Materials Reliability Program: VVER Issue Management Tables (MRP-471)

2021 TECHNICAL REPORT

Materials Reliability Program: VVER Issue Management Tables (MRP-471)

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EPRI Project Manager
M. Burke

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ABSTRACT

A comprehensive, integrated understanding of materials issues and aging management options is a fundamental consideration in ensuring continued safe operation and in the development of overall plant business and operating strategies. Nuclear utilities worldwide continue to face issues related to degradation of pressurized water reactor (PWR) pressure vessels, reactor internals, steam generators and piping components. This VVER, ("Vodo-vodyanoi Energetichesky Reaktor" translating to Water-Water Power Reactor) Issue Management Table (IMT) report identifies key gaps in VVER industry's state of knowledge regarding materials degradation phenomena and associated aging management capabilities for VVER primary system components. The IMTs support the industry needs to develop and maintain research work plans that evaluate strategic materials management issues and ensures that materials degradation is proactively addressed through funding of appropriate R&D.

The set of R&D gaps identified in this report were developed as an analogous set of Issue Gaps to those developed by EPRI for western style PWRs and BWRs as a means of meeting the intent of the NEI 03-08 Materials Initiative in the United States. Those gaps are maintained and updated periodically to address changes to the state of industry knowledge regarding PWR primary system materials degradation as well as utility asset management needs. It is the intent of the VVER IMT to provide similar support to the VVER fleet. The gap assessment results provide a key basis for ensuring that MRP and SGMP program R&D strategies and priorities continue to meet the needs of VVER plant operators.

Keywords

VVER
Materials Research Program
Steam Generator Management Program
Issue management tables
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PRIMARY AUDIENCE: VVER plant operators and materials support personnel for VVER plant operations.

SECONDARY AUDIENCE: Utility advisors involved in VVER support activities.

KEY RESEARCH QUESTION

What are the strategic gaps in state of knowledge with regard to proactively managing VVER materials degradation, and how should these gaps be prioritized?

RESEARCH OVERVIEW

The VVER Issue Management Table (IMT) report is an assessment of the knowledge of aging plants' materials degradation that is intended to identify and prioritize key knowledge gaps that should be addressed to better support proactive management of VVERs. In this assessment, a knowledge "gap" is defined as a deficiency or insufficiency in an area of technical knowledge that has been identified as important to achieving a reasonable standard of confidence that primary system component degradation can be managed such that components will remain serviceable and capable of performing their intended functions for the remainder of plant life. These gaps are known as "R&D" gaps since they form the basis for developing Research and Development (R&D) projects that intend to provide the knowledge needed to close these gaps.

Gaps from the most recent version of the Pressurized Water Reactor (PWR) IMTs were reviewed by a VVER expert panel to determine their relevance to VVERs. This process was performed at a workshop in Prague Czech Republic in 2019. This workshop identified both relevant and non-relevant gaps. Subsequent to the workshop VVER subject matter experts developed detailed technical descriptions of the gaps, including a statement of the issue, descriptions of both operating experiences and recent research results that are relevant to the issue. In several cases strategies for gap closure were included in the descriptions. The resulting set of gaps are systematically categorized as degradation mechanism (DM) gaps, assessment (AS) gaps, mitigation (MT) gaps, inspection and examination (IE) gaps, and repair and replacement (RR) gaps. After collection of the descriptions, the workshop was reconvened to review and develop technical consensus on the descriptions. In addition, in order to facilitate planning for future research programs, within each category, gaps are prioritized into high, medium and low subcategories.

KEY FINDINGS

- There are 37 open R&D gaps in this initial version of the VVER IMTs. These gaps were identified based on comparison with the gaps identified in the PWR IMTs Revision 4 published in 2020.
- A number of gaps (17) that were initially based on PWR gaps were deemed to be not relevant to VVERS. These gaps were collected and included in the section "Closed / Non Relevant" gaps in order to provide and archival record of the discussions and decisions of the VVER IMT workshop.

- The majority of the high priority gaps are Assessment gaps with a significantly lower number of Inspection and Examination gaps. There are then, in turn, fewer Mitigation and Repair and Replacement gaps. This distribution reflects the perspective that mechanisms of degradation in PWRs are relatively well understood and that currently the key gaps relate to the ability to evaluate the extent of degradation and to predict the ability to continue operations without extraordinary actions.

WHY THIS MATTERS

Gap assessment results are a key input to research strategic plans. Gap assessments inform reactor plant management of the technical support behind aging management programs. These gap assessments ensure that utilities continue to meet commitments to implement regulatory aging management programs.

Throughout the consideration of the gaps, a consistent effort was made to ensure that issues important to international members are identified so that R&D proposed for future years can be more effective in addressing international member needs.

HOW TO APPLY RESULTS

VVER program advisors should understand the R&D gaps resulting from this assessment and should consider R&D gap priorities when evaluating prospective program R&D plans.

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PROGRAM: Pressurized Water Reactor Materials Reliability Program (MRP) (P41.01.04)

IMPLEMENTATION CATEGORY: CATEGORY 1

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ACRONYMS AND ABBREVIATIONS

Acronym / Abbreviation	Definition
AMP	Aging Management Plan
AS	Assessment R&D (Gap)
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&PVC	Boiler & Pressure Vessel Code
BOP	Balance of Plant
BWR	Boiling Water Reactor
BWRVIP	BWR Vessel and Internals Program
C&LAS	Carbon and Low Alloy Steel
CASS	Cast Austenitic Stainless Steel
CFR	Code of Federal Regulations
CGR	Crack Growth Rate
Cl	Class
CRD	Control Rod Drive
CRGT	Control Rod Guide Tube
CRDM	Control Rod Drive Mechanism
CS	Carbon Steel
CUF	Cumulative Usage Factor
DBTT	Ductile-to-Brittle Transition Temperature
DM	Degradation Mechanism Understanding R&D (Gap)
DMW	Dissimilar Metal Weld
dpa	Displacements per Atom
EAC	Environmentally Assisted Cracking
EAF	Environmentally Assisted Fatigue

Acronym / Abbreviation	Definition
ECP	Electrochemical Corrosion Potential
ECT	Eddy Current Testing
Emb	Embrittlement
Env	Environment(al)
EOL	End of Life
EPRI	Electric Power Research Institute
EVT	Enhanced Visual Test
Ext	External
F_{en}	Fatigue Environmental Effect Factor
FAC	Flow Accelerated Corrosion
FAT	Fatigue
HAZ	Heat Affected Zone
HCF	High Cycle Fatigue
I&E	Inspection & Evaluation (Gap)
IAEA	International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
IE	Irradiation Embrittlement
IGALL	International Generic Aging Lessons Learned
IGSCC	Intergranular Stress Corrosion Cracking
IMT	Issue Management Table
INPO	Institute of Nuclear Power Operators
ISI	Inservice Inspection
LAS	Low Alloy Steel
LRO	Long Range Order
LTCP	Low Temperature Crack Propagation
LTO	Long-Term Operation
MDM	Materials Degradation Matrix
MRP	Materials Reliability Program
MT	Mitigation R&D (Gap)
NDE	Nondestructive Examination (or Evaluation)
NEI	Nuclear Energy Institute

Acronym / Abbreviation	Definition
Ni-(base) Alloy	Nickel-base Alloy
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUREG	Nuclear Regulatory Commission Technical Report
OD	Outside Diameter
ODSCC	Steam Generator Outside Diameter Stress Corrosion Cracking
OE	Operating Experience
ORNL	Oak Ridge National Laboratory
PSSP	PWR Supplemental Surveillance Program
PWR	Pressurized Water Reactor
PWR MRP	Pressurized Water Reactor Materials Reliability Program
PWROG	Pressurized Water Reactor Owners Group
R&D	Research and Development
RCS	Reactor Coolant System
R.G.	Regulatory Guide
RPV	Reactor Pressure Vessel
RR	Repair / Replacement R&D (Gap)
RT	Radiographic Testing
SCC	Stress Corrosion Cracking
SG	Steam Generator
SGMP	Steam Generator Management Program
SS	Stainless Steel
SSRT	Slow Strain Rate Test
T_{cold}	Cold Leg Operating Temperature
T_{hot}	Hot Leg Operating Temperature
Th	Thermal
TLAA	Time Limiting Aging Analysis
TR	EPRI Technical Report
TS	Technical Support (MRP or SGMP)
TSP	Tube Support Plate
TT	Thermal Treatment / Treated

Acronym / Abbreviation	Definition
TTS	Top of Tube Sheet
UNS	Unified Numbering Systems
USE	Upper Shelf Energy
UT	Ultrasonic Testing
VS	Void Swelling
VT	Visual Test
VVER	“Vodo-Vodyanoi Energetichesky Reaktor” translates to “Water-Water Power Reactor“
WCAP	Westinghouse Commercial Atomic Power
WRTC	EPRI Welding and Repair Technology Center
xLPR	Extremely Low Probability of Rupture.

DEFINITIONS

The following set of definitions are provided to clarify some of the terminology used within the gap descriptions and the IMTs. This listing is not intended to be a comprehensive glossary of technical terms, but rather a key listing of terms as they are used in this report which are helpful to the user in understanding the IMTs.

Beltline, as defined in the United States Code of Federal Regulation (10CFR50), is the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

A ***Casting*** is an item at or nearly at a finished shape obtained by solidification of a substance in a mold.

Cladding (clad) is a thin layer of corrosion resistant alloy, either stainless steel or nickel-base alloy material, applied to the interior surfaces of a carbon or low-alloy steel component in contact with reactor coolant.

Class 1 is the Class 1 primary coolant pressure-retaining boundary.

Consequence of Failure describes the operational results that occur when a component item ceases to perform its intended function.

Degradation Mechanism is a term describing the degradation processes that are applicable to NSSS materials.

A ***Dissimilar Metal Weld*** (DMW) is a weld joining (a) carbon or low alloy steels to high alloy steels, (b) carbon or low alloy steels to high nickel alloys, or (c) high alloy steels to high nickel alloys.

