		
Upper ring, middle ring, lower ring	08Ch18N10T	Forgings
Spacing grids	08Ch18N10T	Sheets
Studs, nuts, washers, reducers, tail pieces	08Ch18N10T	Bars and forgings
Tubes	08Ch18N10T	Seamless, high quality
Block of guide tubes		
Shells, slabs, cones	08Ch18N10T	Sheets
Grids, discs, flanges, sleeves	08Ch18N10T	Bars, forgings
Flanges, rods, tips, keys, bolts, pins	08Ch18N10T	Bars, forgings, enhanced mechanical properties
Tubes	08Ch18N10T	Seamless, high quality
Welding materials		
Argon arc welding	Sv-04Ch19N11M3	Welding wire
Submerged automatic welding	Sv-04Ch19N11M3 + flux 0F-6 or FC-17	Welding wire
Electroslag welding	Sv-04Ch19N11M3 + flux 0F-6	Welding wire
Manual arc welding	EN-400/10T EN-400/10V	Electrode rod, grade Sv-04Ch19N11M3
Wear resistant cladding	CN-6L	Electrode rod, grade Sv-04Ch19N9M3

. Table 6. Chemical Composition of RVI Base and Bolting Materials

Material	C	Mn	Si	S	P	Ni	Cr	Ti	Al	W	B	Fe	Note
08Ch18N10T	≤0.08	1.0–2.0		≤0.020	≤0.035	9.0–11.0	17.0–19.0	≥5C ≤0.6	–	–		base	GOST 5632
ChN35VT(VD)	≤0.12	1.0–2.0	≤ 0.6	≤0.02 ≤0.01	≤0.030 ≤0 .025	34–38 34–36	14.0-16.0	1.1–1.5	–	2.8–3.5		base	GOST 5632 GOST 23304

Note: Under Material, 'VD' indicates a vacuum arc remelted material.

TABLE 7. CHEMICAL COMPOSITION OF RVI WELD MATERIALS

Material	C	Mn	Si	S	P	Ni	Cr	Mo	Other	Note
EA-400/10T	≤0.10	1.15–3.10	≤0.60	≤0.025	≤0.030	9.0–12.0	16.8–19.0	2.0–3.5	V: 0.3–0.75	
Sv-04Ch19N11M3	≤0.06	1.0–2.0	≤0.60	≤0.018	≤0.025	10.0–12.0	18.0–20.0	2.0–3.0	–	GOST 2246

Note: Under Material, 'Sv' refers to wire type filler metal for inert gas shielded or submerged arc welding and 'EA' denotes a covered electrode.

3. DESIGN BASIS, CODES, STANDARDS AND REGULATIONS

Before the development of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code [5] requirements specifically applicable to RVIs, the design of RVIs was based on criteria specific to each NSSS vendor. However, most western NSSS vendors used Section III of the ASME Boiler and Pressure Vessel Code, Subsection NB as a guideline for the development and establishment of reactor vessel internals system design criteria. Allowable stresses were established consistent with structural components. In the USA, the use of Subsection NB of the ASME Section III received concurrence from the United States Nuclear Regulatory Commission (USNRC). The methodology used for the establishment of RVI system design criteria was approved by the ASME Code. Using the methodology, the ASME developed Subsection NG to the ASME Boiler and Pressure Vessel Code, Section III specific to reactor vessel internals. Basically the same design basis applies for PWR RVI in western countries throughout the world. In France the applicable standard is RCC-M [6]. In Germany, the applicable standard is KTA 3204 [7]. In Japan the applicable standard is Ministry of Trade and Industry (MITI) notification 501 [8] and the Japan Society of Mechanical Engineers (JSME) Code on Design and Construction for Nuclear Power Plants, JSME SNC1-2001 [9], which is based on the ASME Code. In the Russian Federation, the applicable codes and standards are listed below in Section 3.5 (these codes and standards have been also adopted in most other countries operating WWER reactors).

3.1. REQUIREMENTS IN THE UNITED STATES OF AMERICA

Part 50 of the Code of Federal Regulations, Title 10 (10CFR50) [10] regulates construction of nuclear power plants. Section III of the ASME Code is the industry standard for construction of nuclear power plant facilities, while Section XI of the ASME Boiler and Pressure Vessel Code prescribes in-service inspection requirements, including inspection and evaluation of defects.

Reactor internals design fabrication and installations are covered by rules given in Section III of the ASME Boiler and Pressure Vessel Code, Subsection NG, Core Support Structures. Core support structures are those structures or parts of structures, which are designed to provide direct support or restraint of the core (fuel and blanket assemblies), within the reactor pressure vessel. Structures, which support or restrain the core only after the postulated failure of the core support structures are considered to be reactor internals structures. Before Subsection NG was published, Subsection NB of the ASME Code was used as a guideline for the development of vendor-specific internals system design criteria. The rules for reactor internals design are covered in Article NG-3000.

The rules for reactor internals materials are covered in Article NG-2000. The majority of reactor internals are fabricated using austenitic stainless, both wrought and castings and nickel based alloys (Alloy X-750, Alloy 600, etc.). Fabrication and installation of reactor internals are covered in Article 4000. Pre-service inspection is addressed in both Articles NG-2000 and NG-4000.

Although a large number of PWR reactor internals were designed and fabricated prior to the publication of Subsection NG, the design philosophies of the NSSS vendors throughout the western world were such that the intent of Subsection NG was met.

In the USA in addition to the requirements of the ASME Code, Section III, Subsection NG, a limited number of regulatory guides and bulletins are relevant to reactor vessel internals components assessment and management of ageing.

ASME Boiler and Pressure Vessel Code Section III, Subsection NG

ASME Section III, Subsection NG, Boiler and Pressure Vessel Code, Article NG-3000 which is divided into three subsections covers the design of reactor vessel internals. The three subsections of Article NG-3000 are:

– NG-3100 General Design
– NG-3200 Design by Analysis
– NG-3300 Core Support Structural Design.

Subarticle NG-3100 deals with Loading Conditions specified by the Owner (or his agent) in the form of an Equipment Specification. The equipment specification identifies the Design Loading in terms of Design Pressure Difference, Design Temperature, Design Mechanical Loads and Design Stress Intensity Values. The Equipment Specification identifies the Design and Operating Conditions.

Subarticle NG-3200 deals with the stresses and stress limits which must be considered for the analysis of the component. The reactor vessel internals are designed to withstand steady state and fluctuating loads produced under handling, normal operating transient and accident conditions. The equipment specification identifies the operating conditions. In the 1974 Edition of Subsection NG, there are four categories entitled:

– Normal Conditions
– Upset Conditions
– Emergency Conditions and
– Faulted Conditions.

Later code editions clarified this nomenclature but basically retained the same stress allowables. The corresponding new categories are:

– Service Level A
– Service Level B
– Service Level C
– Service Level D.

Subarticle NG-3300 gives the general design requirements for core support structures. The design of core support structures must meet the requirements of NG-3100 and NG-3200. However, if there is a conflict between NG-3200 and NG-3300, the requirements of NG-3300 shall govern.

System hot functional test

Reactor vessel internals are required to undergo pre-operation testing under hot conditions. Nuclear Regulatory Guide 1.20 establishes guidelines for the pre-operation assessment programme. The system hot functional test is done at the plant site and follows the primary system hydrostatic test. The reactor vessel internals are installed in the reactor vessel without fuel. The coolant temperatures are established by pump heating. Hot functional tests are conducted only once during the plant life. The hot functional test is conducted at elevated temperature and at a flow rate, which is greater than during normal plant operation due to the absence of fuel assembly resistance. The hot functional tests consist of operation with all loops pumps for a minimum of ten days. The hot functional tests ensure that the flow-induced load cycling (vibration) of the reactor vessel internals will be well into the high cycle range of their material fatigue design curves, thus providing assurance that the high cycle fatigue usage of the reactor vessel internals will be low throughout the plant design life.

ASME Boiler and Pressure Vessel Code Section XI

Section XI of the ASME Boiler and Pressure Code [5] is the standard for operation and inservice inspection of nuclear power plant facilities. Examination Category B-N-3 of Section XI, Subsection IWB, provides requirements for the visual examination of removable core support structures. These requirements refer to the relevant conditions defined in IWB-3520.4 which include loose, missing, cracked, or fractured parts, bolting, or fasteners. Inspection of baffle former bolts would fall under these inspection requirements.

3.2. REQUIREMENTS IN FRANCE

In France, the parts of reactor internals supporting the core and maintaining the fuel elements are classified as ES parts and are designed, constructed and inspected following the rules of the Subsection G “Equipements internes du réacteur” of the RCC-M code [6]. The subsection G of the RCC-M is similar to the Subsection NG of the ASME code as described in the previous section.

Allowed materials are given in Subsection G 2000. Rules for design are in Subsection G 3000. Construction and welding are done following Subsection G 4000 and inspection and controls following Subsection G 5000.

All the design analyses have to be updated periodically to take into account different changes or the monitoring of results such as plant transients, the fuel programme, the primary flow rate. All new R&D results have also to be considered, including computer code performance improvements.

Consequently different studies have been engaged on: vibration behavior, fatigue analysis of baffles and barrels, analysis of baffle bolts, accidental conditions for different parts of internals (bolts, control rod guides, control rod pins, baffles).

3.3. REQURIREMENTS IN GERMANY

In Germany, the appropriate standards for RVIs are at present the safety standards of the Nuclear Safety Standards Commission, specifically the KTA-3204. The latest issue is from June 1998. This standard shall be applied to the RVIs of light water reactors as well as to the tools and equipment, used for the installation and removal of the components.

During the design and manufacturing of RVIs for plants built before 1984 these rules were covered by specifications related to the project.

Components of the RVIs in these standards are categorized in three requirement levels, AS-RE 1 to AS-RE 3, depending on their individual tasks and functions.

One chapter deals with the design (construction) and one with the rules for the stress analysis for the RVI, which are primary derived from the ASME code. The rules for the RVI materials and material testing are covered in another chapter. The requirements for the materials are fixed in special material sheets in the annex. Further chapters contain requirements for the manufacturing and the operational surveillance and testing.

In all the chapters of these standards, the actual German standards and regulations, as e.g. DIN-EN, AD, SEW, VdTÜV guidelines, are to be applied.

3.4. REQUIREMENTS IN JAPAN

In Japan, the structural analysis for RVI is described in METI notification 501 [8] and JSME Code on Design and Construction for Nuclear Power Plants, JSME SNC1-2001 [10], which are based on subsection NG in the ASME Code. Reactive force caused by guillotine break of the primary piping system was not fully taken into account in the core barrel design of the older PWRs. To solve this issue, the LBB concept [11] was adopted to the primary piping system of these plants.

3.5. REQUIREMENTS IN THE RUSSIAN FEDERATION

RVI in the Russian Federation are designed and manufactured in line with basic nuclear standards [3, 12-16] (earlier designs were developed as per previous issues of these standards, except for WWER-440/230 plants, which were designed in line with industrial standards prior to the establishment of special nuclear ones).

Quality control and quality assurance procedures are applied during design, manufacture, assembly and installation of the reactor internals in accordance with applicable standards. The reactor internals are tested at manufacturer using vessel and core mock-ups for each unit as a part of RVI and RPV commissioning.

The design provides for ISI and partial repairs of reactor internals. The inspection is basically carried out every four years. The inspection is mainly visual and is focused on the various fixing and interconnection elements. After inspection of the guide tube block and its insertion into the reactor, the control rod movement is tested within the working range.

Reactor internals are designed to withstand various operating regimes. Following a maximum design basis accident at nominal power with superimposed SSE loads, the reactor internals design provides for emergency reactor shutdown, residual heat removal and core withdrawal. This has been verified and supported by experimental investigations.

In 2001 the procedure for strength analysis of WWER-440 RVI was developed [17]. This procedure is based on Ref. [4] and defines the scope and methods of RVI strength analysis to substantiate the possibility of its operation beyond 30 years taking into account material degradation caused by operationg condition.

4. OPERATING CONDITIONS

This section discusses the operating conditions for RVI. The operating conditions determine the presence of specific ageing mechanisms and the rate of potential degradation.

4.1. TEMPERATURE AND NEUTRON EXPOSURE

Western RVI

Information available concerning the service temperature and the maximum fluence received by the reactor internals of US, French and German PWR plants is shown in Table 8.