Extended Beltline is a general term, denoting components that, although they are outside the beltline region (as defined above), have accumulated neutron fluence exceeding 10^{17} n/cm^2 (E > 1MeV)

Fitting is a manufactured component used in piping systems to effect a change in fluid flow direction, pipeline size or a special connection. Such components may be manufactured by processes similar to those used in manufacturing pipe. While fittings are generally not identified as separate IMT items a notable exception is made for CASS reactor coolant loop fittings which have shown thermal aging characteristics different from the associated wrought piping sections.

Fluence (neutron fluence) is the time-integrated neutron flux. Units of fluence are neutrons per cm^2. Unless otherwise noted, neutron fluence values are reported based on fast neutron fluence only (E > 1.0 MeV).

Flux (neutron flux) is the product of neutron density and neutron speed. Units of flux are neutrons per (cm^2•sec). Unless otherwise noted, neutron fluence values are reported based on fast neutron fluence only (E > 1.0 MeV).

Forging is plastically deforming metal, usually when the metal is hot, into a desired shape by means of localized compressive forces. These forces can be exerted by presses, special forging machines, or by manual or power hammers. A section of material produced by this process and used for (e.g. pressure retaining) reactor components is also referred to as "a forging".

Fracture Resistance is a generic term for measures of a material's resistance to extension of a crack.

A ***Gap*** is an identified segment of technical knowledge where additional research & development is considered to be warranted to improve the fundamental understanding of degradation or to develop improved solutions for prevention of materials degradation or for the detection of degraded conditions.

Heat-Affected Zone (HAZ) describes the area of base material that has had its microstructure and properties altered primarily by welding, but also possibly from heat intensive cutting operations.

Low-Cycle Fatigue is used to describe fatigue concerns associated with low numbers of fatigue cycles, well below one million cycles, where low-cycle design fatigue curves are applicable.

Nozzles are defined in the IMTs to include full penetration welded vessel penetrations, sometimes with thermal sleeves or other features. The IMTs distinguish between full penetration welded nozzles and other vessel penetrations using partial penetration welds. See "penetrations" below. Examples of nozzles include reactor vessel feedwater and main steam nozzles.

Passive describes components that perform their intended function(s) without moving parts or without a change in configuration of properties.

Penetrations are defined to include vessel connections which use tube inserts and partial penetration welds (e.g. are not full penetration welded nozzles). Examples of penetrations include CRDM penetrations, pressurizer heater penetrations and various instrument penetrations.

Piping Component is general term used in the IMTs to describe pipe segments, pipe fittings, branch connections, welded attachments, thermowell bosses and thermowells.

Qualification is a term used generally in the IMTs to describe performance demonstrations for NDE technologies, procedure qualification, and NDE personnel certification.

A ***Safe End*** is a transition piece welded to the terminal end of a "nozzle" or "branch connection" prior to connection with the external piping.

Stress Improvement is a process that produces sufficient compressive stress on the inside diameter wetted surface to inhibit initiation and propagation of stress corrosion cracks. Stress improvement activities that include welding are considered to be repair / replacement activities. Stress improvement activities that do not include welding are considered to be mitigation activities.

A ***Thermal Sleeve*** is a thin sleeve provided inside a nozzle or branch connection to mitigate sudden large thermal shocks at a physical junction when fluids of highly dissimilar temperatures are mixed.

The term ***Welded Attachment*** is generally used in the IMTs to describe groups of components which are welded to the interior or exterior surfaces of primary pressure-retaining components.

A ***Wrought*** alloy is a metal alloy that has been mechanically worked following an initial casting step.

CONTENTS

LIST OF FIGURES

LIST OF TABLES

1 INTRODUCTION

1.1 Purpose

The purpose of the VVER[1] issue management table (IMT) is to proactively identify and prioritize VVER materials degradation knowledge state and management capability R&D gaps. The results are considered in development of R&D plans and in prioritization of R&D projects to meet EPRI member needs. For utilities in the operating VVER fleet, the VVER IMT is an essential tool used to identify commitments identified in the International Generic Aging Lessons Learned (I-GALL) document.

1.2 Background

The development of the VVER Issue Management Tables is based upon the pre-existing PWR Issue Management Tables. The initial PWR IMT development had originally included comprehensive identification of all of the major PWR nuclear steam supply system subcomponents. For each subcomponent, the applicable degradation mechanisms, consequences of failure, mitigation techniques, repair / replacement approaches and inspection & evaluation guidance were summarized. From this basis, gaps in the current state of knowledge with regard to understanding relevant degradation mechanisms or managing materials degradation were identified in a set of “R&D gaps.”

The initial version of the PWR IMTs was completed in 2006 [1], with subsequent revisions published in 2008 (Revision 1) [2], 2010 (Revision 2) [3] and 2013 (Revision 3) [4]. In 2020 both the PWR and BWR IMTs were updated to Revision 4 [5], [6]. Essentially, from Revision 4, the PWR and BWR IMTs consisted of prioritized technical gaps reflecting the state of knowledge supporting the aging degradation of these reactors’ materials and components which is the format employed herein for the VVER IMTs. However, since background data from the original documentations, for example the “Component Tables”, may provide valuable perspective for use with the VVER IMTs those older documents are provided as references here.) The IMT developed now for VVERs provide a parallel set of technical gaps for the VVER designs and assessments of their prioritizations for them to be addressed by research programs.

[1] VVER, “Vodo-vodyanoi Energetichesky Reaktor” translates to Water-Water Power Reactor. Represents a series of water reactors originally developed in the Soviet Union and now Russia by OKB Gidropress, the Russia Design Bureau Hydropress. Sometimes VVER is represented as WWER to reflect the original Russian pronunciation; however, the VVER form will be used in this document

1.3 VVER IMT Content Management and Approach Changes

The IMTs include several significant and important changes in approach that users familiar with prior versions of the IMTs will notice. These most significant changes are summarized below. See the Appendix A revision log for additional changes.

1.3.1 Development of VVER Technical Gaps

Because of the similarity in concept and operation of VVERs and Western Style PWRs, EPRI proposed an approach to developing Issue Management Tables for VVERs by using the IMTs established for Western Style PWRs as the basis for developing parallel issues for VVERs. This approach would avoid the considerable efforts needed to develop detailed component tables for the VVERs. To test the validity of this approach a workshop was held in Prague, Czech Republic, in November 2019 with attendees from VVER plant sites, research institutes and engineering consulting companies in the Czech Republic, Hungary, Ukraine and Finland. At this workshop all of the current technical knowledge gaps identified for Western Style PWRs in the latest revision of the PWR IMTs were discussed. Specifically, the workshop identified the relevance of these gaps to VVER aging management and how the gaps could be adapted to provide parallel technical gaps for the VVERs. Based upon this workshop the team of experts identified that VVER specific descriptions, relating plant OE and related research results could be developed for VVER materials and components in the same manner that they had been for PWR materials and components.

In 2020 individual technical descriptions were developed for the identified technical gaps by technical experts from the VVER communities in the Czech Republic, Hungary, Finland, and the Ukraine. These initial descriptions were shared with all of the workshop representatives and comments and modifications were invited. Once consensus for the technical gaps had been developed, they were presented to an on-line workshop (made necessary by the 2020 global pandemic) where the technical descriptions' contents were refined and assessments of the priorities of the gaps with respect to VVER plant aging management strategies were identified. The descriptions were then consolidated and compiled into this document.

1.3.2 Component Table

Component tables have been omitted from the VVER IMTs.

1.3.2 Regulatory Issues Tracking

Prior to Revision 4 of the PWR and BWR IMTs a separate R&D gap category for regulatory gaps was included in the IMTs. This section was, however, incomplete in that it primarily reflected a United States regulatory perspective. More importantly, however, it represented the reformulation and presentation of previously developed technical arguments in forms that would address regulators concerns. Thus, while these gaps did require the industry's technical advisors to spend technical efforts formulating and explaining the previously generated research results, they represented the resolution of concerns of local regulators that did not require new research work. In going forward EPRI identified that regulatory issues should be removed from the IMTs and maintained as internal EPRI document documents to tracking local regulators concerns and to identify where consistent explanations could be employed. To this end the former regulatory gaps will be identified as Regulatory Issues and maintained in a separate, proprietary document.

Because of the similarity in design and operations of VVERs and Western Style PWRs the VVER regulatory issues will be maintained in the MRP Regulatory Issues Matrix that mainly addresses PWRs. The identification of regulators issues for VVERs will be performed after the development of this VVER Issue Management Tables Document.

1.3.3 Closed R&D Gaps

The later revisions (1, 2, 3 and 4) of the PWR and BWR IMTs contain listings of previous technical gaps that were opened in response to operating experiences and then closed as a result of research progress. In the first revision of the VVER IMTs there has been no time directed research to close any opened technical gaps based on research results. However, because the technical gaps for the VVER IMTs were based on the list of technical gaps that had been identified for western style PWRs it was possible to identify a set of these gaps that were not relevant to VVERs. This identification was performed by the workshop team during analysis and assessment of the proposed gaps. In a similar manner to the development of the descriptions for the opened technical gaps complete technical discussions were developed for the non-relevant gaps.

The non-relevant gaps and their technical descriptions are included in this document as Appendix B. This appendix documents the history of the workshop discussions on the relevant of the gaps proposed from the PWR gaps. The appendix will also serve as a placeholder for closure of gaps in future editions of the VVER IMT. For these reasons the gaps are identified as “Non-relevant / Closed”

1.3.4 R&D Gap Focus

The definition of an R&D gap carried forward from the BWR and PWR IMTs is specifically a description of the technical issue, a summary of operating experiences and a discussion of the up to date research results that pertain to the issue. This approach supports IAEA aging management goals for VVER reactors. The VVER R&D gaps have been developed to capture VVER operating experiences and research results relevant to the materials and operating conditions of the VVERs. In particular, in developing the technical gap descriptions, emphasis was laid on collecting all of the technical results published in the eastern European technical literature on issues, specifically those that may be unique to the VVER materials and reactors.

1.4 Implementation Requirements

This report is provided for information only. Therefore, the implementation requirements of Nuclear Energy Institute (NEI) 03-08, Guideline for the Management of Materials Issues [7] and international statutes, are not applicable.

2
SCOPE AND APPROACH

2.1 Evaluation Scope

2.1.1 Systems and Components

The systems assessed in this IMT report include VVER components within the Class 1 Pressure Boundary (e.g. reactor coolant pressure boundary components, steam generator tubing, and the reactor vessel internals). In addition, secondary side steam generator components are included, as failures of steam generator secondary side components could affect steam generator tube integrity. Essentially all of the components included in this assessment of the VVER plants are included by analogy with the Western style PWR plants and the genesis of their IMTs.

Consistent with the typical approach for defining the scope of materials management programs for western style PWRs, only passive components (i.e., components that perform their primary function without motion or a change in state) and long-lived components (i.e., components that are not periodically replaced with new or refurbished components) are included in the scope of this assessment. Active components (e.g., control rod drives, valve internals and pump rotating assemblies) and short-lived components (e.g., fuel assemblies and in-core instruments) are excluded from the evaluation. RPV components within scope of the evaluation include vessel shells, top and bottom heads, flanges, nozzles, nozzle safe ends, reactor vessel upper head penetrations including housings for control rod drives, bottom head penetrations, instrument penetrations, and ID and OD welded brackets and supports.

Identification of components within the scope for reactor vessel internals is also by analogy with those considered for the western style PWR IMTs. The identification of those components was based on consequence of failure evaluations documented in EPRI Reports MRP-156 [8] and MRP-157 for western style PWRs [9]. Reviews of the degradation modes and consequences of failure at the 2019 workshop indicated that these issues also pertained to VVERs and that the same approach could be employed for the VVER IMTs as for the PWR and BWR IMTs. For the VVER reactor designs considered (VVER213, VVER440 and VVER1000), reactor internals evaluated include the lower core support system, core baffle assembly, upper core support systems, fuel supports, control rod guide tubes (CRGTs) and control rod systems, surveillance capsules and holders, and guide rods.

Other components of the primary systems including all of the subcomponents of the steam generators (SGs) and pressurizer components are also included in the evaluation scope. Additionally, secondary side steam generator components are included as failure of the steam generator secondary side components could affect the integrity of the steam generator tubes. Note that the horizontal Steam Generator system for VVER has significant differences to the U-tube design of the PWR Steam Generators and nickel base alloys, like Alloy 600, are not employed in VVERs. Thus, several technical gaps were specifically identified for the VVER

Steam Generators and several gaps specific to the PWR Steam Generator design were eliminated.

2.1.2 Operating Conditions

In a similar manner to the development of the PWR IMTs aging degradation was considered to occur under all normal and reasonable operating conditions. The impacts of these conditions were considered in identifying the expectations of degradations in developing the VVER IMT gap assessments. These include startup / shutdown operations as well as normal power operations. “Normal operating conditions” are defined to also include less controlled environmental conditions, such as significant oxygen ingress that may occur during maintenance periods and persist for some time during startup as well as water chemistry transients that have been historically known to occur (e.g., condenser tube leak, resin intrusion events). All likely operational variations are considered, including power uprate conditions and flexible power operations.

The gap assessment is based on the assumption of extended operation, defined as anywhere from 40 to 60 or more years of service. Gaps consider the expected fluence accumulation at the currently projected end of plant operations.

2.1.3 Materials of Construction

The major difference between the VVER and PWR considerations for the IMTs are in the materials of constructure. Low Alloy Steels are employed in the pressure vessel components of both the VVERs and the PWRs and austenitic stainless steels are used for high irradiation bearing internals components as well as piping and other fittings in PWRs and VVERs. However, the specific types of steels employed in the VVERs are slightly different being based on Russian grades of steels. The major difference in the low alloy steels is that the grades of steel (15Ch2MFA, 18CH2MFA, 25Ch3MFA) employed in VVERs are produced by different processing (Ca-killed rather than Al-killed or Si-killed) The intent of the alloying chemistry and thermomechanical processing is, however, to produce a refined bainitic microstructure that is needed to impart the required toughness in the steels used in pressure retaining configurations. In this way the properties and behaviors of VVER LAS are still very similar to those of the LASX used in Western style PWRs. Nonetheless potential differences in properties, behaviors and performance should be taken into consideration when anticipating issues for VVERs.

Similarly, the stainless steels used for VVER internals and piping are titanium stabilized grades, e.g. grade 08CH18N10T, which is similar to the western grade 321 whereas PWRs most widely employ non-stabilized grades of stainless steels including grades 304, 304L, 316 and 316L although some use has been made of the Niobium stabilized grade 347, generally as baffle bolts in some units. Although the properties of the Russian specified materials are reported to be very similar to those of the grades used in western style PWRs, in developing the IMTs for VVERs attention was paid to potential differences in the specific properties and performance of the materials of construction of the VVERs. The VVER gap evaluations explicitly considered the alloys and variants used in VVER plants operated in the Czech Republic, Hungary, Ukraine and Finland. Implicitly, the gap evaluations also consider the presence and performance of welds, including similar metal and dissimilar metal welds (DMWs). These evaluations also took into account the behavior of weld-associated heat affected zones.

2.2 Evaluation Approach

Section 3 of the VVER IMTs documents the results of the R&D gap assessment. Overall methodology and guiding principles used for gap identification, categorization and prioritization are provided in the following sections.

To develop the list of potential gaps for VVER issues EPRI IMR and MRP technical staff provided the listing of active gaps that were present in the latest revision (Rev 4, 2020) of the PWR IMT MRP-205. These gaps were reviewed and discussed at special workshop that was held in the Czech Republic in November 2019, and attended by VVER experts from the Czech Republic, Hungary, Finland and the Ukraine. This workshop identified which PWR gaps were relevant to VVER operating experience and issue managements. Correspondingly it also identified which PWR issues would not be relevant for VVERs. Additionally, six more VVER specific issues (based on the design features of the VVER plans and the materials of construction were also identified as VVER specific issues.

Following the workshop, the individual gaps were assigned to cognizant VVER subject matter experts for the development of detailed technical descriptions for both relevant/open and non-relevant/closed gaps. Technical descriptions (generally termed as “write-ups” in Figure 2-1) consisted of issue descriptions, operating experiences and expected consequences of inattention to the gaps, relevant research results, recommendations for research actions and comprehensive technical references pertaining to the issues. In developing the review of the research background particular attention was paid to collecting information from the eastern European and Russian documents that are relevant to the VVER experiences. The set of technical descriptions were delivered to EPRI in the third quarter of 2020.

The technical descriptions were presented to a re-convened workshop that, due to circumstances in 2020, was held virtually. Prior to the reconvened workshop all of the draft VVER gap descriptions were distributed to the VVER IMT workshop team for review and comment. At the reconvened workshop, held in November 2020 final comments and recommendations items to be included in the descriptions were made by the workshop team. The meeting also identified consensus prioritizations for the proposed gaps. After the workshop, the individual gaps were reassigned to the VVER subject experts for editing and inclusion of more technical factors as identified at the online workshop. The final gap descriptions were delivered to EPRI in the first quarter of 2021 for compilation into a formal EPRI IMT document.

Figure 2-1 provides a summary view of the process employed to develop the VVER IMTs in 2019 through 2021.

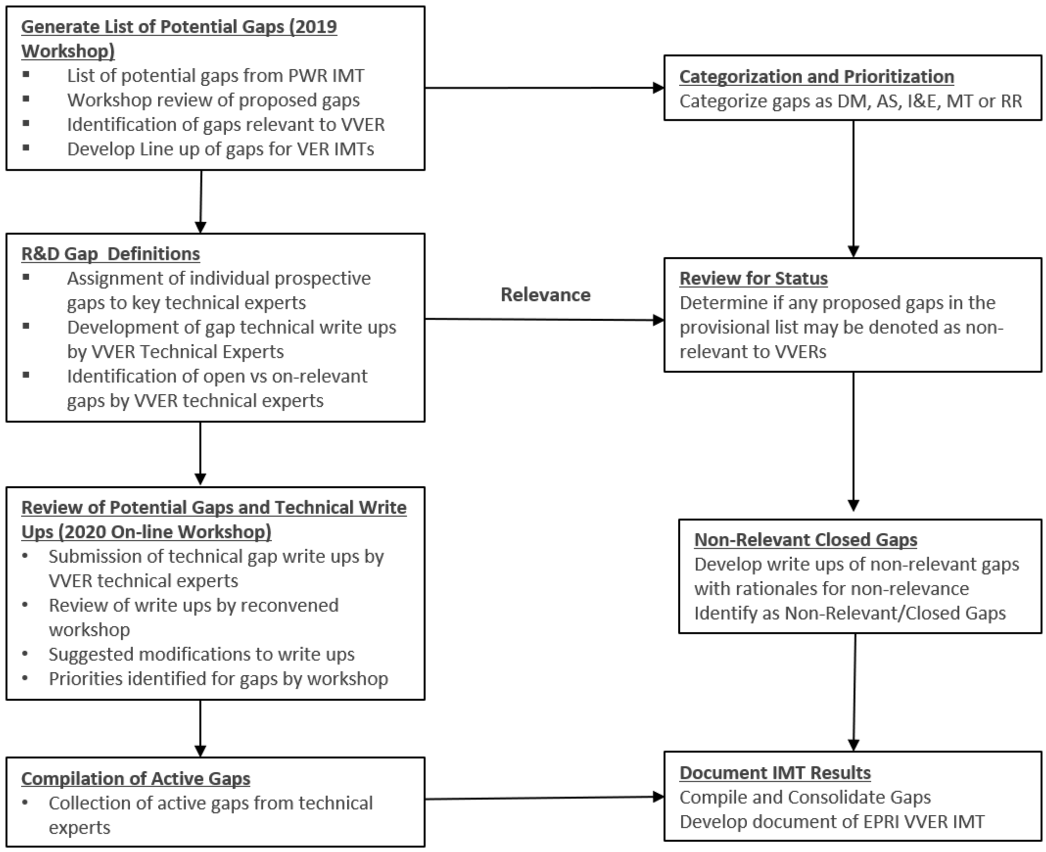

Figure 2-1
Overview of EPRI Issue Review Process For Developing VVER IMT R&D Gaps and Final Documentation

2.3 R&D Gaps

2.3.1 R&D Gap Definition

An R&D "Gap" is defined as an area of research identified as important to achieving a reasonable standard of confidence that primary system component degradation can be managed such that components will remain serviceable and capable of performing their intended functions for the remainder of plant life. R&D gaps address issues that could represent direct challenges to nuclear safety, that have the potential to result in events that challenge the operability of safety-related systems, or that represent challenges to long-term aging management. R&D gaps may encompass more than one technical area and solution paths should NOT be predefined. For example, a gap could guide R&D from multiple disciplines including structural and metallurgical engineering (analyses intended to quantify available structural margins), and NDE (development of methods to better detect and characterize degradation).