TABLE 8. OPERATING CONDITIONS FOR WESTERN PWR RVI (FOR 40YEARS OF OPERATION)

Plant type **Component**	**Temperature [°C].**	**Fluence**		
		10^{21} n/cm^2		**dpa**
		E> 0.1 MeV	E> 1 MeV	
US 900MW NPP				
Core Barrel	-	18	6,9	12
Core Baffle	-	160	74	110
Formers	-	18-160	7-74	12-110
Bolts	-	160	74	110
Upper Core Plate	-	0.43	0.22	0.3
Lower Core Plate		6.2	3.2	4.6
French 900 MW NPP				
Core Barrel	260 - 320	14	7	9,6
Core Baffle	290 - 370	109	54	80
Formers	290 - 370	13-76	6-38	10-56
Bolts	290 - 370	82	41	58
Upper Core Plate	≈ 320	0.5	0.25	0.3
Lower Core Plate	≈ 290	3-8	1.5-4	2-5.6
French 1300 MWe NPP				
Core Barrel	290 - 330	-	≈ 3	≈ 3.6
Core Baffle	≈ 350	-	≈ 11	≈ 13
German Konvoi 1300 Mwe NPP				
Core Baffle (barrel)	≈ 325	1.4	0.8	1.2
Core Envelope (shroud)	≈ 325	106	50	75
Bolts	≈ 325			
Upper Core Plate	≈ 325	0.48	0.3	0.45
Lower Core Plate	≈ 290	0.17	0.11	0.17

Table 8 gives only an order of magnitude of fluences as there are very strong gradients in the internal structures. As an example, in the formers, the fluence near the baffles is about 10 times higher than the fluence close to the core barrel. The differences in temperature and fluence between the various designs which appear in the Table VIII are not always significant, as the exact location in the different parts is not necessarily identical.

There are also relatively large uncertainties on the temperature reached due to the fact that γ heating is not always taken into account in the same manner.

The data for the 3 loops in the 900 MWe type reactors are based on 32 effective full power years of operation at 2,785 MWth.

The number of displacements per atom is estimated using an average cross section of 600 barns (for E >0.1 MeV) for the baffle plate, which leads to: 1 dpa ≈ 6.10^{-22} fluence (E >0.1 MeV, n/cm^2).

The neutron flux in fact varies with the azimuth angle in the core and along the baffles; for a 3 loop 900 MWe reactor, the maximum value of flux on the baffles is obtained for an azimuth around 13°. The ratio of the maximum flux over the flux at azimuth 0° is equal to about 2. For the core barrel, the maximum neutron flux is obtained in the azimuth range of 0° to 13° and the minimum neutron flux is close to the azimuth 45° with a factor of about 10 between the maximum and the minimum flux for E >0.1 MeV.

Operating conditions for the French 4-loop 1,300 MWe reactors are based on 32 effective full power years of operation at 3,817 MWth. In the 4-loop 1,300 MWe reactor the maximum fluence is reached at azimuth 45° for the barrel. The variation between the minimum flux and the maximum flux on the barrel is equal to a factor of about 5. The estimation of the temperature of operation of the different parts are also indicated and range between 286°C and 370°C.

The operating conditions for the Siemens/KWU Konvoi 1,300 MWe reactor and for 32 equivalent full power years lifetime are also summarized in Table 8. Normal service temperature varies between 290°C and 325°C, with a peak at 370°C due to gamma heating.

All components are subject to static and cyclic loading and must withstand additional loading arising from accident conditions.

WWER RVI

Information available concerning the service temperature and the maximum fluence received by the reactor internals of WWER reactors is given in Table 9.

TABLE 9. OPERATING CONDITIONS FOR WWER RVIS

Plant component	Temperature [°C]	Fluence	
		10^{21} n/cm^2 E >0.5MeV	dpa
WWER-440			
Core basket	269–298	20	
Core barrel outer surface	269–298	2	
WWER-1000			
Core shroud (thermal shield)	290–320	45	100
Core barrel	290–320	6	
Block of guide tubes	290–320	0.065	

The fluence values for WWER-440 RVI are estimated from the calculated maximum end of design life fluence for the inner surface of the RPV. The employment of dummy fuel (shielding) elements or low leakage core, which is used in most of these plants, lowers these values.

The WWER-1000 core baffle fluence value is based on 30 effective full power years.

4.2. PRIMARY COOLANT CHEMISTRY SPECIFICATIONS

Western RVI

The important parameters of the PWR primary coolant chemistry are the boric acid, lithium hydroxide, and hydrogen concentrations, and the resulting pH level. A minimum high temperature (~300°C) pH of 6.9 (pH_{300} = 6.9) is required to avoid heavy crud deposits on fuel rods, which can cause accelerated corrosion of fuel rod cladding and increased radiation fields. Some test results show that operation at pH300 of 7.4 results in less crud deposits than operation at 6.9. For current PWR operation, the typical range of pH300 is 6.9 to 7.4. The pH300 for most of the Electricité de France (EDF) plants is 6.9, and the pH is fixed at 7.2 if the cycle duration is 18 months.

Boron is added in the form of boric acid (H_3BO_3) as a neutron absorber for reactivity control. The boric acid concentration is changed throughout a reactor cycle to compensate for other changes in reactivity and is not varied independently. The boron levels are relatively high (1,000 to 2,000 ppm) at the beginning of the fuel cycle. Then, they are gradually reduced by 100 ppm/month. The concentration of lithium hydroxide (L_iOH) is co-ordinated with the boric acid concentration to achieve the desired pH of approximately 6.9 or higher at operating temperature. At the beginning of the fuel cycle, the typical lithium level is about 4 ppm for a boron level of 2,000 ppm, and is then reduced as the boron level decreases.

Hydrogen is added to the primary coolant to suppress the buildup of oxygen from radiolysis. A hydrogen concentration of 25–50 cm^3/kg has typically been used. Recent EPRI sponsored studies indicate that increasing the hydrogen concentration in the primary coolant raises the rate of primary water stress corrosion cracking (PWSCC). Consequently, EPRI is encouraging utilities to maintain hydrogen concentrations near the low end of the specified range (i.e. 25–35 cm^3/kg).

The Revision 2 of the EPRI guidelines for PWR primary coolant system water chemistry are listed in Table 10. EPRI is also about to issue a third revision of their guidelines, which will include the following changes: sulphate is added as a control parameter (50 ppb); the Action Level 1 for chlorides and fluorides will be 50 ppb, each; and the limitation on hydrogen control at 25–50 cc/kg will be removed for plants with steam generators susceptible to PWSCC. Also, there are some minor changes to the pH optimization principles, and the Level 1 definition. (The Level 1 value is now the value outside of which data or engineering judgement indicates that long term system reliability may be affected, thereby warranting an improvement of operating practices.)

Primary water chemistry in US PWRs is controlled in accordance with EPRI TR-105714. This document is periodically updated based on plant operating experiences.

TABLE 10. EPRI PRIMARY COOLANT CHEMISTRY GUIDELINES FOR POWER OPERATION (REACTOR CRITICAL)

Control parameter	Sample frequency	Typical value	Action level		
			1	2	3
Chloride, ppb	3/wk[(a)]	<50	–	>150	>1500
Fluoride, ppb	3/wk[(a)]	<50	–	>150	>1500
Lithium, ppm	3/wk[(b)]	Consistent with station lithium programme	–	–	–
Hydrogen, cc(STP)/kg H_2O	3/wk[(c)]	25–50[(d)]	<25 >50	≤15	≤5
Dissolved oxygen, ppb	3/wk[(a)]	<5	–	>100	>1000

(a) These frequencies are a minimum based on Standard Technical Specifications. Typical industry frequencies are daily.
(b) An increased frequency of sampling is recommended during operations that may significantly impact the lithium concentration (i.e., feed and bleed).
(c) An increased frequency of sampling is recommended during operations that may significantly impact the hydrogen concentration (i.e., feed and bleed, purging of pressurizer vapour, etc.)
(d) Maintain near the low end of this range.

WWER RVI

The WWER primary reactor coolant chemistry is a reducing, weak alkaline chemistry treated with the addition of ammonia, potassium, and boric acid. The allowable at power pH and dissolved hydrogen, oxygen, ammonia, chloride, fluoride, iron, oil, copper, and boric acid concentrations are listed in Table 11 for the WWER-440 plants with corrosion resistant stainless steel cladding on the inside surface of the reactor pressure vessel, the WWER-440 plants without reactor pressure vessel cladding, and the WWER-1000 plants.

TABLE 11. WWER PRIMARY COOLANT CHEMISTRY REQUIREMENTS FOR POWER OPERATION

Control parameter	WWER-440 with cladding	WWER-440 without cladding	WWER-1000
pH at 25°C	≥6.0	6.0–10.2	5.9–10.3
Ammonia, ppm	≥5.0	≥5.0	≥5.0
Hydrogen (at 0°C, 0.1 MPa), ppm	2.7–5.4	2.7–5.4	2.7–5.4
Dissolved oxygen, ppm	≤0.01	≤0.005	≤0.005
Chloride and fluoride, ppm	≤0.1	≤0.1	≤0.1
Corrosion products in terms of iron at steady state operation, ppm	≤0.2	≤0.2	–
Oil, ppm	≤0.05	–	–
Copper, ppm	–	≤0.02	≤0.02
Boric acid, depending on core reactivity margin, g/kg	0–8	0–9.0	0–10.0
Total iodine isotopes radioactivity at the time of sampling, Bq/1	$\leq 3.7 \times 10^8$	$\leq 3.7 \times 10^8$	–

– The lithium to boron ratio is controlled to maintain a pH of 6.9. If the lithium decreases to 2.2 ppm, it is held at 2.2 ppm until a pH of 7.4 is reached.
– Normally at 10.

5. AGEING MECHANISMS

This section describes the ageing mechanisms that can affect PWR RVIs and evaluates the potential significance of the effects of these mechanisms on the continued performance of safety functions of the RVIs throughout the plant service life.

Ageing mechanisms are specific processes that gradually change characteristics of a component with time and use. Ageing degradation are those cumulative changes that can impair the ability of a component to function within acceptance criteria. Service conditions outside prescribed limits, which are caused by design, fabrication, installation, operation, and maintenance errors, can accelerate the rate of degradation.

Evaluation of ageing mechanisms is based on PWR service experience, pertinent laboratory data, and relevant experience from other industries. In this report, the following ageing mechanisms were reviewed and assessed for relevance to RVIs:

– embrittlement;
– fatigue;
– corrosion;
– radiation induced creep, relaxation and swelling;
– mechanical wear.

The process for evaluation of an age related degradation mechanism relevant to the continued performance of an RVI important to safety leads to one of two possible conclusions:

– the ageing degradation is potentially significant to the extent that plant specific evaluation to manage ageing is required; or
– the ageing degradation does not significantly impair the ability of the RVI to perform its intended safety function.

In this revision, the significance of embrittlement and irradiation assisted stress corrosion cracking was re-evaluated.

5.1. EMBRITTLEMENT

There are two types of embrittlement which could affect PWR vessel internal components. These are irradiation embrittlement, which may affect core region internals, and thermal ageing embrittlement, which may affect the cast stainless steel parts and parts manufactured from martensitic stainless steel.

5.1.1. Description of irradiation embrittlement

Neutrons produce energetic primary recoil atoms which displace large numbers of atoms from their crystal lattice positions by a chain of atomic collisions. The number of neutrons bombarding a given location is traditionally measured by the fluence (n/cm^2 with E >1.0 MeV). A more recent neutron damage exposure measure is displacements per atom (dpa), which accounts for a wider neutron energy spectrum than the fluence. The fluence or dpa provide part of the information needed to assess irradiation embrittlement.

Wrought austenitic stainless steels do not exhibit the sharp ductile to brittle transition behaviour characteristic of low alloy and carbon steels. Rather, toughness losses due to irradiation tend to accumulate with increasing fluence and saturate at levels $>1\times10^{21}$ n/cm^2.

Until recently, there was little information available to quantify the effects of irradiation embrittlement on RVIs. New information describes the results of a fracture toughness study performed on irradiated Type 304 stainless steel reactor internal material taken from operating BWRs with fluences ranging from 1×10^{21} to 6×10^{21} n/cm^2, (E >1 MeV). This study confirmed a fracture toughness saturation level of 55 MPa√m for all fluences considered and can be directly applied to the evaluation of highly irradiated RVIs. Available information/data from the results of fracture toughness studies performed on irradiated Type 304, 316 CW and 347 stainless steel reactor internal material taken from operating PWRs with fluences as high as 2×10^{22} n/cm^2 (E >0.1 MeV) shows high fracture toughness values for all fluences considered and can be directly applied to the evaluation of highly irradiated RVIs.