R&D gaps for VVERs were initially identified by EPRI technical staff based on gaps previously identified for PWRs and then adapted for VVERs using the input of VVER subject matter experts.

2.3.2 R&D Gap Categories

R&D gaps are categorized into the following groups based on the technical focus of the gap:

Degradation Mechanism Understanding (DM) Gaps
Degradation Mechanism understanding gaps are identified using data obtained from the EPRI Materials Degradation Matrix (MDM) [10]. These degradation mechanisms are shown as "?" and highlighted in blue in the MDM.

Assessment (AS) Gaps
Assessment gaps are associated with R&D needs related to characterizing the potential impact of a degradation mechanism shown to be applicable to the PWR operating environment. Additionally, assessment may be needed to determine the proper approach for management of a degradation mode or to develop additional data to better characterize and manage a known degradation issue.

Mitigation (MT) Gaps
Mitigation gaps are associated with R&D needs in the area of new technology development or verification of technique effectiveness for preventing degradation mechanism initiation or limiting degradation mechanism progression. Gaps may involve mitigation of degradation via either water chemistry or mechanical means (e.g., surface stress modification).

Inspection & Evaluation (I&E) Gaps
Inspection & evaluation gaps are associated with component inspection guidance, NDE qualification, or development of new NDE technology to effectively detect and size indications in degraded components.

Repair / Replacement (RR) Gaps
Repair / Replacement gaps are associated with needs for further development or verification of the effectiveness of repair techniques.

2.3.3 R&D Gap Identification Numbers

Each gap ID number includes three aspects:

Design Type: VVER IMT gaps are prefixed with "V-" to associate them with the VVER IMTs and prevent confusion with Western Style PWR and BWR IMT gaps in documentation that links IMT gaps to ongoing research projects, planned research projects, and research proposals. This may occur for Issue Programs with responsibilities bridging VVER, PWR and BWR designs, such as the EPRI Weld Repair Technology Center or the Water Chemistry Program.

Gap Type: Each gap includes a category identifier consistent with the R&D gap categories presented in Section 2.2.2, AS, DM, IE, MT or RR.

> Sequential Numbering: Within each gap category, gaps are sequentially numbered. Note that these sequential numbers are never reused, even if the gap is closed. This approach reduces confusion when referring to gaps by ID number only.

Figure 2-2 illustrates the elements of the R&D Gap identification number.

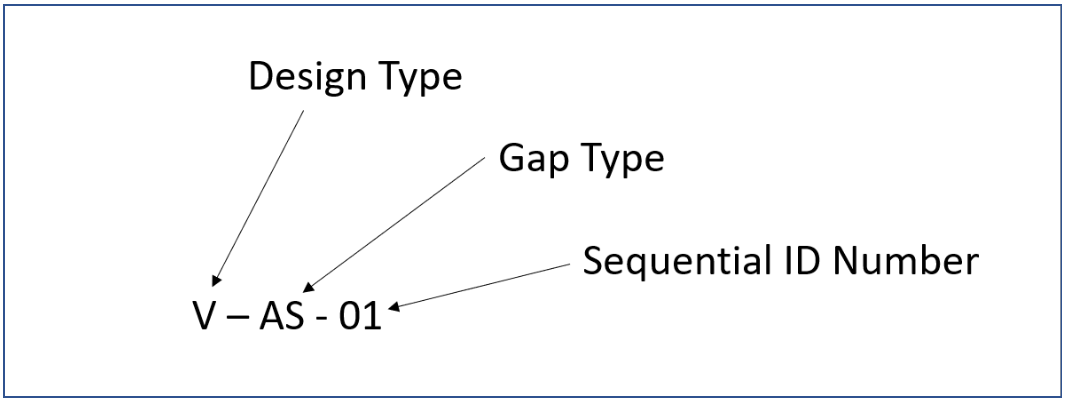

Figure 2-2
Gap ID Nomenclature

2.3.4 R&D Gap Status and Priority

The statuses of gaps are given in the IMT report and will be maintained in the interim gap listings on the issue program website as open, closed, or new:

- Open: Applies to gaps opened in this first version (“Rev. 0”) of the VVER IMT report.
- Closed: Gaps which are designated as “Closed” in IMTs are usually gaps that were opened in previous versions of the IMTs and have been closed due to progress in technical knowledge. In this first version of the VVER IMT this category is retained to cover those proposed technical gaps which were taken from the PWR IMT gaps but which were deemed to be not relevant to the VVER reactors. For this purpose they are identified as “Not relevant / Closed” in this first version of the VVER IMT’s. The selection will serve as a place holder for gaps that will be closed by technical progress in future versions of the VVER IMT reports. A listing of these technical gaps considered to be not relevant is documented in Appendix B Table B-1 of this report. The “Non-relevant / Closed” gaps are also listed in the summary of all the gaps in Table 3-1 of this report for completeness.
- New: Applies to gaps which have been identified since the last issued revision of the IMT. Since this is the first version of the VVER IMTs this category is not used in this revision of the VVR IMTs. It is held as a placeholder for future versions of the VVER IMTs.

R&D gaps that do not have a closed status are prioritized into High, Medium, and Low priority categories. These categories have been proposed by EPRI staff and the individual VVER technical experts and confirmed by the reconvened workshop in 2020.

2.4 Regulatory Issues

Based on the R&D gap definition planned for Revision 4 (see Section 2.1 above), all R&D gaps should represent missing knowledge that provides a barrier to effective resolution of issues during plant aging. Using this approach, all R&D gaps are “technical” in nature and it should be possible to define these gaps from a fundamental research perspective using only technical rationales and without reference to regulatory authorities, regulations, or regulatory policies.

Work to close the gaps should be pursued as resources are available regardless of regulatory perspectives and inputs. However, it is recognized that implementation of solutions to technical issues frequently cannot occur until regulatory approval has been obtained. Cases for which technical solutions have been obtained but are awaiting regulatory approval or concurrence are now identified as “Regulatory Issues”.

Oftentimes developing regulatory acceptance of the technical solutions involves the preparation of in-depth technical arguments and technically based work must be performed to develop the appropriate technical arguments for submission to the regulators. This represents work that EPRI would otherwise not undertake but does so because of the significant benefit to members to have comprehensive solutions accepted by the regulators. These issues do not constitute true technical gaps since by closure of the technical gaps the issues have been resolved. However, EPRI believes that there is value in maintaining a database of these open concerns of the regulators. EPRI recognizes that the concerns may be different for different regulators around the world. Nevertheless, EPRI perceives that there will be some synergy is concerns from groups of regulators can be addressed in a coherent and collaborative manner. On this basis EPRI believe it is reasonable to track these issues in a coordinated manner. To track these issues and to provide a basis for assigning resources EPRI will tracking the regulatory issues for VVERs in a global Regulatory Issues Matrix which will be available to members on the issue program website.

2.6 Component Tables

Through Revision 3, the IMTs had contained a set of tables that listed all of the relevant PWR components within the scope of the IMT evaluation. However, these tables are not essential to performance of a comprehensive assessment of R&D gaps nor to meeting the intent of NEI 03-08. The tables can be found in previous versions of the IMT report. Since there are great similarities between the components and generic materials of western style PWRs and VVERS there is little value in repeating the extensive process for VVERs. Operators of VVERs may refer to a streamlined version of these tables on the issue program websites or to the archived copies of the PWR IMT reports for more complete discussions.

3
R&D GAPS

The gap assessment results from the IMT process are provided in Tables 3-1 through 3-6. Table 3-1 contains a summary listing of all the R&D gaps that are were considered for opening in Revision 0 of the VVER IMTs. Tables 3-2 through 3-6 provide, for each gap type (DM, AS, I&E, MT and RR) a description of each gap that the workshop team considered as relevant issues for VVERs and for which gaps should be opened. (Those gaps that were considered by the workshop, but were identified as not being relevant for VVERs listings and technical discussions are provided in Appendix B. The gaps there are considered as “non-relevant/closed”.) It is expected that this section will be a placeholder to track gap closures resulting from research progress in future revisions of the VVER IMTs.

Table of all gaps considered in this revision, Rev 0, of the VVER IMTs, together with their currently assigned priorities. Also included in this list are the gaps that were flowed down from the PWR IMT but were considered by the workshop to be “non-relevant” to the VVER IMTs.

Table 3-1
Summary of R&D Gaps and Gap Priorities (Listed in order of Priority)

Degradation Mechanism (DM) Gaps		
Gap ID	**Gap Description/Title**	**Priority (R0)**
V-DM-01	**Environmental Effects on Fracture**	High
V-DM-03	**Long Term SCC Susceptibility**	Medium
V-DM-04	**Steam Generator ODSCC at Extremely Highly Alkaline pH**	Medium
V-DM-02	**SCC of Thermally Aged CASS Pressure Boundary Components**	Low
V-DM-05	**Increased Fastener SCC Susceptibility due to Long Term Aging**	Low
V-DM-07	**Hydrogen Embrittlement of Piping between Pressurizer and Safety Valve**	Low
V-DM-06	**Thermal Embrittlement of Low alloy Pressure Vessel Steels**	Not Relevant/ Closed

Table 3-1 (continued)
Summary of R&D Gaps and Gap Priorities (Listed in order of Priority)

Assessment (AS) Gaps		
Gap ID	**Gap Description/Title**	**Priority (R0)**
V-AS-01	**Reactor Internals Aging Management Program during LTO period**	High
V-AS-05	**IASCC Characterization: Generic Data Needs**	High
V-AS-06	**IASCC Characterization: Baffle Bolting**	High
V-AS-07	**Void Swelling of Stainless Steels**	High
V-AS-09	**Fluence Impact on Stainless Steel Mechanical Properties (Fracture Toughness, Tensile Strength)**	High
V-AS-18	**Environmental Effects on Fatigue Resistance: Pressure Boundary Components**	High
V-AS-19	**Environmental Effects on Fatigue Resistance: Reactor Internals**	High
V-AS-28	**Methods for Determination of Locations of Class 1 Piping Susceptible to Thermal Fatigue**	High
V-AS-03	**SCC of Stainless Steels Exposed to Primary Water**	Medium
V-AS-04	**Thermal & Irradiation Embrittlement Combined Effects on SS Welds**	Medium
V-AS-08	**Steam Generator Sludge Deposits and Scale Buildup**	Medium
V-AS-26	**Lack of sharing information of latest updated and developed probabilistic fracture mechanics codes through benchmarking**	Medium
V-AS-27	**Fatigue Usage at J-Welds in CRDM Penetrations**	Medium
V-AS-10	**Low Temperature Crack Propagation (LTCP) Assessment**	Low
V-AS-14	**Neutron Embrittlement of Reactor Pressure Vessel Steels**	Low
V-AS-17	**Fracture Toughness Properties of Low Alloy Vessel Steels (Plates and Forgings)**	Low
V-AS-20	**Flow Induced Vibration and Wear of Reactor Internals**	Low
V-AS-02	**Improved Method of Calculating Accumulated Fluence in RPV and Reactor Vessel Internals Structures**	Not Relevant/ Closed
V-AS-11	**Cracking of Steam Generator Collector Bridges**	Closed/Revised as V-IE-06
V-AS-12	**ODSCC of Stainless Steel Steam Generator Tubing at High Alkalinity**	Not Relevant/ Closed
V-AS-13	**Steam Generator Flow-Accelerated Corrosion Assessment**	Not Relevant/ Closed
V-AS-15	**Fluence Spectra and Dose Rate Effects on Low-Alloy Steels RPV Materials**	Not Relevant/ Closed
V-AS-16	**Neutron Embrittlement of Nozzle Forgings and Upper Shell Course**	Not Relevant/ Closed
V-AS-21	**High Cycle Fatigue Potential at Branch Line Locations**	Not Relevant/ Closed
V-AS-22	**Outstanding Issues Associated with Thermal Fatigue of ASME Class 1 Piping**	Not Relevant/ Closed
V-AS-23	**Lack of Appropriate Methods for Evaluating Low-Cycle Fatigue of Elbow Pipes with Welds under Seismic Loading**	Not Relevant/ Closed

Table 3-1 (continued)
Summary of R&D Gaps and Gap Priorities (Listed in order of Priority)

Assessment (AS) Gaps (continued)		
Gap ID	**Gap Description/Title**	**Priority (R0)**
V-AS-24	**Lack of Methods for Evaluating Non-proportional Multiaxial Low Cycle Fatigue of Pipes under Seismic Loading**	Not Relevant/ Closed
V-AS-25	**Lack of an Advanced Method for Determining the Number of Fatigue Cycles as Part of the Low Cycle Fatigue Evaluation for Piping under Seismic Loading**	Not Relevant/ Closed

Inspection and Evaluation (IE) Gaps		
Gap ID	**Gap Description/Title**	**Priority (R0)**
V-IE-06	**Detection of Cracking of Steam Generator Collector Bridges (From closed V-AS-11)**	High
V-IE-04	**Steam Generator Tubing Eddy Current Technology Improvements**	Medium
V-IE-02	**NDE Accessibility for Reactor Internals**	Low
V-IE-03	**NDE Technology for Examination of CASS**	Low
V-IE-01	**NDE Technology for Detection and Characterization of Baffle and Former Assembly IASCC**	Not Relevant/ Closed
V-IE-05	**NDE Qualification for Reactor Internals Inspection (VT Evaluation)**	Not Relevant/ Closed

Mitigation (MT) Gaps		
Gap ID	**Gap Description/Title**	**Priority (R0)**
V-MT-04	**Loss of Critical Chemical Supply for Required Chemical Mitigation**	High
V-MT-01	**Steam Generator Tubing ODSCC Mitigation via Water Chemistry Technologies**	Medium
V-MT-02	**Steam Generator Startup Chemistry Excursions after Major Component Replacement**	Medium
V-MT-03	**Guidance for Extended Layup of SGs and BOP Systems**	Medium
V-MT-07	**Steam Generator Sludge Deposits and Scale Buildup**	Medium
V-MT-05	**Mitigation of SCC in Nozzle DM Welds – To avoid Stainless Steel Cracking Propagating into LAS Pressure Boundary Materials**	Low
V-MT-06	**Develop Recommendations and Guidance for Mitigating Fatigue Failures in Piping**	Not Relevant/ Closed

Table 3-1 (continued)
Summary of R&D Gaps and Gap Priorities (Listed in order of Priority)

Repair and Replacement (RR) Gaps		
Gap ID	**Gap Description/Title**	**Priority (R0)**
V-RR-03	**Welding Process for Repair of Irradiated Material of RPV Internals**	Medium
V-RR-06	**Validation of Nickel Plating DMW Surfaces to Protect against SCC in Secondary Water**	Medium
V-RR-01	**(Fleetwide) Replacement Strategy for Baffle Bolts**	Low
V-RR-04	**Repair Guidelines for Reactor Internals**	Low
V-RR-02	**Repair/Replacement Guidance for Thermal Fatigue of ASME Class 1 Piping**	Not Relevant/ Closed
V-RR-05	**Alternative DM Weld Repair Solutions**	Not Relevant/ Closed

Table 3-2
Degradation Mechanism (DM) R&D Gaps

Gaps related to understanding of the degradation mechanisms of components. (Note that although DM is out of the normal, alphabetical order of gaps, the IMT considers this set of gaps first, since the assessment, inspection and evaluation, mitigation and repair and replacement gaps can only be properly considered once the active mechanism of degradation have been identified and defined.)

V-DM-01 – Environmental Effects on Fracture **Issue:** Reactor vessel internal materials of VVER can exhibit some reduction in fracture resistance properties due to exposure to the reactor coolant medium. **Description:** In the case of VVERs environmental effect on fracture is relevant failure mode only for reactor internal (RVI) materials, i.e. Ti stabilized stainless steel and its welds, martensitic stainless steel (only in VVER-440) and PH high Ni stainless steel materials. RVI materials can exhibit some reduction in fracture resistance properties due to exposure to the primary system coolant. The increased yield strength resulting from irradiation can cause the environmental effect to be more significant. Fracture resistance of the irradiated material can decrease in water environment. It is important to know and to use the real values for crack analyses in the RVI. If any crack will be indicated in the RVI the crack will be filled by primary water environment and it needs to be taken in account in all safety analyses. EPRI report 3002020892 summarizes publicly available data about the principal degradation mechanisms related to neutron irradiation degradation of Titanium and Niobium stabilized stainless steels. It describes effects of the loss of ductility and reduction of fracture toughness due to irradiation, IASCC, void swelling and irradiation creep. Measurements of fracture toughness in primary water environments on VVER RVI materials are limited. As well, the database for environmental effects on fracture toughness in the irradiated condition is insufficient. Assessment is presently based on standard (short term) tests in air. Some experimental results are available on RVI materials from tests that were performed on active samples from the retired Greifswald NPP. Slow strain rate tests (SSRT) and crack growth rate tests were done in primary water coolant environment at the VVER operation temperature. These test results showed that the active core barrel material is sensitive to EAC in a cyclic loading regime (corrosion fatigue) by the same way as the non-active material. The active core barrel material did not exhibit any increase of environmental effects due to irradiation. The small islands of IG fracture (IASCC) observed on SSRT (slow strain rate test) fracture surfaces are observed on CGR (crack growth rate) test specimens as well. These island areas are very small in relation to the total fracture area and with negligible influence to the material behavior assessment, but it indicates that more IASCC can be expected in highly-irradiated material. R&D projects are in progress on wrought Ti-stabilized stainless steel and its welds and are planned for the martensitic stainless steel and high Ni stainless steel. Further research needs depend on the results of these ongoing projects.	**Status:** Open **Priority:** R0 - High

Table 3-2 (continued)
Degradation Mechanism (DM) R&D Gaps

V-DM-01 – Environmental Effects on Fracture (continued)	**Status:** Open
References: *EPRI Materials Degradation Matrix, Revision 4.* (Notes v4-11a-d) EPRI, Palo Alto, CA: 2018. 3002013781. *Irradiation Effects in Stabilized Stainless Steels,* EPRI, Palo Alto, CA: 2021. 3002020892. • IAEA-SSR-82, Rev. 1, "Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL)," International Atomic Energy Agency, Vienna, August 2020.AMP 113 Reactor Pressure Vessel Internals • AMP 163 Dissimilar Metal Welds • AMP 122 PWR Flux Thimble Tube Inspections Brabec, P., Burda, J., Ernestová, M., Falcník, M., Keilová, E., Kočík, J., Kytka, M., Novosad, P., Málek, P., Pešek, P., Postler, M., Rapp, M., "Final report of project Greifswald - Report No.: DITI 302/520a Rev.0," (2008). Hojná A., Burda J., Ernestová M., Falcník M., Keilová E., Kočík J.: "Material characteristics of Materials from Greifswald non-active samples/ Baffle, Report NRI Rez DITI 302/421 Rev. 1," March 2008. Hojná, A., Intergranular fracture and fracture toughness of irradiated austenitic stainless steels of reactor core internals, in: Proceedings of the 16th International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, 2013.	**Priority:** R0 - High