Although resistance to crack propagation in internals materials decreases with increasing neutron fluence, integrity of stainless steel RVIs can be effectively assessed using fracture mechanics analyses. This will, however, require a detailed finite element analysis and a material database that is sufficient to provide the crack growth rates and the fracture toughness for the materials of interest.

5.1.2. Description of thermal ageing embrittlement

Thermal ageing embrittlement is a time and temperature dependent degradation mechanism. It is caused by the thermally activated movement of lattice atoms over a long time period, a process which can occur without external mechanical load. Changes in material properties (e.g. a decrease in ductility and toughness and an increase in strength properties and hardness) are the consequence of these diffusion processes. The significant parameters responsible for these ageing processes are:

- temperature;
- material state (microstructure);
- time.

Susceptible to this kind of mechanisms are cast stainless steels, to a lesser extent weld metal and some Cr rich martensitic steels. Several research projects funded by the USNRC, EPRI, George Fisher Limited of Switzerland, and a consortium of Westinghouse, Framatome and EDF have evaluated mechanical property degradation which results from thermal ageing embrittlement in typical cast duplex stainless steel materials [19].

Thermal ageing embrittlement of cast stainless can lead to precipitation of additional phases in the ferrite, e.g. formation of Cr-rich α–prime, phase by spinoidal decomposition; nucleation and growth of α -prime; precipitation of a Ni- and Si-rich phase, M_{23} C_6 carbides and growth of existing carbides at the ferrite/austenitic phase boundaries. Cast duplex stainless steel used in the piping of primary pressure boundary can be susceptible to thermal ageing embrittlement at operating temperatures, i.e. 290ºC–325ºC of NSSS. Thermal ageing embrittlement of cast duplex stainless steel at these temperatures can cause an increase in the hardness and tensile strength and a decrease in ductility, impact strength and fracture toughness of the material. The susceptibility of ASTM A-351 grades of cast stainless steel to thermal ageing embrittlement is a function of the ageing temperature, time at temperature and material composition including ferrite content.

All the cast duplex stainless steel reactor internals components are made using CF-8 or CF-8A. ASTM A-351 Grades CF-3, CF-3A, CF-8, CF-8A, and CF-8M are used in the piping of the primary pressure boundary. The Mo-bearing CF-8M is the most susceptible cast duplex stainless steel to thermal ageing embrittlement. The temperature at which CF-8M is susceptible to thermal ageing embrittlement decreases with Mo content. Non Mo bearing cast austenitic stainless steel, used in the reactor internals are much less susceptible to thermal ageing embrittlement. Based upon the available data on thermal ageing embrittlement of CF-8 materials, thermal ageing embrittlement is a non-significant degradation mechanism.

For reasons of stress relieving after welding, some parts of the internals are heat treated after welding in Siemens/KWU designed RVI. The heat treatment temperature is chosen to be 580°C +10°C/–20°C to avoid any impact on the material properties of the wrought austenitic stainless steels and the weld metal, but has been 700°C in the past, a temperature at which the formation of sigma-phase from the delta-ferrite and the respective influence on the material properties cannot be ignored.

5.1.3. Significance of embrittlement

Margins on crack tolerances

Embrittlement, either due to irradiation or thermal effects, does not directly cause cracking. However, the margin of a material to resist propagation of cracks due to other causes such as fabrication, fatigue or SCC is reduced. The significance of embrittlement to a given component depends on the probability of cracking, and the loading of the component. For cracked components, a fracture mechanics evaluation of material that has been exposed to high neutron fluence should be performed to assure crack stability on a component specific basis.

Thermal ageing embrittlement is not a significant degradation mechanism for RVIs made from wrought steel or Ni-Cr-Fe because these materials are not susceptible to the mechanism or the stress levels are not of sufficient magnitude to cause cracking irrespective of the delta ferrite content.

In the case of RVI component parts heat treated at 700°C for stress relieving as mentioned above, plant specific evaluations and irradiation surveillance programmes have shown that there is no significant ageing effect to be considered for the plant life.

Margins on design rules

Another consequence of the embrittlement is the loss of ductility of stainless steel material from more than 30% maximum elongation for un-irradiated material to 0.5% for some irradiated material. Consequently all the secondary stress classification has to be reviewed to check the new margin levels.

5.2. FATIGUE

5.2.1. Description

Fatigue is defined as the structural deterioration that occurs as a result of repeated stress/strain cycles caused by fluctuating loads and temperatures. After repeated cyclic loading of sufficient magnitude microstructural damage can accumulate, leading to macroscopic crack initiation at the most highly affected locations. Subsequent continued cyclic loading can lead to the growth of the initiated crack.

Fatigue behaviour is related to a variety of parameters, such as stress range, mean stress, cycling frequency, surface roughness and environmental conditions. Cracks initiate at stress concentrations such as geometric notches and surface defects. Fatigue crack initiation curves indicate how many stress cycles it takes to initiate fatigue cracks in components. These curves are materials related and indicate the allowable number of stress cycles for applied cyclic stress amplitudes. Design curves for RPV materials are given in ASME Section III [5], Appendix I or respective national standards such as KTA 3204 [7] or RCC-M [6].

Environment can significantly influence fatigue crack initiation. Environmentally assisted fatigue, often referred to as corrosion fatigue, must be considered when dealing with components in the PWR environment.

There are three sources of fatigue significant to the PWR. These are system cycling, thermal cycling and flow induced vibration.

System cycling

System cycling refers to changes in the reactor system, which cause variations in pressure and temperature. Examples of system cycling are startup, shutdown, scram and safety/relief valve blowdown. System cycling was the best understood source of fatigue during the time of vessel design.

Many vessel components were designed against system cycling fatigue crack initiation, using conservative amplitudes and frequencies of normal and upset loading cycles, together with the fatigue design curves of the ASME Code, Section III, Appendix I. Although the design process for internals considered system cycling fatigue in a less formalized manner, operating experience to date demonstrates that consideration of system cycling in the internals design process was adequate. The few documented cases of RVI fatigue failures have been attributed to other fatigue sources.

Thermal cycling

Fatigue thermal cycling may occur due to temperature fluctuations. Temperature transients during operation can cause local or global temperature gradients, resulting in thermal cycling at the interface of material and environment. Smooth and sharp temperature transients result in slow or rapid thermal cycling, both being a source for accumulation of fatigue usage. Causes for smooth transients are generally start up and shutdown procedures or load following operation modes. Connections and disconnection of systems, ECCS water injection, and leaking of hot or cold water through untight valves may result in rapid thermal cycling (e.g. thermal shock). The effect of both is ageing of material in terms of low cycle fatigue (slow cycling), or high cycle fatigue (rapid cycling).

Flow induced vibration

Flow induced vibration is caused when coolant flowing past a component sheds vortices which create cyclic loads. These loads generally occur in a frequency range up to about 20 Hz, leading to the expectation that flow induced vibration cycles accumulate early in operation, probably during pre-operation tests. However, it is possible that some modes of flow induced vibration are associated with a particular operating mode, which occurs infrequently.

5.2.2. *Significance of fatigue*

Fatigue life estimates include both crack initiation and crack propagation. Crack initiation is estimated by determining the fatigue usage at a specific location that results from either actual or design basis cyclic loads. Time to initiation can be predicted only if the sequence of the applied loads and recurrence frequency is well known. Such estimates are uncertain if the cyclic loading is random.

ASME Code Section III fatigue analyses are performed to satisfy design requirements and are not normally the best estimate of actual fatigue usage. The conservatisms applied to the laboratory fatigue data base and design-basis transients are substantial. The effects of environment and high cycle thermal and mechanical loads may not have been explicitly considered, so the service duty may be higher than reported.

Fatigue damage is significant for some RVI components (bolts and pins) based on cracking incidents reported in service and because of the component design bases. Some internals were designed against fatigue crack initiation using conservative amplitudes and recurrence frequencies for normal and upset loading cycles, together with the fatigue design curves of the ASME Code, Section III, Appendix I. Typically, fatigue usage factors are less than 0.10. The number of loading cycles considered during design and the conservatism of the cyclic amplitudes in combination with low fatigue usage factors should be sufficient to justify continued operation.

In the case of most internals components the stress and cycling ranges are such that cyclic analysis is not required per ASME Code Section III NB-3200. This is confirmed by design calculations, startup test measurements and service experience. Due to an absence of significant cyclic stress, fatigue is not significant for other RVI components important to safety.

5.3. CORROSION

Corrosion is the reaction of a substance with its environment that causes a detectable change which can lead to deterioration in the function of the component or structure. In the present context, the material is steel and the reaction is usually an electrochemical reaction.

The appearance of corrosion is governed by the so-called corrosion system consisting of the metal and the corrosive medium (the environment) with all the participating elements that can influence the electrochemical behaviour and the corrosion parameters. The variety of possible chemical and physical variables leads to a large number of types of corrosion:

– corrosion without mechanical loading (general corrosion and local corrosion attack, selective corrosion attack as e.g. intergranular corrosion);
– corrosion with mechanical loading (stress corrosion cracking, corrosion fatigue) and synergistic effects of neutron irradiation (irradiation assisted stress corrosion cracking);
– flow induced corrosion attack (e.g. erosion corrosion).

During the electrochemical processes, the metal ions dissolve in liquid electrolyte (anodic dissolution) and hydrogen is produced. This is the process of material loss and creation of corrosion products. When mechanical stresses or strains are also present, the anodic dissolution of the metal can be stimulated, protection layers (oxide layers) can rupture or hydrogen interaction with the metal (absorption) can be promoted which can produce secondary damage. The combined action of a corrosive environment and mechanical loading can cause cracking even when no material degradation would occur under either the chemical or the mechanical conditions alone.

Water chemistry control during operation, as well as during shutdown, is very important with respect to avoiding corrosion problems. Thus the content of all additives has to be carefully monitored and the ingress of impurities has to be strictly avoided, e.g. during standstill periods and maintenance work. The water chemistry regimes which are used in the primary coolant circuits of the various types of reactors and which have proven effective are presented in Tables 10 and 11 of this report.

5.3.1. General corrosion

5.3.1.1. Description

General corrosion is typically characterized by an oxidizing reaction, which occurs uniformly over a material surface. This reaction causes a thinning of the surface, and corrosion proceeds until the surface fails by localized penetration or insufficient crosssectional area to support a load. However, PWR internals are made from austenitic steel with very low corrosion rates in the PWR environment.

5.3.1.2. Significance of general corrosion

Evaluations have established that general corrosion of austenitic RVIs is not a significant ageing mechanism. These conclusions are based on the very low general corrosion rates which have been experienced in PWR operating plants for all RVI materials. For steel 08Ch18N10T which is used for WWER 440 and WWER 1000 RVIs, the general corrosion rate has been investigated during 30 years of operation in primary coolant water with temperature up to 300°C and the results showed that the corrosion rate does not exceed 0.1 mm.

5.3.2. Stress corrosion cracking

Stress corrosion cracking (SCC) is the term given to crack initiation and subcritical crack growth of susceptible alloys under the influence of tensile stress and a “corrosive” environment. SCC is a complex phenomenon driven by the synergistic interaction of mechanical, electrochemical and metallurgical factors. There are two forms of SCC, which can proceed through a material in either of two modes: intergranular (along the grain boundaries) or transgranular (through the grains). Sometimes the modes are mixed or the mode switches from one mode to the other. Intergranular stress corrosion cracking (IGSCC) and transgranular stress corrosion cracking (TGSCC) often occur in the same alloy, depending on the environment, the microstructure, or the stress/strain state. Stainless steel castings and welds containing high levels of delta ferrite are unlikely to experience SCC. SCC usually proceeds perpendicular to the tensile stress. Cracks also vary in degree of branching or formation of satellite cracks. Most ferritic and duplex stainless steels (stainless steel castings and welds containing high levels of delta ferrite) are either immune or highly resistant to SCC.

Perhaps the most critical factor concerning SCC is that three conditions necessary for producing SCC must be simultaneously present. The elimination of any one of these three factors or the reduction of one of these three factors below some threshold level eliminates SCC. The three necessary conditions for SCC are:

– corrosive environment.

– tensile stress;

– susceptible material.

5.3.2.1. Intergranular stress corrosion cracking

IGSCC usually appears like brittle material behaviour, since the crack propagates with little or no attendant macroscopic plastic deformation. Many alloys are susceptible to IGSCC in at least one environment. However, IGSCC does not occur in all environments, nor does an environment that induces IGSCC in one alloy necessarily induce IGSCC in another alloy.

Influence of environment

Two major parameters influence IGSCC. These are water conductivity and electrochemical potential (ECP). The greatest benefits with respect to preventing IGSCC are attained when both water conductivity and ECP are controlled. Crevices significantly increase the probability for SCC due to the highly aggressive local environment that may form within the crevice.

The PWR coolant is a high purity electrolyte. Although pure water is a low conducting molecular liquid, it conducts electricity due to the presence of hydrogen (H+) and hydroxide (OH–) ions that result from ionization of a small fraction of the water molecules. It must be noted that even the theoretically lowest conductivity level may not prevent IGSCC in the normal water chemistry environment since the presence of other ionic species in the solution increase conductivity and thereby the probability for corrosion.

Electrochemical potential is a thermodynamic measure of the tendency of a material to undergo a corrosion reaction under certain fixed conditions. Radiolysis of the water passing through the core results in large concentration of H_2, O_2, H_2O_2 and several other free radicals.

In the PWR coolant, the level of oxidizing species (i.e. oxygen and hydrogen peroxide) primarily controls the ECP of all structural materials. For PWR the ECP of the stainless steel is lower than –230 mV (SHE). When the ECP of stainless steel is lower than –230 mV (SHE), IGSCC of thermally sensitized stainless steel is not possible.

Influence of stress

There are three primary sources of tensile stresses for the RVI: these are fabrication induced stresses, primary stresses, and secondary stresses. Fabrication stresses consist of stresses introduced during manufacture and installation (i.e., fit up and assembly in the shop or field plus those introduced by machining or forming operations and welding). As for weld residual stresses, abusive grinding can produce surface residual stresses near or above the yield point of the material. Service related primary stresses, including mechanical and pressure load stresses, may also be as high as the yield stress at constraints in an assembly. Corrosion product stress may also play an important role in crack propagation. Secondary stresses such as thermal transient stresses can also be very high.

The material may exhibit a threshold stress intensity factor, K_{ISCC}, below which IGSCC does not occur. Threshold level is related to environment and material and may vary if these conditions change.

Influence of material

IGSCC susceptibility varies from alloy composition and with metallurgical condition. Given conditions of normal stress and BWR environment, several materials have shown susceptibility to IGSCC as a result of the material itself or due to its fabrication history. For the RVI, these materials include Type 304/304L austenitic stainless steel and nickel-based alloys 600, X-750, and 182 weld metal.

When non-stabilized austenitic stainless steels containing more than 0.030 wt.% carbon are furnace or weld heated in a temperature range of approximately 450 to 850°C for a sufficient period of time, a precipitation reaction occurs due to the insolubility of carbon in the alloy. Austenite containing more than 0.030 wt.% carbon precipitates complex metal carbides (primarily $Cr_{23}C_6$) at the grain boundaries. This chromium carbide precipitation at the grain boundary creates an envelope of chromium depleted austenite that in certain environments is not resistant to corrosion. The chromium depleted zone is no longer a stainless steel, but rather a localized low alloy steel anode galvanically coupled to a large area stainless steel cathode. If a sufficient tensile stress is placed on an austenitic stainless steel that has become thermally "sensitized" by this phase transformation, then IGSCC can occur if the environment can support the corrosion reaction.

Recent BWR service experience has shown that stabilized stainless steel may undergo similar cracking if low stabilization ratios in combination with inadequate heat treatment occur.

Cold work operations such as bending, cutting, forming, rolling, and especially grinding can cause austenitic stainless steel to become susceptible to SCC in respective environment.

The nature of the cold work will affect the degree of SCC susceptibility, since the cold worked region of the material may only be at the surface. However, the combination of cold work followed by sensitization is synergistically damaging. Even low carbon content materials can become highly susceptible to IGSCC. When SCC initiates in cold-worked material, subsequent crack propagation in the annealed material beyond the cold-worked region is slow.

Solution heat treatment after fabrication also eliminates SCC concerns for components with cold-worked regions to preventing IGSCC are attained when both water conductivity and ECP are controlled.

Alloys 600 and X-750 have suffered IGSCC in the PWR (e.g. steam generator tubes, RPV head penetrations, split pins, etc.). In components other than the RVI, Alloy 182 has also experienced IGSCC in applications where weld residual stresses and fairly high applied stresses were present. It must conservatively be assumed that Alloy 182 exposed to normal coolant conditions is susceptible to IGSCC.

5.3.2.2. Transgranular stress corrosion cracking

Austenitic stainless steels are particularly susceptible to SCC in chloride environment; temperature and the presence of oxygen tend to aggravate chloride SCC of stainless steels.

Stress corrosion cracking occurring in these environments is typically transgranular. Sensitivity to IGSCC is not necessary for TGSCC to occur. There is evidence that the combined effects of chlorides and oxygen promote TGSCC in solution annealed austenitic stainless steels. Very high levels of chlorides do not result in stress corrosion cracking in the absence of oxygen in the water. Furthermore as the oxygen content of the environment is increased, the required concentration of chloride ions to produce TGSCC becomes smaller.

5.3.2.3. Significance of IGSCC and TGSCC

As primary water conditions are such that ECP values of –230 mV (SHE) are controlled, which is generally the threshold value for the occurrence of IGSCC in austenitic stainless steel material, there is no risk for this kind of cracking in 18/10 CrNi type of steel even in the case of sensitized material condition.

Ni based alloys are susceptible for this cracking mechanism in high temperature water environment.

TGSCC can occur in austenitic stainless steel if impurities such as chloride concentrate in crevice locations under operational conditions. Attention should be paid for this reason to maintenance and replacement work where lubricants are in use, to avoid any contamination. The latter had been reported to be the root cause of cracking in some replaced bolting. The use of chloride free and qualified lubricants and consumables is mandatory.

5.3.2.4. Irradiation assisted stress corrosion cracking

Irradiation assisted stress corrosion cracking (IASCC) is also characterized by intergranular crack initiation and propagation. Many of the factors discussed for IGSCC also apply to IASCC (e.g. pH, conductivity, crevices, etc.). However, there are subtle differences between the two phenomena. Austenitic stainless steels that undergo IASCC need not be thermally sensitized. Also IASCC is highly dependent on neutron fluence exposure level. Annealed and irradiated austenitic stainless steel becomes susceptible to IASCC when certain criteria (i.e., threshold fluence levels as a function of stress level) are met or exceeded. Both stabilized and non-stabilized stainless steels appear to be equally susceptible to IASCC.

Based on available field and laboratory data, a neutron fluence (E>1 MeV) “threshold” of $\sim 5 \times 10^{20}$ n/cm^2 (approx. 0.8 dpa) appears to exist for annealed Type 304, 304L, 347, and 348 stainless steels in highly stressed components, and $\sim 2 \times 10^{21}$ n/cm^2 (approx. 3.1 dpa) for lower stress components.

In France, the IASCC threshold is estimated over 40 dpa for 304 stainless steel and over 3 dpa for low stressed cold work 316L (like baffles). For high stressed 316L (like bolts), the value is not yet finalized.

5.3.2.5. Significance of IASCC

Although IASCC of PWR internals has not been observed for structural component parts globally so far, it may be a concern that increases with time. Some cracking that occurred in the baffle bolts has been attributed to IASCC. IASCC has been observed in PWR core internal components such as the control rod cladding. This indicates that IASCC could become a significant ageing mechanism for RVI.

5.3.3. Erosion corrosion

5.3.3.1. Description

The effect of solution velocity or the movement of a metal in a solution, on the rate and form of corrosion is extremely complex. From a fundamental viewpoint, an increase in fluid velocity can increase the corrosion rate.

The movement of solutions above a certain threshold velocity level can result in another form of attack that is the result of the interaction of fluid induced mechanical wear or abrasion plus corrosion. The general term "erosion corrosion" (E/C) includes all forms of accelerated attack in which protective surface films and/or the metal surface itself are removed by this combination of solution velocity and corrosion such as impingement attack, cavitation damage and fretting corrosion.

The terms "flow assisted corrosion" and "flow accelerated corrosion" (FAC) have been used to describe the erosion (or thinning) of carbon steel in nuclear and fossil power plants where there is no threshold solution velocity. FAC is a complex phenomenon that is a function of many parameters of water chemistry, material composition and hydrodynamics.

FAC involves the electrochemical aspects of general corrosion plus the effects of mass transfer and momentum transfer. FAC and E/C are characterized by the constant removal of protective oxide films, ranging from thin invisible passive films to thick visible films of corrosion products, from the metal surface.

5.3.3.2. Significance of erosion corrosion

Stainless steel and nickel based alloys are generally resistant to erosion corrosion. Because the vessel internals are made of stainless steel, erosion corrosion resistance of PWR RVIs has been excellent under design basis operating conditions. There has been no evidence of erosion corrosion in the PWR RVIs. This successful experience support the conclusion that erosion corrosion is not a significant degradation mechanism for PWR RVIs.

5.4. RADIATION INDUCED CREEP, RELAXATION AND SWELLING

5.4.1. Description

Neutron irradiation creates a large number of interstitials and vacancies that can annihilate on sinks such as dislocations, grain boundaries, surfaces, etc. by diffusion controlled processes. The kinetics of annihilation is different for interstitials and vacancies and depends on stress, temperature, material microstructure, etc.

If interstitials are eliminated rapidly, the excess vacancies coalesce into voids or bubbles inside the metal leading to swelling of the structure.

If a significant stress is applied, interstitials can migrate towards locations perpendicular to the applied stress creating an irradiation creep or irradiation relaxation phenomena. Although thermally induced creep is almost insignificant in the temperature range considered for PWR RVI operation, there are examples of an increase in the diameter of pressurised test pieces of annealed 304L irradiated at 390 °C. This strain can be attributed to the irradiation-induced creep and swelling. Irradiation induced creep is linear in dose *and in strain with an incubation period*, at least for temperatures and doses for which the swelling and thermally induced creep are negligible. The effect of the irradiation parameters, temperature, neutron flux, and helium production rate on irradiation-induced creep is relatively unknown for the materials used in PWR RVI design, particularly in the relevant operating temperature range.

Swelling is characterised by an incubation dose and a stationary rate of swelling: The temperatures which are relevant to the RVIs correspond to a temperature range where swelling is slight and is usually located within the incubation phase for the doses most often reached.

5.4.2. Significance of irradiation creep and swelling

The operation temperature of RVIs is generally low enough to limit the effect of swelling. However, locally, if temperature increases, e.g. due to gamma heating in thick parts, swelling can occur and create local straining. The major possible swelling concern is related to the geometrical changes that could occur. This could have potential negative impact on, e.g., control rod movement or coolant flow.

Void swelling is inhibited by the formation of helium bubbles in thermal reactors. The maximum amount of void swelling reported to date is less than 0.25 %. In addition, there is no credible evidence that void swelling can cause cracking in austenitic stainless steels. Void swelling has not been observed in operating PWR RVIs in the USA, France or Japan.

Creep is a function of stress level, temperature and time at temperature. Fast neutron exposure enhances austenitic stainless steels to creep. Some creep/relaxation of baffle bolts has been observed during testing and replacement of baffle bolts in the USA, France, Japan, and Belgium.

5.5. MECHANICAL WEAR

5.5.1. Description

This degradation type is broadly characterized as mechanically induced or aided degradation mechanism. Degradation from small amplitude, oscillatory motion, between continuously rubbing surfaces, is generally termed fretting. Vibration of relatively large amplitude, resulting in intermittent sliding contact between two parts, is termed sliding wear, or wear. Wear generally results from concurrent effects of vibration and corrosion.

The major stressor in fretting and wear is flow induced vibration. Initiation, stability, and growth characteristics of damage by these mechanisms may be functions of a large number of variables, including the local geometry, the stiffness of the component, the gap size between the parts, flow velocities and directions, and oxide layer characteristics. Wear is defined as the removal of material surface layers due to relative motion between two surfaces or under the influence of hard

5.5.2. Significance of wear

Mechanical wear has been identified as degradation mechanism at specific locations in the RVI due to flow induced vibrations. Today, as a result of monitoring systems of vibrations, loose parts and specific ISI programme, this degradation mechanism is of minor importance concerning RVI capability to perform its safety function.

5.6. HANDLING

Internal structures have to be removed periodically either partially or completely for refuelling or for in-service inspection. The handling of these internals has to be made with great care since some parts of the internals (pins, etc.) can be easily distorted for instance by impact on other parts. Several incidents related to this type of event have been reported.

6. INSPECTION AND MONITORING

6.1. INSPECTION AND MONITORING METHODS

Inspection

RVI components are inspected in accordance with Section XI of the ASME Code [5], or according to corresponding national standards [10] as applied in other countries, such as KTA 3204 in Germany [7] or the respective Russian standards. While monitoring is not a requirement in all countries, most if not all plants utilize monitoring techniques.