Table 3-2 (continued)
Degradation Mechanism (DM) R&D Gaps

V-DM-02 – SCC of Thermally Aged CASS Pressure Boundary Components	**Status:** Open
Issue: There is currently a lack of data associated with SCC of Cast Austenitic Stainless Steel (CASS) Components after thermal aging.	**Priority:** R0 - Low
Description: It has been found that the mechanical properties of cast austenitic steels changes significantly over the course of long-term operation. For cast austenitic metal with a ferritic phase content of more than 10%, long-term thermal ageing results in a decrease in impact strength and fracture toughness. This process has a latent period followed by a stage of decreasing impact strength that is caused by the nucleation of precipitates and formation of zones with elastic stresses of the 2nd type. The change in impact toughness of the base cast metal of the main gate valve over time can be determined using the Charpy notch test from samples with either U- or V-notches, constructed according to the proposed model of thermal aging. It has been established that after 45 years of operation, the impact toughness of the cast metal of the main gate valve decreases by 15% in terms of KCU and by 40% in terms of KCV in comparison with the initial certified values. Despite these decreases, as they are both above the 39 J/cm^2 limit, the impact toughness is still within acceptable levels. Deterioration of the mechanical properties occur due to spinodal decomposition. Spinodal decomposition results in elemental redistribution at the nanometer scale and formation of Fe-rich α and Cr-rich α' regions. Precipitation of G-phase has been observed in materials with spinodal decomposition. It is possible that a synergistic effect exists between the precipitation of G-phase and spinodal decomposition. In an investigation carried out to describe the structure of 10Ch18N9TL cast stainless steel, spinodal decomposition in the δ-ferrite was not observed; neither in the cast Ti-stabilized stainless steel material nor in the Mo-alloyed weld metal. No data was therefore obtained on either the intergranular corrosion resistance or crack growth rate. It is therefore recommended that 10Ch18N9TL steel specimens are prepared for fracture testing from representative samples of cast austenitic stainless steel and tests are conducted to determine both its fracture mechanical properties and stress corrosion properties.	
References: Ehrnstén, U., Karjalainen-Roikonen, P., Nenonen, P., Korhonen, R., Timofeev, P. T., and Bloomin, A.A. “Investigations on aged Ti-stabilised stainless steels,” pp. 241-251, VTT Symposium, Espoo, 2003. Petrov, V. A., et. al: “Materials Research Justification of Lifetime Extension of Pressurizer, Main Coolant Pipeline, Pressurizer System Pipelines of VVER-440 Reactor Plant Exemplified by the Condition of Base Materials and Welding Consumables Used in Unit 3 of Novovoronezh NPP After 45 Years’ Operation,” Proceedings of the 11th Conference - Safety Assurance Of NPP with VVER. JSC OKB "GIDROPRESS,” 2019. (in Russian)	

Table 3-2 (continued)
Degradation Mechanism (DM) R&D Gaps

V-DM-03 – Long Term SCC Susceptibility	Status:
Issue: Long-term exposure of materials to environments conducive to SCC may lead to increasing rates of incidents of IGSCC initiation or initiation in materials or components where it has not previously been observed. **Description:** There are several materials and components in VVER reactors that are sensitive to SCC, i.e. primary system piping and pressurizer, steam generator tube bundles and internals, and steam generator secondary pressure boundary. In most of them Ti stabilized stainless steel can be found for which the SCC is one of the critical degradation processes, both for base material, weld as well as cladding. In the case of VVER-1000 Ni alloy of washers and low alloy steel in the SG collector are subjected to SCC. Although for some cases according to MDM there is no need for further research (either because there is enough information about the SCC process, or the SCC sensitivity could be decreased with proper water chemistry), there are several gaps in most cases. Secondary side stress corrosion cracking is a significant degradation mechanism affecting heat exchanger tubes, albeit very much less seriously than for Alloy 600 tubes in PWR SGs. Today, a significant discrepancy exists between accelerated autoclave tests carried out with relatively highly concentrations of impurities and operational experience that indicates low salt impurity concentrations. For life management, it is necessary to be able to identify the beginning of any accelerating trend curve in order to initiate proper and timely corrective measures. Only a limited number of autoclave tests (Czech Republic, Slovakia) are in progress to answer some of these questions. Austenitic stainless steel is affected by SCC from the secondary water side mainly in crevice regions. From an LTO perspective, the SCC performance of tube support plates (TSPs) is uncertain. There is a possible risk of TSP cracking when stress is applied due to deformation in service. This is not supported by operational experience to date, but a high TSP cracking rate could have a detrimental effect on SG lifetimes. Dissimilar metal welds of VVER-440 SG collectors are affected by SCC (or SICC); the cracks were detected in the dissimilar weld area. It appears to initiate in the carbon steel and propagate along the weld metal/C&LAS interface. This degradation is being addressed by a comprehensive research program. Since the investigations performed on the mechanism of cracking have not been published in the open literature, it remains a serious problem for several operating VVER-440s. Information is needed on the cracking mechanism, on material properties, residual stresses, crack growth rates, NDE, LBB issues, and on the consequences of a failure. Other problematic parts are threads at the top of the primary collector in the region of thread holes, since the collector to closure head junction is considered as one of the weakest links in the VVER LOCA analysis. The threaded holes in the upper part of collector have history of cracking at Paks, Rivne, Bohunice, and Dukovany NPPs. In the case of VVER-1000 the cracking of the SG weld No. 111 is being studied today. This is based on operating experience in Russia that has been attributed to poor water chemistry following the accumulation of sludge in a dead-leg crevice. However, at present not enough information is available about this problem. In addition to these gaps, there still remains a concern that a late life upward trend in degradation events related to aging is possible and should be taken into consideration in internals aging management strategies. Aging may be metallurgical in origin and / or due to some slow accumulation of a degrading environmental reaction. Deterioration of the surface condition could lead to crack initiation at progressively lower levels of stress intensity, potentially leading to cracking in locations or components not previously experiencing SCC.	Open **Priority:** R0 - Medium

Table 3-2 (continued)
Degradation Mechanism (DM) R&D Gaps

V-DM-03 – Long Term SCC Susceptibility (continued)	**Status:** Open
A better understanding of all of the variables that are known to impact IGSCC initiation including combinations of materials, fabrication factors, surface treatments and their persistence, the development of oxide layers and environmental conditions would support the development of greater confidence in aging management assessments and could potentially support eliminating or minimizing some inspections. Closure of this gap would require substantial advances in over a broad range of SCC risks at long service times. **References:** *EPRI Materials Degradation Matrix, Revision 4.* (Note Y-v3-6a, Y-v3-6d, Y-v5-7a-f, Y-v6-7a-e) EPRI, Palo Alto, CA: 2018. 3002013781. • IAEA-SSR-82, Rev. 1, “Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL),” International Atomic Energy Agency, Vienna, August 2020.AMP 112 PWR Cracking of Nickel Alloys • AMP 102 In Service Inspections Reactor Coolant Pressure Boundary Components • AMP 101 Low Cycle Fatigue Monitoring • AMP 103 Water Chemistry Guidelines for Integrity and Lifetime Assessment of Components and Piping in WWER Nuclear Power Plant (VERLIFE), 2014. Junek, L., Žďárek, J., “SG indications in DMW of collectors,” IAGE Metal Group 17.-18. April 2012.Lucas, T., The Effect of Thermal Aging and Boiling Water Reactor Environment on Type 316L Stainless Steel Welds, Doctoral Thesis, MIT, Cambridge, MA, 2011. Spisák, B., Siménfalvi, Z., Szávai, Sz., Bézi, Z., “Numerical simulation methods of stress corrosion cracking, Solutions for Sustainable Development,” Solutions for Sustainable Development, Taylor & Francis Group, 2019.	**Priority:** R0 - Medium

Table 3-2 (continued)
Degradation Mechanism (DM) R&D Gaps

V-DM-04 – Steam Generator ODSCC at Extremely Highly Alkaline pH	Status:
Issue: Both VVER440 and VVER 1000 steam generators use titanium stabilized stainless steel tubes. This material is known to be susceptible to SCC initiated from secondary side (outer diameter) in acidic environment, but according to chromium oxide solubility curves the steel is susceptible to degradation in (highly) alkaline pH as well. (Note that there is no corresponding gap in the PWR IMT for this situation.) **Description:** The titanium stabilized stainless steel is extensively used in VVER design and it is also a material of steam generator. Although the material has an outstanding performance record for using in nuclear plants, it is known to be susceptible to SCC in acidic environment. As a result, the secondary chemistry focuses on minimizing the chloride and sulfide ingress to mitigate ODSCC. Also, micro-dosing of alkali hydroxide is used at some plants to shift crevice pH(T) to higher values. To date only the acidic crevice pH(T) was a concern, but extensive retubing of VVER fleet condensers for stainless steel improved secondary chemistry and highly alkalic crevices (according to Hide Out Return calculations) are observed in some plants. There is a need for developing scientifically strong basis for crevice pH control specifying both acidic and alkalic limit to prevent corrosion issues of the steam generator tubes. **References:** *EPRI Materials Degradation Matrix, Revision 4.* (Notes v5-7a, v5-7b) EPRI, Palo Alto, CA: 2018. 3002013781. *Steam Generator Reference Book, Volume 1, Revision 1.* (Table 24-31), EPRI, Palo Alto, CA: 1994. TR103824-V1R1	Open **Priority:** R0 - Medium

Table 3-2 (continued)
Degradation Mechanism (DM) R&D Gaps

V-DM-05 – Increased Fastener SCC Susceptibility due to Long Term Aging	Status: Open Priority: R0 - Low
Issue: There is the possibility that some fasteners could exhibit susceptibility to either IG or TG SCC as a result of long-term exposure to operating temperatures. No study of material degradation of low-alloy steel fasteners is available. An evaluation of this issue is needed to address the potential significance of fasteners degradation, especially crack growth rates. **Description:** Fasteners in VVER plants are made from low alloy steels and stainless steels. Two grades of low alloy steel are used – relaxation resistant steel 25Ch1MF and structural alloy 38ChN3MFA. Austenitic precipitation hardened stainless steel ChN5VT-VD, which is a transition class alloy between highly alloyed SS and Ni-base alloys, is used in some locations [MDM]. Operating experience shows that austenitic precipitation hardened stainless steels are susceptible to localized corrosion and intergranular or transgranular stress corrosion cracking [UJV reports]. It is known that some types of precipitation hardening heat treatment produce material with very high strength, but with less than optimal corrosion resistance. If such material is exposed to an environment with higher concentration of contaminants, notably Sulphur or Molybdenum contained in the lubricant, it may result in IG SCC. In other cases, the stainless steel may have optimum corrosion resistance but still some cases of TG SCC occur. This OE was used for stipulation of the standard heat treatment and limitation of use of certain types of lubricating materials. No OE of degradation of low alloy steel fastener exists. As austenitic steel OE indicates, the fastener materials are susceptible to cracking. For this reason, a robust NDE program exists for evaluation of their fitness-for-service. Several types of NDE methods are employed, including Visual test, Dye penetration test, Eddy current test, Ultrasonic test and others. Each bolt, nut or washer is inspected by at least one NDE method at least once per outage. Fasteners with any type of degradation are removed from service for further destructive examinations. This procedure ensures thorough understanding of fastener degradation and should be sufficient for early detection of any increase of degradation rates. The fasteners are pressure-retaining components, but as reliable NDE methods for their examination exist and they can be easily replaced, no significant research work is envisioned at this point of time. The SCC characteristics of these materials are relatively well understood, but no systematic materials' structure-properties and SCC studies have been performed on long term aged fastener materials at operating temperatures, the potential for late life onset of enhanced SCC in fasteners cannot be ruled out. Also, the study of degradation of low alloy steel fasteners would help for better planning of replacement bolting made from this type of material. This gap is therefore retained as a low priority gap. **References:** *EPRI Materials Degradation Matrix, Revision 4.* (Note v7-5gh) EPRI, Palo Alto, CA: 2018. 3002013781. Ernestova, M., et al.: "Destructive Evaluation of Mechanical and Structural Properties of Temperature Monitoring System Bolts," UJV Report DITI 2.3.2.9.2, 2015. Keilova, E., et al.: "Metallographic Evaluation of Steam Generator Collector Bolt," UJV Report DITI 302/240, 2002.	