Non-destructive examination is required by the regulatory agencies and code and standards of each of the Member States. The objective of the visual examination is to discover relevant conditions including distortion, cracking, loose or missing parts, wear or/and corrosion. Underwater TV is a reliable examination tool coupled with photographic capabilities, enlargement, immediate printouts, and a permanent record. Further enhancement is available with an underwater conveyance system.

Supplemented ultrasonic examination is useful for the evaluation of components such as guide tube support pins and baffle/former assembly bolts where detection of indications is an essential part of reactor internals ageing management. Ultrasonic examination techniques such as the cylindrically guided wave technique can be used to detect flaws in bolts and threaded rods using transducers which emit ultrasonic sound waves that travel through solids and liquids at different velocities. Ultrasonic examination of reactor internal components can be an accurate and reliable technique for detecting flaws in RVI components. Ultrasonic examination can be utilized to measure stress relaxation in reactor internals bolting. Ultrasonic examination techniques must be customized for specific RVI components geometrical configuration, i.e. the presence of locking devices to the fastener heads, and/or accessibility restrictions.

If any defect or degradation mechanism is observed by inspection or monitoring, it should be assessed according to applicable national codes and standards. These are discussed in Section 6.2 below.

Monitoring

While monitoring techniques/systems cannot detect RVI materials degradation. They are a useful tool to provide information on RVIs behaviour during plant operation. The following monitoring techniques are recommended for use during plant operation:

– loose parts monitoring;
– neutron noise monitoring;
– direct vibration monitoring;
– on-line primary water chemistry monitoring.

If the loose parts, neutron noise or vibration monitoring systems indicate that there is a loose part in the reactor vessel or that the fuel or reactor internals are vibrating, the information/data should be diagnosed. In the case of a loose part, the size or weight and the location in the primary coolant system can be determined and a decision as to plant shut down could be made based on safety and/or economic consideration. In the case of neutron noise or direct vibration monitoring, if there is an indication that either the fuel or a component of the reactor internals is vibrating, the information/data should be diagnosed in accordance with the applicable code, such as the ASME Section on Operation and Maintenance. Based upon the diagnosis of the information/data from the vibration monitoring, a decision can be made to shut the plant down or continue operating until the next outage.

If the on-line chemistry monitoring system detects that the primary coolant is out of specifications, the source of the ingress of the impurities should be identified and corrective actions taken to meet the chemical specifications. If halogens are detected out of specifications, a clean or flushing operation will be required during the next outage.

6.2. NATIONAL PRACTICES

6.2.1. France

The basic inspection technique for the RVI at all plants is the visual examination, which is carried out every 10 years. For the baffle bolts, visual inspection is applied to the 3 rows of bolts at the bottom and to the upper row. Complementary inspections using ultrasonic testing are applied periodically (every 5 to 10 cycles) on some specific 900 MW plants where baffle bolt cracking was detected. (in particular the CP0 oldest plants)

Monitoring of the RVI is achieved by the techniques mentioned above, i.e. by using neutron ex-core monitors and accelerometer sensors. Neutronic measurements allow monitoring the amplitude and the spectrum of the vibrations of internals. Accelerometer sensors are used for monitoring loose parts in the primary circuit.

A complementary ISI programme is developed on some French plants in other locations as anticipative or preventive maintenance on, for example, control rod guide pins or thimbles.

6.2.2. Germany

In Germany after the replacement of the X750 baffle former bolts with austenitic stainless steel 1.4571, intensive ultrasonic testing was performed by the utilities. After an extensive period without any indications, again the standard requirements as per KTA 3204 [7] were applied, which means visual examination for the RPV Internals. The scope of inspection is defined specifically for each NPP. According to KTA 3204 it can be distributed to 4 or 5 refuelling outages, resulting in a scope of 20 to 25% of the baffle former bolts per refuelling outage.

6.2.3. Japan

(a) General Inspection Requirement

The basic inspection requirements are given in JEAC-4205 [20], the Japan Electric Association Code for ISI of light water cooled nuclear power plant components and also in the JSME Code on Fitness-for-Service for Nuclear Power Plants, JSME S NA1-2002 [21]. The basic examination required by the above codes is a periodical visual examination of the reactor internal structures (Section 2, Class 1 Components, Examination Category B-N-3 (in JSME Code Category GP-1). The objective of the visual examination is to discover relevant conditions including distortion, cracking, loose or missing parts, wear or/and corrosion. The examination is performed once every 10 years, using an examination tool coupled with an underwater TV camera. In the JSME code, the examination method using an underwater TV camera to inspect weld lines on core internals is called MVT-1, modified VT. The result of this is recorded on videotape.

(b) Voluntary inspections and guideline for inspection and evaluation of vessel internals

Other examinations which are performed as voluntary inspections by a utility, such as ultrasonic examinations of guide tube support pins, baffle/former bolts and visual inspection for locking nuts of guide tube support pins, etc., are useful for evaluation of components soundness.

Japanese PWR utilities, a manufacturer and universities produced Guidelines on inspection and evaluation of Core Internals for the following internals, which were published by the Thermal and Nuclear Power Engineering Society in March 2002 [22–25]:

- Baffle-former bolts
- Barrel former bolts
- Core barrel
- Control rod cluster guide tube

These guidelines will be incorporated into the JSME Code on Fitness-for-Service for Nuclear Power Plants. The detail of the guidelines for baffle-former bolts and the control rod cluster guide tube are shown in Section 7.5.4 and 7.7 respectively.

(c) Monitoring

A number of monitoring systems are used at nuclear power plants. These monitoring systems include loose parts monitoring systems, nuclear instrumentation systems, leak detection systems and a primary water chemistry monitoring system.

6.2.4. Russian Federation

The scope and methods for RVI ISI are outlined in appropriate operating instructions developed by the general designer of a reactor unit. For example, WWER-1000/320 RVI are inspected as per instruction “The reactor. Description and operating instruction, 320.00.00.000 TO, EDO Gidropress, 1989”. The inspection methods for base metal, welds, fixing, etc. consist of visual examination and dye penetrant testing. The inspections are usually carried out during the first refuelling and subsequently every four years.

The requirements for RVI visual examination are given in the procedure developed in 1997 and updated in 2001, “Procedure of the automated television inspection of WWER RVI”. MTK-RVI-003-01, St.-Peterburg, 2001.

Since 2000 the scope of RVI ISI of WWER-440/179 (units 3 and 4 of Novovoronezh NPP) WWER-440/230 (units 1 and 2 of Kola NPP) was expanded (additional visual examination and instrumentation measurement) according to the special programmes developed by the general designer of a reactor unit. During examination, special attention is given to zones of RVI subjected to high stresses, welds under high neutron irradiation, places of core basket where baffle to former bolts and RVI fixing and interconnecting elements are located.

Monitoring of WWER-440 RVI is achieved by the techniques as mentioned in sec. 6.1 by using neutron ex-core and in-core monitors, accelerometer and pressure pulsation sensors. The WWER-1000 NPP units are equipped with the abovementioned systems only partially.

Provision of the WWER-1000 NPP units with the missing monitoring systems is being carried out according to the corresponding programme developed by the utility.

6.2.5. United States of America

ASME Section XI [5], in-service inspection requires visual inspection of reactor vessel internal components. The required visual inspections of reactor internal components are carried out within a given 10 year period. Currently, enhanced visual inspection of reactor vessel internal components is under development and is expected to be an ASME XI requirement. In addition, it is expected that in the near future, volumetric inspection of critical reactor vessel components such as baffle bolts will be required.

7. SERVICE EXPERIENCE AND RELATED MAINTENANCE

Up to now, the majority of events reported about problems in internals of PWR are related to stress corrosion or fatigue failures, including or not neutron irradiation effects. There are no events listed for WWER reactors since none have been reported to date. In addition to the detailed information given below, a summary of failures observed to date in relation to dominant ageing mechanisms is given in Table 12.

7.1. FAILURE OF BOLTS AND SUPPORTING BEAMS OF THE THERMAL SHIELDS

In the earlier PWRs, failure of bolts occurred in the core barrel (e.g. in France, Italy, USA in 1968). The failure of the bolts, made of cold-worked 316 stainless steel, was attributed to fatigue associated with flow induced vibrations of the thermal shield. The thermal shield was removed and the bolts replaced by new ones made of nickel alloy Inconel X-750.

Later on, in 1984 in France, at more than 100 000 hours of operation, several core barrel bolts fabricated of Inconel X-750 failed again. In that case, the intergranular cracking was attributed to IGSCC in water. The bolts were replaced by new ones, using an improved thermal treatment for the alloy. The neutron irradiation was not considered to have played a significant role.

In the 1980s, similar problems were reported in USA or in Europe and were attributed again to flow induced vibration.

TABLE 12. FAILURES OBSERVED IN RELATION TO AGEING MECHANISMS

	Embrittlement	Fatigue	Corrosion	Wear	Irradiation creep and Swelling	Handling Incident
Upper internals						
Control rods guide tubes				X		
Guide		X				
Guide tube support split pins			X			
Thermocouple columns						X
Upper core plate fuel alignment pins			X			X
Lower internals						
Baffles (bolts, plates)	X		X			
Thermal shield fasteners or welds		X				
Locating systems (Radials key, etc.)			X			
Lower instrumentation guide columns						X
Bottom instrumentation thimble		X				

7.2. CRACKING OF THE GUIDE TUBE SUPPORT SPLIT PINS

The guide tubes support pins are fixed to the control rod guides and are inserted in the upper core plate. They are made of alloy Inconel X-750. There is a thread at one end of the pin and the other end has a long slit parallel to the axis which is inserted in the upper core plate. The first failure of these pins was recorded in Japan in 1978, then in France in 1982, in the USA in 1983 and in the Republic of Korea in 1997.

The failure was attributed to IGSCC in water, which is known to occur in high nickel materials. The neutron irradiation at this location is very low and was not considered as being a significant factor in this case. Soon after the failure occurred, tools to control the degradation and replace the pins became available.

Improved replacement pins were developed with a modified material (solution annealing at a higher temperature ~1,100°C), a new fabrication route, a new design reducing the in service stresses and new installation specifications. Shot peening was also used at the final stage of the fabrication of the pins. In the Republic of Korea, the support pins were replaced with strain hardened Type 316 stainless steel ones.

The life of the pins has been significantly increased by these mitigation methods but the problem is not yet considered to be completely solved.

7.3. CRACKING OF FUEL ASSEMBLY ALIGNMENT PINS MADE FROM INCONEL X-750

A design feature common to all Siemens/KWU PWRs is that each fuel assembly is fixed in position relative to the core internals by means of two lower (in the lower core support) and two upper alignment pins (in the grid plate). Since 1980, there have been isolated reports of broken upper alignment pins in some plants and this damage was attributed to IGSCC of the material Inconel X-750, which had been chosen for its wear resistant properties. Although the number and location of the broken pins did not necessitate immediate repairs, a number of defective pins were replaced with pins made from austenitic stainless steel type 316 Ti (German designation 1.4571), protected against excessive mechanical wear by a chromium carbide hard-facing coating. All the upper alignment pins in the Konvoi plants are equipped with pins made of the new material with some modifications in the design of the pins.

7.4. BAFFLE JETTING

At the beginning of the 1980s, jets of water through the baffles created damage to the fuel elements. This effect was related to the pressure between the baffles and the core vessel with the “down flow” of the water in this location. Some modifications in the formers, assuring the “up flow” of the water, have reduced the pressure and suppressed the baffle jetting. This “up flow” measure, was implemented practically everywhere later on.

7.5. CRACKING OF THE BAFFLE BOLTS

7.5.1. General

In 1980’s inspections indicated baffle bolts cracking. The bolts are made of 316 cold worked stainless steels. They failed by intergranular cracking. Normally, 316 steel is not prone to IGSCC in this water environment and all the bolts cracked were located in the second and third rows from the bottom, that is exactly the place corresponding to the highest neutron irradiation. This demonstrates that the neutron irradiation is a significant feature for this cracking, even if the exact mechanism is unknown now. It is difficult to conclude whether the bolts cracked by IASCC, or due to irradiation embrittlement, or by other IGSCC phenomena. To date, baffle bolt cracking was observed only in the “down-flow” design.

This cracking is a concern and made necessary the development of ultrasonic methods for the non destructive examination of the bolts. Table 13 shows Baffle-Former Bolts inspection results in some IAEA Member States.

TABLE 13. BAFFLE-FORMER BOLTS INSPECTION RESULTS IN SOME IAEA MEMBER STATES

Country	Material	Plants Inspected	Operation Hours	Inspection Method	Number of Crack Bolts/ Indications per plant
France	CW316SS	CP0: 6/6	around 150 000	Visual + UT	2 to 100
		CPY: 12/28	around 12 000	Visual + UT	1 to 4
United States of America	347/ CW316SS	4*	655 000	UT	56**

* Southern Nuclear Farley Units 1 and 2 — no indication of cracked bolts after accumulation of 278,000 EFPH of operation (CW316SS)
** Confirmed cracked bolts are those with observed cracks after removal from service. Maximum number are the sum of the confirmed and those with UT indications and not removed from service
The possibility of changing all the lower internals is also studied in Europe and in Japan [26, 27].

7.5.2. Belgium

In Belgium, the situation is similar at Tihange 1. Twenty-one bolts were detected as cracked in 1991 and 90 bolts replaced in 1995 (as reported at the 1995 meeting of the International Working Group on Life Management of Nuclear Power Plants).

7.5.3. France

In the 1980s baffle jet failure occurred in the older French reactors CP0 (6 3-loop plants in Fessenheim and Bugey). In 1988, the first UT inspection was done on baffle bolts and led to crack detection. Between 1989 and 1993, all the CP0 plants were converted to up-flow instead of down-flow, and an inspection plan was defined. Some baffle bolts were extracted for metallurgical investigation and to confirm the degradation mechanism IASCC (in Figure 5b). Consequently three units changed 1/3 of their baffle bolts between 2000 and 2003. Major inspection results are presented in Figure 5a. Similar degradations have been encountered on bolts with 10 years and 20 years of operation.

The other 3-loop plants, CPY, are less sensitive due to up-flow from beginning of operation, better cooling of the bolts (Figure 5c), lower torque, and the parabolic shape of the bolt under the head. Similar improvements can be noted for other younger plants (all the 4-loop French plants).

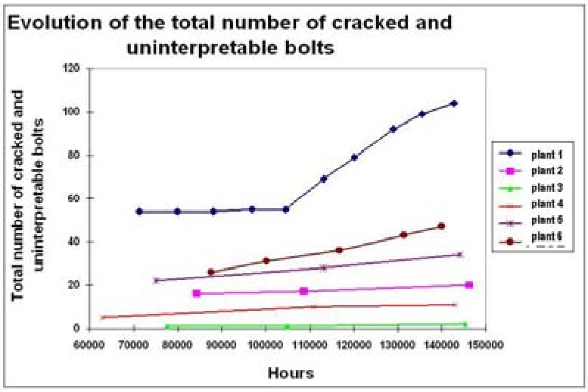


Fig. 5a. Results of EDF baffle bolt inspection on CP0 plants (from reference [28]).

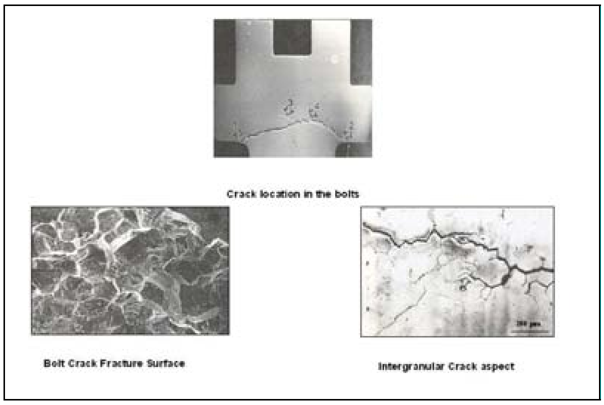


Fig. 5b. Investigation on removed bolts (from reference [28]).

Other aspects were checked during bolt replacement and on extracted bolts. The investigations confirm the irradiation induced relaxation for some bolts and the hardness gradient along the bolts connected to irradiation level (Figure 5d).

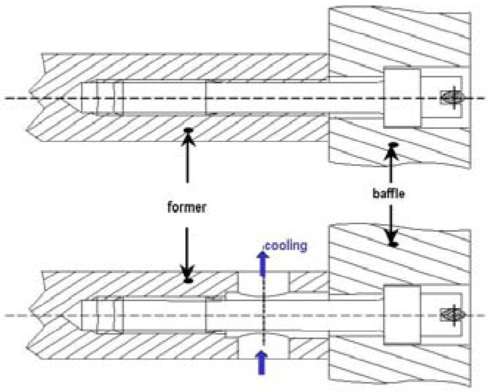


Fig. 5c. Removal bolt torque versus irradiation level [28].

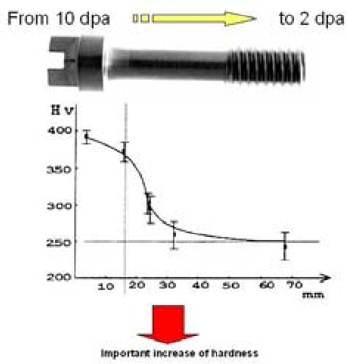


Fig. 5d. Hardness gradient along removal bolt [28].

Some complementary investigation on bolts has been done on CHOOZ A shutdown plant after 140,000 hours of operation. four bolts were used for metallurgical analysis with a cumulative dose between 0 and 22 dpa [29]. The bolt of the core barrel, considered as unirradiated, shows a constant value of hardness along the bolt, equal to the unirradiated material value. The axial profile of hardness carried out on an irradiated bolt shows that there is a gradient of hardness between the most irradiated location (400 Hv) and the least irradiated location (270 Hv). The hardness of this bolt starts to decrease at 2.5 dpa and the maximum of hardening is reached for the most irradiated location (3.6 dpa). The hardness of the most irradiated bolt (between 10 and 22 dpa) indicates that hardness is homogeneous all along the bolt at 400 Hv. It confirms the existence for these conditions of a threshold dose beyond which hardness does not vary anymore; this threshold is estimated between 3.6 and 10 dpa.

7.5.4. Germany

In Germany, damage to the baffle former structure was first discovered in 1978 at a PWR plant. During the inspection, an enlarged gap of approximately 2 mm was found between two baffles. An ultrasonic examination revealed that a number of the Inconel X-750 baffle former bolts were defective (some bolts gave no back echo, others gave back echoes which did not correspond to the bolt length. When the bolts were removed, some of them were found to be broken, others had cracks of different sizes. Studies of the cracks showed that the cause of damage was not fatigue, but rather IGSCC. The crack locations in the damaged baffle-former bolts were in the transition between the bolt head and the necked-down shaft (extending into the hexagonal socket), within the necked-down shaft and in the transition between the shaft and the thread and in the thread itself, Fig.6. All of the baffle former bolts were examined by ultrasonic testing. Damage to Inconel X-750 baffle former bolts was subsequently found in other German plants as well. No defects were found in bolts made from austenitic stainless steel. Today it is well known that the nickel based alloy Inconel X-750 is susceptible to IGSCC in highly pure high temperature reactor coolant.

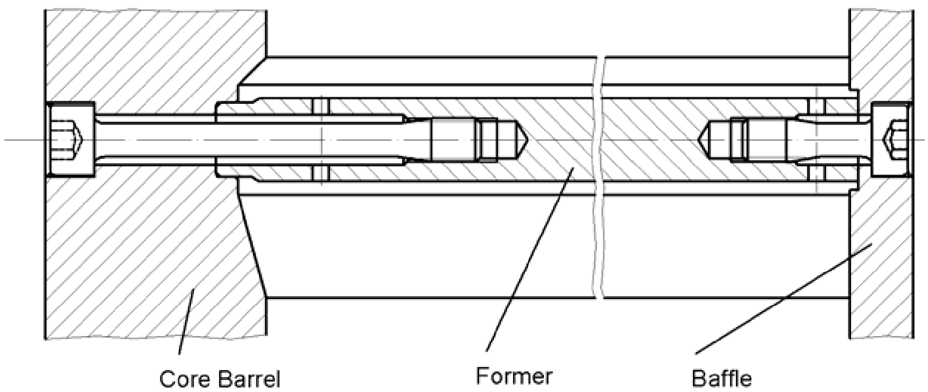


Fig.6. Siemens/KWU baffle-former bolts.

Following this event, in Siemens/KWU plants, the baffle former bolts made of Inconel X-750 have been replaced (a total of over 5,100) with bolts made of material 1.4571 (type 316 Ti, cold worked material). In new plants, erected after the detection of this kind of cracking, bolts made from the alternative material were already installed during the original assembly. Following this material change no defects due to this degradation mechanism have been found until very recently. However, in 2005 some indications in the austenitic baffle former bolts were detected at one German PWR plant.

7.5.5. Japan

Inspection programme for baffle-former bolts

Japanese PWRs have not experienced any IASCCs on baffle-former bolts. However, Japanese PWR utilities and manufacturer produced guidelines on inspection and evaluation of Core Internals for baffle-former bolts taking into account service experience of foreign PWRs. The guidelines were published by the Thermal and Nuclear Power Engineering Society in March 2002 [22]. In addition Japanese PWR utilities have been replacing all baffle-former bolts at some of their 2-loop plants.

The PWRs are divided into four groups, i.e. Groups 1 to 4. Baffle-former bolts of the Group 1 plants have the highest susceptibility to IASCC. Inspection initiates at these plants. For the Group 1 plants, the guideline requires ultrasonic examination of baffle-former bolts which cover more than 1/4 of the core. The initial inspection time and inspection interval are determined on the basis of an acceptable number of damaged bolts and the bolt damage prediction curve. According to the guideline, utilities produce and revise the bolt damage prediction curves for their plants using damage prediction equations which consider fluence, temperature and stress; inspection results of other Japanese PWRs as well as foreign PWRs are also taken into account. The damage prediction equations are as follows:

Time to IASCC initiation

$$t_1 = \left(1\times10^{21}\times F^{-1}\right)\exp\frac{Q}{R}\left(\frac{1}{T}-\frac{1}{613.15}\right)$$

where
F: neutron fluence per unit time per area (n/<u>hour</u>. cm^2)
t_1: time to IASCC initiation when the fluence is F
T: temperature (K)
Q: activation energy (J/mol)
R: gas constant (J/mol. K)

Time from IASCC initiation to damage initiation

$$t_N = k\left(\frac{\sigma}{\sigma_y}\right)^{-n}\times\left[\exp\frac{Q}{R}\left(\frac{1}{T}\right)\right]\times\Phi^{-m}$$

where
$t_{N:}$ time from IASCC initiation to damage initiation
σ: bolt stress (N/mm^2)
σ_y: yield stress of the bolt material (N/mm^2)
Q: activation energy of the bolt material (J/mol)
R: gas constant (J/mol. K)
T: temperature (K)
Φ: neutron fluence (n/cm^2)
k: proportional constant (<u>hour</u>/(n/cm^2)$^{-m}$)
n= 1/0.11
m=0.9
Q=16.7 kJ/mol (4 kcal/mol)
k= 1.4×10^{19}

Inspection programmes of Group 2-3 plants will be produced based on the inspection results of Group 1 NPPs.

The guidelines will be incorporated into the JSME Code on Fitness-for-Service for Nuclear Power Plants. [21]

Replacement of RVIs

Replacement of only Upper Internals was performed at Prairie Island in 1986. The replacement of all the reactor internals at Ikata 1 in 2005 is the first one in the world. Shikoku Electric Power Company, which is an owner of Ikata 1, decided that the replacement of all the RVIs would provide an effective and proactive method of preventive maintenance.

The basic strategies of reactor vessel internals replacement (RVIR) are

- Basic Policy of Replacement work
- To minimize radiation exposure and schedule
- To set new reactor internals in the water-filled reactor vessel with high accuracy as same as the requirement during initial construction
- To provide the latest standard design of reactor internals to improve reactor equipment reliability

New reactor internals were designed and manufactured based on the latest design of the Japanese two-loop standard reactor internals (see Fig. 7). The design features are: an inverted 'top hat' type of upper support plate; panel type neutron pad with reduced width; and modification — including an increase in number — of guide tubes.

Modifications to the baffle-former bolts include:

- L-type baffle plate;
- Longer baffle and barrel-former bolts;
- Flow holed on the former plate of the bolt location for cooling.

Improvements to the baffle structure consisted of stress reduction, better material and environment in order to guard against SCC. Fig. 8. compares the old and the new baffle structures. The temperature of the bolts can be reduced by setting cooling holes in the baffle-former plate of the bolt location, and increasing bypass flow in the baffle region to 1.5% (best estimated flow).

The L-type baffle plate applied to convex corner in the baffle section was adopted in order to decrease thermal bending stress (secondary stress) on the bolt neck due to relative thermal expansion between the baffle plate and the baffle-former plate. Consequently the L-type baffle plate can have a lower number of baffle-former bolts and an elongated bolt shank. [30]

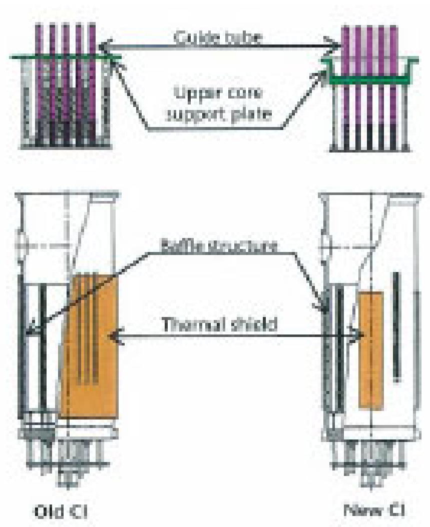


Fig. 7. Comparison between the old and new RVIs

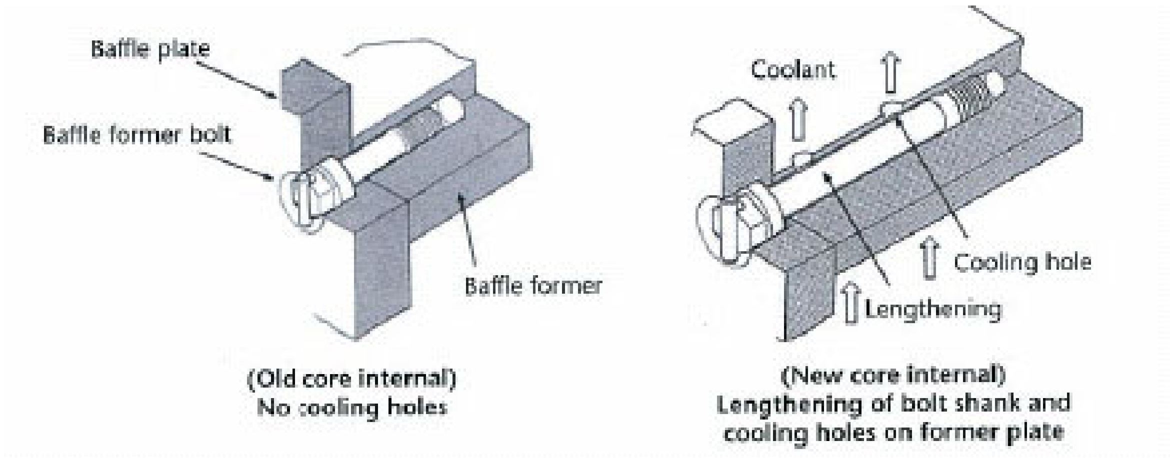


Fig. 8. Illustration of the old and new baffle structures.

7.5.6. United States of America

PWR owners groups plans for dealing with cracking of baffle former bolts are discussed in Information Notice 98-11 [31] and are summarized below.

The Westinghouse Owners Group (WOG) provided an assessment of the cracking of the baffle former bolts identified in foreign PWRs, including the potential impact of cracking on domestic Westinghouse plants, and provided information on its current and planned activities. The WOG stated that because of the large number of baffle former bolts in the baffle assembly, the failure of a few bolts should not have a significant safety impact.

The WOG activities include:

(1) development of analytical methods and acceptance criteria for bolt analysis,
(2) performance of risk-informed evaluations,

(3) performance of analysis for three plant groupings (2-loop, 3-loop, and 4-loop) of what constitutes acceptable bolting,
(4) continued participation in domestic and foreign related activities. The WOG identified lead plant candidates for the 2-loop and 3-loop groups and a proposed inspection schedule for each group.

In the USA the Nuclear Energy Institute (NEI) formed a Materials Technical Advisory Group (MTAG) made up of representatives from the utilities. The MTAG requested that EPRI support efforts to prepare guidelines/recommendations for the ISI of RVI components that are critical to the continued and safe operation of nuclear power plants. In preparing the inspection guidelines, degradation of reactor internal components due to irradiation, fatigue, wear, and corrosion were taken into consideration. Currently, Westinghouse and AREVA are preparing a report as part of an EPRI MRP to identify critical reactor internal components using screening based on the various degradation mechanisms. The guidelines are to be discussed with the US NRC prior to the end of 2006 because under the USA License Renewal schedule two plants are scheduled for ISI before the end of 2009.

7.6. WEAR OF BOTTOM INSTRUMENTATION THIMBLES

Bottom instrumentation thimbles are present in reactors in the USA and France. They are relatively long tubes having a small wall thickness (1.15 to 1.31 mm). The water flow around them can generate vibrations and wear by local contact with their guides, leading then to leakage. To date, thimble tube wear has been observed in at least 20 plants. Replacement by a thicker thimble is sometimes performed when this occurs.

7.7. WEAR OF CONTROL ROD GUIDES

In RVI components, wear between control rods and guide tubes results from the transverse motions caused by flow induced vibration, and/or from the axial sliding which occurs during insertions and withdrawals. This can be improved either by the use of a wear resistant deposit or by design modification, or both.

In Japan, Japanese PWR utilities, a manufacturer and universities have decided to produce guideline on inspection and evaluation of the control guide tubes, even though the safety evaluations in the plant life management programmes concluded that the predicted wear rate of the control guide tubes was very small and it would not become a safety significant issue. The background of the decision is as follows:

- Some plants have experienced wear of control rods and thus the possibility of wear of the control guide tubes cannot be completely denied;
- If wear of the control guide tubes takes place, it might impede the insertion of control rods.

The guideline was published by the Thermal and Nuclear Power Engineering Society in March 2002 [24].

The guidelines require implementation of an examination method of control rod guide holes of the tube which can measure amount of wear. The initial inspection time and inspection interval are determined based on a prediction of an acceptable amount of wear and wear progress. The guidelines will be incorporated into the JSME Code on Fitness-for-Service for Nuclear Power Plants. [21]

8. AGEING MANAGEMENT PROGRAMME

The information presented in this report indicates that the operating environment, primary coolant water and radiation can cause ageing degradation of RVIs. In particular, ageing mechanisms of radiation embrittlement, stress corrosion cracking (SCC), and the combination of radiation and stress corrosion cracking, known as irradiation assisted stress corrosion cracking (IASCC) can lead to both safety and economic concern. While the age related mechanisms of thermal ageing, corrosion, wear and fatigue are not considered to be safety significant by themselves, they can increase the safety significance of irradiation embrittlement and SCC. However, these age- related mechanisms could be of economic concern on their own.

The previous sections of this report dealt with the key elements of a PWR RVIs ageing management which should aim to maintain the fitness for service of RVIs at nuclear power plants. This section describes how these elements are integrated within a plant specific RVIs ageing management programme utilizing a systematic ageing management process which is an adaptation of Deming's Plan-Do-Check-Act cycle to ageing management (Fig.7). Such an ageing management programme should be implemented in accordance with guidance prepared by an interdisciplinary RVI ageing management team organized at a corporate or owners group level. For guidance on the organizational aspects of a plant ageing management programme and interdisciplinary ageing management teams refer to IAEA Safety Report Series No. 15 "Implementation and Review of Nuclear Power Plant Ageing Management Programme." [32]. In addition, the IAEA is currently preparing a new Ssafety Guide on ageing management which will provide key recommendations on ageing management of SSCs important to safety.

A comprehensive understanding of the RVIs, their ageing degradation and the effect of the degradation on the ability of the RVIs to perform their design functions are fundamental basis for an ageing management programme. This understanding is derived from the knowledge of the design basis (including the applicable codes and regulatory requirements), the operating and maintenance history (including surveillance results), the pre-service and in-service inspection results, and generic operating experience and research results.

In order to maintain the integrity or fitness for service of RVIs, it is necessary to control within defined limits the aged-related degradation of the RVIs. Effective ageing degradation control is achieved through the systematic ageing management process consisting of the following ageing management tasks, based on understanding of RVI ageing:

- operation within operating guidelines aimed at minimizing the rate of degradation (managing ageing mechanisms)
- inspection and monitoring consistent with requirements aimed at timely detection and characterization of any degradation and validating the ageing prediction;
- assessment of the expected or observed degradation in accordance with appropriate guidelines to determine integrity and fitness for service;
- maintenance (i.e. repair or parts replacement) to correct unacceptable degradation (managing ageing effects).

RVIs ageing management programmes coordinate programmes and activities contributing to the above ageing management tasks in order to detect and mitigate ageing degradation before RVIs safety margins are compromised. These programmes reflect the level of understanding of the RVIs ageing, the available technology, the regulatory licensing requirements, and the plant life management consideration/objectives. Timely feedback of experience is essential in order to provide ongoing improvements in the understanding of the RVIs ageing degradation and in the effectiveness of the ageing management programme. The main features of RVIs ageing management programmes, including the role and interface of relevant programmes and activities in the ageing management process, are shown in Fig. 9 and discussed in Section 8.1 below. Application guidance is provided in Section 8.2. National and international R&D programmes in support of the RVIs ageing management are outlined in Section 8.3.

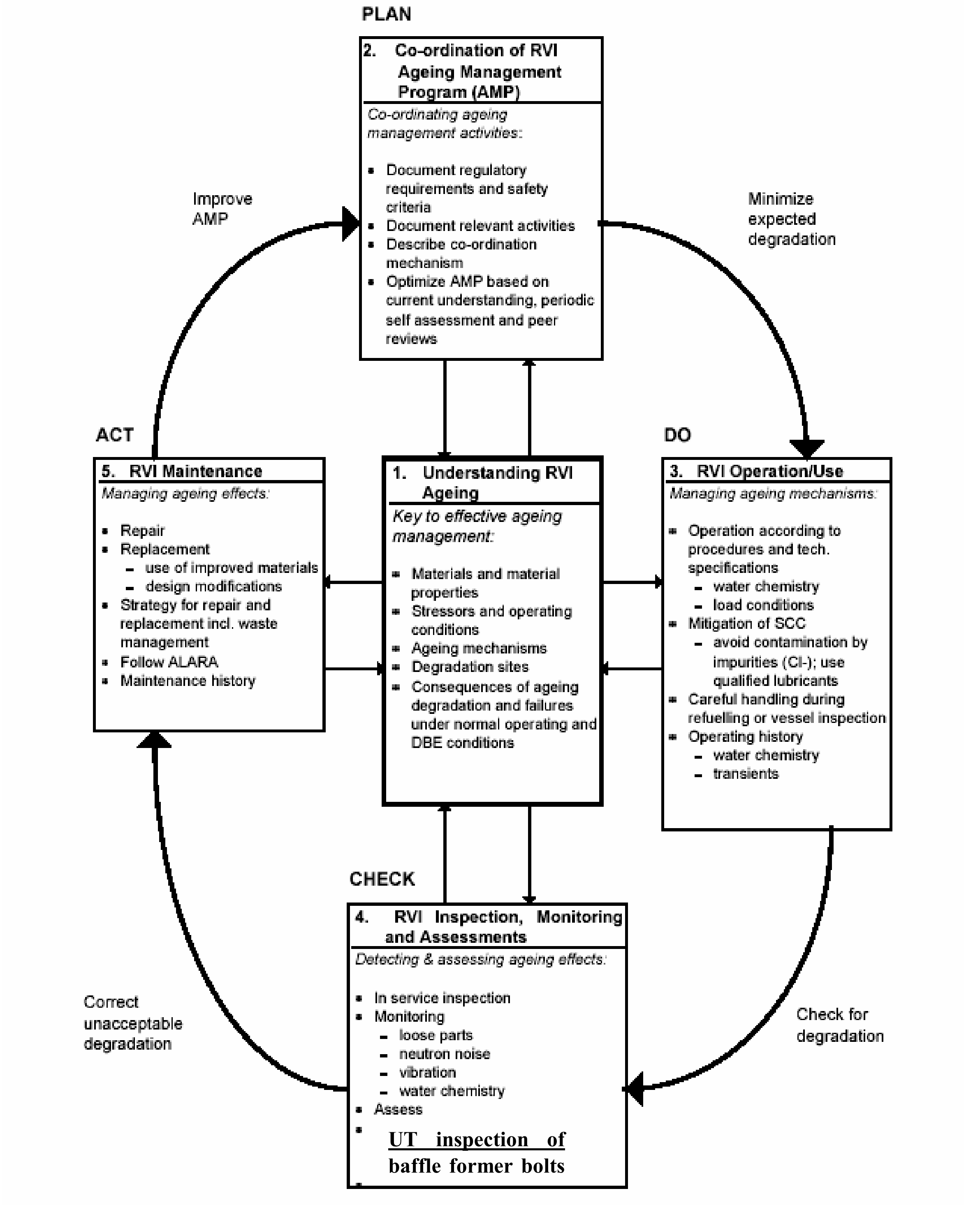


Fig. 9. Ageing management programme for RVIs.

8.1. KEY ELEMENTS OF THE AGEING MANAGEMENT PROGRAMME

8.1.1 Understanding ageing

Understanding RVI ageing is the key to effective management of RVI ageing. It is the key to: integrating ageing management activities within systematic ageing management programmes, managing ageing mechanisms through prudent operating procedures and practices (in accordance with technical specifications); detecting and assessing ageing effects through effective inspection, monitoring, and assessment methods and managing effects using proven maintenance methods. This understanding consists of: a knowledge of RVI materials and material properties, stresses and operating conditions, likely degradation sites and ageing mechanisms, condition indicators and data needed for assessment and management of RVI ageing and effects of ageing on safety margins.

The understanding of RVI ageing is derived from the RVI baseline data, the operating and maintenance histories, and external experiences. This understanding should be updated on an ongoing basis to provide a sound basis for the improvement of the ageing management programme consistent with operating, inspection, monitoring, assessment and maintenance methods and practices.

The RVIs baseline data consist of the performance requirements, the design basis (including codes, standards, regulatory requirements), the original design, the manufacturer's data (including material data), and the commissioning data (including pre-service inspection data). The RVI operating history includes the pressure/temperature records, number of transients, system chemistry records, fluence/dpa log, and all ISI results. The RVI maintenance history includes design modifications, replacement parts/components, inspection records and assessment and timing of maintenance performed. Retrievable up to date records of this information are needed for making comparison with applicable codes, standards, regulatory rules, and other external experience.

External experience consists of the operating and maintenance experience of (a) RVIs of similar design, materials of construction, and fabrication; (b) RVIs with similar operating histories, even if the RVI designs are different; (c) relevant research results. It should be noted that effective comparisons or correlation with external experience requires a detailed knowledge of the RVI design and operation. The present report is a source of such information. However, this information has to be kept up to date using feedback mechanisms provided, for example by owner groups. External experience can also be used when considering the most appropriate inspection method, maintenance procedure, and technology.

8.1.2 Coordination of the ageing management programme

Existing programme relating to the management of RVI ageing include operations, surveillance and maintenance programmes, operating feedback, research and development and technical support programmes. Experience shows that ageing management effectiveness can be improved by coordinating relevant programmes and activities within systematic ageing management programmes can improve ageing management effectiveness utilizing the systematic ageing management process. Safety authorities increasingly require licensees to define such ageing management programmes for selected systems, structures, and components important to safety. The co-ordination of an RVI ageing management programmes includes documentation of applicable regulatory requirements and safety criteria, of relevant programmes an activities and their respective roles in the ageing management process, and mechanisms used for programme co-ordination and continuous improvement. Continuous ageing management programme improvement or optimisation is based on current understanding of the RVI ageing and on results of periodic self assessment and peer reviews.

8.1.3 Operation/use of reactor vessel internals

NPP operation has a significant influence on the rate of degradation of plant systems, structures and components. Exposure of RVIs to operating conditions (e.g. temperature, pressure, water chemistry) outside prescribed operational limits could lead to accelerated ageing and premature degradation. Neutron and gamma radiation also has an effect on the rate of RVIs degradation. Since operating practices influence RVI operating conditions, NPP operations staff has an important role to play in minimising age related degradation of the RVIs by maintaining operating conditions within prescribed operational limits to avoid accelerated ageing. Examples of such operating practices are:

- performing maintenance according to procedures designed to avoid contamination of RVI components with boric acid or other reagents containing halogens;

- on-line monitoring and record keeping of operational data necessary for predicting ageing degradation and defining appropriate ageing management actions;

- fuel loading scheme to influence the rate of radiation embrittlement, IASCC, and swelling.

Effective ageing management of the RVIs and a possible plant life extension requires prudent operation and maintenance of plant systems that influence RVIs operational conditions (not only the primary system but also the auxiliary systems like water purification and injection systems), and record keeping of operational data.

8.1.4 Inspection, monitoring and assessment

Inspection and monitoring

RVI inspection and monitoring activities are designed to detect and characterize significant component degradation before the RVI safety margins are compromised. Results of RVI inspections together with the understanding of the RVI ageing degradation, provide the basis for managing detected ageing effects through maintenance and/or changes in operating conditions.

Inspection and monitoring of RVI degradation falls in two categories:

- in-service inspection;
- monitoring of temperature and pressures, water chemistry, transients (relative to fatigue).

The ISI programme should adequately address the most critical sections identified through the evaluation of ageing mechanisms.

Monitoring temperature provides input for the assessment of radiation embrittlement, swelling, and IASCC. Transient monitoring provides realistic values of thermal stresses fluctuations as opposed to design basis thermal stress fluctuations for fatigue assessments.

Assessment

The main safety function of the RVIs is to protect the core (fuel), maintain stability to receive the control rods when inserted and to provide a passage for coolant flow. Safety margins are part of the design and licensing requirements of a NPP to ensure the integrity of the RVIs under both normal and accident conditions. An integrity assessment is used to assess the capability of all the components (baffle plates, former and baffle bolts, core barrel, etc.) to perform the required safety function within the specified margin of safety, during the entire operating interval until the next scheduled inspection. The safety assessment must also include the potential of a double-ended-pipe break and the resulting blow down forces on the RVIs.

8.1.5 Maintenance, repair and replacement

Maintenance actions that can be used to manage ageing effects detected by inspection and monitoring methods in different components of the RVIs are described in Section 7. Decisions on the type and timing of the maintenance actions are based on an assessment of the observed ageing effects, available decision criteria, and an understanding of the applicable ageing mechanism(s), and the effectiveness of available maintenance technologies.

Maintenance actions for management of radiation embrittlement are rather limited. Unlike the reactor pressure vessel fuel management programmes (low leakage core) and reactor vessel annealing, irradiation embrittlement of the reactor vessel internals can only be minimized by reducing the reactor power. Maintenance actions for managing IASCC, irradiation assisted cracking (IAC), or IGSCC can be modification of the coolant chemistry or replacement of RVI components with materials that are less susceptible to IGSCC. RVIs components that have been determined by inspection to exhibit cracking, such as baffle plates, baffle bolts, guide tube support pins, etc. can be replaced.

Currently, a number of methods to mitigate SCC in PWR RVIs are being studied. The methods currently under evaluation are the addition of zinc to the primary reactor coolant and the coating of RVIs components with one of the noble metals. The addition of zinc to the primary coolant loop was intended to reduce the activity of the RVIs as part of the ALARA programme. However, EPRI studies indicated that zinc additions to the primary reactor coolant could also retard SCC in nickel based alloys such as alloy 600. There are also indications that zinc additions to the primary reactor coolant could prevent or retard SCC in austenitic stainless steel alloys.

It has recently been reported that introducing nobles metals as an alloy addition for austenitic stainless steel or as a coating to already fabricated RVI components could retard or prevent SCC.

Coating of the RVIs components subject to wear with nickel or chromium can minimize the amount of wear. Due to the ingress of Co into the medium, Co-bearing materials such as Stellite should be avoided as hard-facing materials.

8.2 APPLICATION GUIDANCE

The RPV ageing management programmes should address both the safety and economic aspects of RVI ageing to ensure both the integrity and serviceability of RVIs during the design life and any extended service life of the RVIs. The following sections provide guidance on dealing with the relevant ageing mechanisms.

8.2.1 Embrittlement

As already stated, there are two types of embrittlement which could affect RVI components. These are irradiation embrittlement, which may affect core region internals, and thermal ageing embrittlement, which may affect the cast stainless steel parts and parts manufactured from martensitic stainless steel.

Irradiation embrittlement

Irradiation embrittlement of RVIs fall under the category of a long term safety related ageing management. All RVIs materials are to some degree sensitive to radiation embrittlement. The ageing management programme activities which address radiation embrittlement can provide material data for fracture mechanics analysis. This could be achieved by:

- fluence mapping;
- utilization of radiation embrittlement databases/trend curves to predict the degree of radiation embrittlement for a given RVI component;
- laboratory testing of RVIs that were either replaced during service or that failed during service.

Thermal ageing embrittlement

Thermal ageing embrittlement can be addressed by:

- utilization of databases pertaining to the degradation of cast and martensitic stainless steel due to ageing to predict the degree of the embrittlement for a given RVI component;
- periodic ISI of all cast stainless steel RVI components should be carried out to ensure timely detection of flaws; if any indications/flaws are found during the ISI, the component should be replaced.

8.2.2 Fatigue

A fatigue assessment is made in the design phase in order to prevent any crack initiation. This assessment is made by using the cyclic stresses and number of cycles given in the RVIs design report. These values are determined using the estimates of the type and number of transients provided by the NSSS vendors.

In the ageing management programmes the following should be considered:

- Transient monitoring can be used to obtain more accurate estimates of both the total number of cycles and the stress ranges. For RVIs that went into operation prior to installation of a transient monitoring system, a review of past operating records must be made to determine the number and type of transients prior to the installation of the monitors. Transient monitoring systems are a very valuable tool for predicting the service life of the RVIs and should be part of the ageing management programmes.
- If a flaw is detected during ISI, fracture mechanics analysis including the fatigue crack growth prediction must be performed. Using a correlation between cyclic crack growth rate, da/dN versus ΔK. The growth of the flaw can be determined using the methodology given for instance in Appendix A to ASME Section XI [5] or in any equivalent national code.

- The database available that incorporates the effect of radiation on crack growth rate da/dN versus ΔK and on fracture toughness K should be utilised to determine if the cracked component should be replaced.
- If a flaw is detected during ISI, consideration should be given to removing the flaw by taking a boat sample or removable of the component containing the flaw. A microstructural analysis should be performed to determine if striations are evident on the surface of the flaw. Striations on the surface of a flaw are an indication that the initiation of the flaw or the growth of the flaw was due to fatigue.

8.2.3 Stress corrosion cracking

The following activities of the ageing management programmes address stress corrosion cracking:

- fluence mapping;
- utilization of databases that contain data on the effect of irradiation on the susceptibility of reactor internal materials to SCC (including modes of cracking, IGSCC or TGSCC, materials chemistry, and most importantly fluence/dpa level);
- periodic ISIs performed on the basis of the data given in such databases;
- fatigue initiation and crack growth analysis utilizing fracture mechanics methodology performed if indications/flaws are reported by the ISI;
- deterministic or probabilistic structural analysis performed to determine the maximum number of cracked bolts which can be tolerated.

Cracking of baffle former bolts

Cracking of baffle former bolts is not considered an immediate potential safety concern. The following ageing management actions are aimed at timely detection of cracks in the baffle former bolts:

- Subdivide NPPs according to susceptibility of their baffle former bolts to IASCC using bolt damage prediction equations/curves that take into account fluence, temperature, stress as well as operating experience. However, it is recognized that quantifying the stresses at each of the baffle bolts is very difficult. Perhaps it is more productive to develop damage prediction equations/curves based upon fluence and temperature.
- Develop baffle former bolts inspection programme for the lead NPPs. In the USA and Japan this task has been completed.
- Perform UT examination of baffle former bolts of the lead plants. In the USA this task was completed with the inspection of baffle bolts at Farley, Point Beach, and Ginna.
- Develop baffle former bolts inspection programme for NPPs with lower IASCC susceptibility on the basis of inspection results from the lead NPPs. This task will require development of the damage prediction equation/curves.

In addition, the replacement of all the RPVIs would provide an effective and proactive method of preventive maintenance.

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CONTRIBUTORS TO DRAFTING AND REVIEW

Brumovsky, M.	REZ, Czech Republic
Elliot, B.	US NRC, United States of America
Faidy, C.	EDF, France
Inagaki, T.	International Atomic Energy Agency
Kang K.S	International Atomic Energy Agency
Kastner, B.	Framatome ANP, Germany
Mager, T.R.	Westinghouse Electric Corporation, United States of America
Makihara, Y.	International Atomic Energy Agency
Pachner, J.	Pachner Associates, Canada
Plyushch, A.	Gidropress, Russian Federation
Piminov, V. A.	Gidropress, Russian Federation

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