Table 3-2 (continued)
Degradation Mechanism (DM) R&D Gaps

V-DM-07 – Hydrogen Embrittlement of Piping between Pressurizer and Safety Valve	**Status:** Open
Issue: The piping for periodical removal of hydrogen gas is part of the line between the pressurizer and safety valve. Material of the piping is titanium stabilized stainless steel. Although no OE exists to date, due to the operational parameters it is not possible to rule out the risk of hydrogen embrittlement of the piping resulting in severe deterioration of its material properties. **Description:** The pressurizer contains steam-gas mixture with high concentration of hydrogen in its upper part. In order to remove hydrogen, the piping for periodic removal has been installed into the system additionally. On some of the plants the hydrogen is removed periodically during unit operation, on others it is performed continually. The temperature of the inactive piping system is close to the temperature of environment (~60 °C) with spikes up to 340 °C during hydrogen removal process. On the other hand, with relatively low hydrogen partial pressure, the risk of hydrogenation remains probably low. It is assumed that hydrogen uptake takes place during operation of the piping, not during manufacturing or welding. Experimental results show that hydrogen damage to the performance of 304 stainless steel is significant even at very low levels. The fractographic analysis indicates the high penetration ability of hydrogen in stainless steel [3002001474]. Nickel and temperature are two critically important parameters for assessing tensile ductility in the presence of hydrogen. Another step in the absorption process presents the oxide film in austenitic steels. It can effectively reduce the solubility of hydrogen in the metal. An important implication of the reduced hydrogen uptake due to the protective oxide film is that film rupture associated with dynamic plastic strain will be expected to play a key role in hydrogen entry and should be considered a potential factor in designing a test program and in modelling of hydrogen assisted cracking [Gangloff]. With no direct OE, the hydrogen embrittlement of the piping is currently considered only as a probable degradation mechanism. Several components removed from the piping and subjected to destructive material analyses showed no signs of surface embrittlement or cracking. **References:** *Embrittlement of Power Plant Steels*, EPRI, Palo Alto CA: (2013). 3002001474. Gangloff, R., Somerday, B.: "Gaseous hydrogen embrittlement of materials in energy technologies," Vol. 1, Woodhead, 2012. Krpec, M., et al., "Destructive evaluation of the components of the hydrogen removal piping," Internal Report UJV Rez, DITI 2302/706, 2020.	**Priority:** R0 - Low

Table 3-3
Assessment (AS) Gaps

Technical gaps related to characterizing and assessing degradation, the potential for occurrence and predictions of the progression of the degradation.

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V-AS-01 – Reactor Internals Aging Management Program during LTO period

Issue:

The existing MRP reactor internals aging management program was based on the features of the PWR reactors (B&W, CE, and Westinghouse). Consideration of VVER specifics and their LTO programs results in a need to revisit and extend the MRP program applicability to VVER fleet considering the targeted 50-70 years operation life of these reactors.

Description:

MRP-227 provides aging management guidance for reactor internals components. This guideline is supported by a number of technical basis documents. Some of these documents are based specifically on PWR technology (e.g. MRP-175, MRP-189, MRP-190, MRP-191, MRP-320, MRP-135, MRP-257, etc.).

Consideration of VVER specifics and their LTO programs results in a need to revisit and extend the MRP program applicability (including MRP-227) to VVER fleet considering the targeted 50-70 years operation life of these reactors.

IASCC susceptibility in austenitic Ti stabilized stainless steels is proven at Greifswald and Loviisa baffle to former bolts. The threshold stress dependence curve for IASCC initiation of RVI materials (grade 08cH18N10T) as a function of neutron fluence recommended in the VERLIFE code is based on PWR data. Similar data on VVER materials are not available. An R&D project to determine the initiation curve for 08Ch18N10T and its welds is needed.

According to the VERLIFE code, the recommended curve for the formation of small volumes of irradiation embrittled material (denoted LEA for Local Embrittlement Area) predicts for RVI materials, that:

1. VVER 440: for the maximum RVI temperature (333 °C), there is no risk of LEA formation up to ~ 55 dpa; according to predictive modeling, the inner surface of the core shroud reaches that dose after 65 years of operation.
2. VVER 1000: for the maximum RVI temperature (434 °C), there is no risk of LEA formation up to ~ 38 dpa exposure; according to the predictive modeling, the inner surface of the core shroud reaches that dose after 20 years of operation.

Irradiation embrittlement of these stainless alloys needs additional research that considers these new findings related to void swelling.

Void swelling is expected to be significant at irradiation temperatures above 350 °C. TEM examination of a baffle bolt removed from the Greifswald VVER 440 with a fluence up to12 dpa and an estimated temperature below 400 °C identified initial signs of swelling; <0.1% was roughly estimated from the cavity volume fraction. It is considered that there is a possible risk of RVI dimensional changes due to void swelling leading to problems with coolant flow and/or control rod movement.

Temperatures at the core barrel and its weld due to gamma heating should be lower than in baffle bolts. Nevertheless, according to VERLIFE, void swelling is predicted at the core barrel leading to a limitation of the coolant flow rate between core barrel and core shroud due to dimensional changes. This remains to be verified in service and checked by dimensional measurements.

This issue represents an unexplored area for specific VVER RVI structural materials. Synergistic effects of irradiation creep, stress relaxation and void swelling are anticipated, as is also the case for PWR internals where this aspect is considered relatively well characterized for unstabilized stainless steels.

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Status:

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Priority:

R0 - High

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Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-01 – Reactor Internals Aging Management Program during LTO period (continued) Synergistic interactions between these irradiation-induced degradation mechanisms are considered to need a comprehensive research on VVER specific materials. **References:** *EPRI Materials Degradation Matrix, Revision 4.* (Notes v4-7a,b , v4-12a,b,d ; v4-13a,b; v4-14,a,b) EPRI, Palo Alto, CA: 2018. 3002013781. NP 102-17 Federal Rules and Regulations in the Field of Nuclear Energy Use “Basic Requirements for Justification of Strength of VVER-Type Reactor Internals, Rosatom, 2017. Margolin, B.Z., Murashova, A.I., & Neustroev, V.S. (2011). Effect of stress on radiation swelling of austenitic steels. Voprosy Materialovedeniya, 4(68), (2011) 124–139.	**Status:** Open **Priority:** R0 - High
V-AS-03 – SCC of Stainless Steels Exposed to Primary Water **Issue:** VVER units use almost exclusively titanium stabilized austenitic stainless steel (Type 321) in the primary circuit. Field experience with the stabilized steels in operating VVER reactors has been good, with a relatively small number of failures attributed to SCC. However, there are several factors, that may have impact on the future degradation rates: severe cold work, impurities and late life SCC. There is a need to better understand the factors that control the potential for SCC occurrences so that proactive mitigation or management actions can be implemented. **Description:** In general, VVER operating experience has been very good. No cracking of any primary circuit component exposed directly to primary water with nominal oxygen concentration and nominal concentration of impurities has been observed to date. There have been some instances of SCC in primary water, but they were limited to off-normal primary water chemistry composition. The plants closely cooperate with material laboratories to perform failure analysis on fractured parts in order to identify root causes. One case of SCC in water environment with elevated concentration of impurities was cracking in the connecting flange of the main circulation pump distribution wheel. The surface of the wheel is cooled by the autonomous cooling system. The cracking was attributed to crevice geometry of failed part with increased concentration of impurities, adversely affecting protecting oxide layer on the stainless steel surface [Postler 2012]. Another case of SCC of component exposed to primary water has been the cracking of VVER 440 SG primary collector threaded holes. The threaded holes are not in direct contact with the primary medium, but it was expected that the water have entered during assembly/disassembly. Failure analysis indicated that the primary cause of the cracking was the lubricant, which contained high concentrations of sulfur and molybdenum [Postler 2004]. Experimental program on SCC evaluation was performed on nonirradiated Type 321 stainless steel (RVI components from Greifswald VVER plant, Unit 1) for comparison with irradiated material. No sensitivity of nonirradiated material was found in VVER primary water in contrast to harvested irradiated material (<11,4 dpa) [Hojna 2014]. At VVER reactors, the first hydrostatic pressure test load (strength test) value at the primary coolant piping was conducted at so high a pressure level that the manufactured tubular products were stressed higher than their yield strength in the inner zones of the piping. This resulted in local plastic deformation. This high pressure and the subsequent	**Status:** Open **Priority:** R0 – Medium

Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-03 – SCC of Stainless Steels Exposed to Primary Water (continued)	Status: Open Priority: R0 – Medium
pressure release helped in the relaxation of the remaining residual stresses that had arisen during fabrication. Due to this phenomenon, which is called autofrettage, the subsequent pressure cycles (next pressure test load or operation load) produce stresses in the piping are remain in the elastic range (below the yield strengths). So, the autofrettage method is based on the reciprocal relationship of the plasticized inner zone and the elastically deformed outer zone. The components are placed under so much pressure that their interiors become plastically deformed. After relaxing, pressure stresses are created in these areas. This prevents the outer zone from resuming its original shape. Instead, it remains stretched. This prevents crack formation in the later load cycles and increases the service life of the components. Key questions remain concerning the effect of the chemical and material variables on the susceptibility of stabilized austenitic stainless steels to SCC crack initiation and the late life SCC. Operating experiences and laboratory studies show significant effect of cold work and impurities on SCC of stabilized stainless steels. Typically, chloride induced SCC was observed in the field and is known to take place from the laboratory tests of C-rings. In severely caustic environments, the stabilized steel is very susceptible to SCC. If such conditions exist, e.g. in the environment in crevices, there is a high risk of crack initiation and fast propagation. Better understanding of phenomena, including species and their concentrations in the primary water environment and possible countermeasures (electropolishing, peening) is needed. Late life SCC is currently not considered as a significant issue affecting operation of VVER primary components. No information about any testing program has been found. This phenomenon may become important with respect to ageing fleet and license renewal process. It is the opinion of IMT panel that this degradation mechanism should be monitored/studied for stabilized austenitic stainless steels. At the minimum it should be ensured that activities related to late life SCC are coordinated in both PWR and VVER research areas. **References:** *EPRI Materials Degradation Matrix, Revision 4.* (Note v 3-6a) EPRI, Palo Alto, CA: 2018. 3002013781. *Materials Reliability Program: Pressurized Water Reactor Issue Management Tables – Revision 4 (MRP-205). EPRI, Palo Alto, CA: 2020. 3002018255.* Postler, M. et al.: “Failure Analysis of Cracks in Main Circulation Pump Distribution Wheel,” UJV Report DITI 2302/142, 2012. Postler, M. et al.: “Analyses of Cracks in Threaded Holes of SG collector,” 6th International Seminar on SG, Podolsk, Russia, 2004. Hojná, A., Intergranular fracture and fracture toughness of irradiated austenitic stainless steels of reactor core internals, in: Proceedings of the 16th International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, 2013.	

Table 3-3 (continued)
Assessment (AS) Gaps

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V-AS-04 - Thermal & Irradiation Embrittlement Combined Effects on SS Welds

Issue:

There remains a need for additional data to fully characterize the effects of irradiation, thermal aging, and potential synergistic effects for irradiated austenitic stainless steel welds in VVER reactor internal components.

Description:

While stainless steel welds are exposed to less fluence than the baffle plate, due to the potential residual stresses from the welding processes they may be subject to higher tensile stresses than other internal components. Possible void swelling induced stresses due to non-uniform dimensional changes of reactor vessel internals must be taken into account as well. Unlike in PWR plants, no casting components were used in the internal equipment of VVERs. The seams have a similar microscopic structure to castings but contain less ferrite. Due to the potential for stress corrosion cracking (SCC) in aged and embrittled material, the combined effects of SCC and irradiation-assisted stress corrosion cracking (IASCC) in stainless steel welds remains a concern.

While in the core basket of VVER-440 power plants the rings are welded together with circumferential seams, in the VVER-1000 reactors the rings are held together with screws. Previous operational experience at VVER-440 plants has not identified any failures that have affected the seams of the equipment. Research results from the BOR-60 show that the weld behaves similarly to the base material in several ways when irradiated up to 40 dpa: yield strength and tensile strength stops increasing and plasticity stops decreasing once irradiation levels exceed 20 dpa; the resistance to crack propagation reaches a minimum J_C of 20 kJ m^{-2} both at the test temperature of 300 °C and at 350 °C when irradiated; and, the rate of cyclic cracking does not increase.

There is currently insufficient data from environmentally assisted cracking tests on crack initiation and growth rates on irradiated stainless steel welds in VVER. An R&D project to determine these for internal welds is therefore required.

References*:*

EPRI Materials Degradation Matrix, Revision 4. (Note Y-v4-7b EPRI, Palo Alto, CA: 2018. 3002013781.

Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data—State of Knowledge MRP-211 Rev 1. EPRI, Palo Alto, CA: 2017. 3002001027.

Materials Reliability Program: Thermal Aging and Neutron Embrittlement Assessment of Cast Austenitic Stainless Steels and Stainless Steel Welds in PWR Internals MRP-276. EPRI, Palo Alto, CA: 2010. 1020959.

Margolin, B.Z., et al: "Influence of Neutron Irradiation and Environment on Materials Properties of Internals of VVER Reactors." 11th International Conference on Materials Issues in Design Manufacturing and Operation of Nuclear Power Plant Equipment, TACIS project R2.01/02. CRISM Prometey, St. Petersburg, Russia. 2010.

Margolin, B.Z., Smirnov, V.I.,. Fedorova, V.I., Minkin, A.J.: "Influence of Neutron Irradiation on the Material Properties of Internals of VVER Reactors. Task 7 -Tensile and Fracture Toughness Testing of Irradiated Material (08Ch18H10T with its weld) Compared to Non-Irradiated PVI (reference material)" TACIS Project R2.01/02. 2010. Podolsk, Russia. (in Russian)

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Status:

Open

Priority:

R0 - Medium

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Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-04 - Thermal & Irradiation Embrittlement Combined Effects on SS Welds (continued) Margolin, B.Z., Fedorova V.A., Kokhonov, V.I.,. Chistyakov, D.A., Minkin, A.J.: "Influence of Neutron Irradiation on the Material Properties of Internals of VVER Reactors. Task 8: Stress Corrosion Cracking Testing of Irradiated Material Compared to Non-Irradiated Reference PVI" TACIS PROJECT R2.01/02. 2010 Podolsk, Russia. (in Russian)Margolin, B.Z., et al.: "Structural Integrity Of WWER Internals Issues And Approaches. The Tenth International Conference Material Issues In Design, Manufacturing And Operation Of NPP Equipment" CRISM Prometey, St. Petersburg, Russia. 2008.	**Status:** Open **Priority:** R0 - Medium
V-AS-05 – IASCC Characterization: Generic Data Needs **Issue:** A need exists to better characterize the properties of irradiated austenitic steels and the IASCC susceptibility of stainless steels used in the VVER environment. **Description:** Currently, most material science studies have been carried out on materials irradiated in a fast breeder reactor with sodium coolant. The neutron spectrum of these reactors differs from VVER-440 and VVER-1000, however and damage structures might be expected to be quantitatively difference for equivalent exposures in thermal and fast reactors. Materials harvested from a retired VVER-440 and a fast reactor were investigated up to 100 dpa. The result in both cases was an increase in yield strength up to 20 dpa, after which no further increase was observed. While the mechanical properties increased with the level of exposure and damage accumulation, ductility was found to decrease with increasing levels of damage. The reduction in area decreased to 20 % at 20 dpa, after which no further decrease was observed. In an investigation of 08Ch18N10T irradiated up to 145 dpa, which is the level of exposure experienced in VVER-1000, it was shown that the fracture resistance, $J_{C,}$ decreased with increasing neutron dose until 20 dpa, after which no further decrease was observed. The behavior of highly irradiated AISI 321 / 08Ch18N10T stainless steels in VVER water has been linked to the potential for irradiation-assisted stress corrosion cracking (IASCC). There is a concern that the IASCC susceptibility of irradiated materials from fast reactors used in R&D programs could be lower than what is observed in practice in irradiated materials from PWRs for the same nominal neutron dose. Assessment of IASCC risk should assess two factors: 1. conditions for IASCC crack initiation; and, 2. evolution of IASCC cracks. Conditions for IASCC crack initiation have not yet been clearly defined, however. It is a very difficult and complex task and only very limited data are available. It is unclear if the IASCC process is caused by the increase in the material yield strength and hardening due to irradiation alone or if the neutron irradiation somehow assists the crack initiation process directly.	**Status:** Open **Priority:** R0 – High

Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-05 – IASCC Characterization: Generic Data Needs (continued) As it is suspected that unirradiated austenitic steels become susceptible to intergranular stress corrosion cracking (IGSCC) during long term operation (low temperature sensitization), it is also suspected that irradiated austenitic steels become susceptible to IASCC. The corrosive behavior of samples that have served long periods at operating temperature should therefore be investigated. It is known that IGSCC occurs earlier in large grain sized batches than in small grain sized batches of sensitized steels, so this is likely to be the case for IASCC as well. As larger particle sizes occur at regions of higher thicknesses, analysis of the stresses incurred at these regions should be given priority. The case of the Finnish Loviisa plant clearly demonstrates the effect of IASCC. Here, a screw was welded to the baffle. The process inhibited expansion of the baffle and applied an unintentional load to the screw. This increase in load led to cracking. This increase in local stress, similar to a slow tensile test, increases continuously as the level of swelling in the baffle increases. This case confirms that IASCC can occur due to tension caused by the swelling of the baffle at doses as low as 3 dpa Determination of potential synergistic interactions between loading conditions, swelling and irradiation-induced degradation mechanisms on VVER specific materials requires further research. The temperature dependence of radiation swelling under VVER-440 model 213 conditions should also be determined. **References:** *EPRI Materials Degradation Matrix, Revision 4.* (Note p2-7a) EPRI, Palo Alto, CA: 2018. 3002013781.	**Status:** Open **Priority:** R0 – High
V-AS-06 – IASCC Characterization: Baffle Bolting **Issue:** There is a need to better characterize and quantify the IASCC behavior of baffle bolt materials and to develop predict models of the failure process of assemblies of Baffle bolting. [NOTE: This gap focuses on management of baffle-former bolting IASCC Generic IASCC characterization needs for Ti stabilized stainless steels are addressed in gap Y-v4-7a. Also gaps Y-v4-9a and 13a are relevant to this issue.] **Description:** In the case of PWR there have been a significant number of baffle bolt IASCC occurrences. Ultrasonic inspections of baffle to former bolts in France and the USA have identified indications of potential cracking. Removal of these and bolts without indications has discovered partially and fully crack bolts. In the early French experience mostly bolts found with indications were generally isolated from other bolts with indications; these bolts were generally found to be partially cracked. Concerning VVER440 OE, there was only one case in Finland where a cracked bolt was found. After the removal of this bolt the investigation showed that the welding of the washer to the shielding plate resulted which had resulted high stresses during plant start-up and operation in the bolt, and this was the root cause of the cracking. Although there have not been other cases so far, the IASCC susceptibility in austenitic Ti stabilized stainless steels has been proven (Greifswald and Loviisa baffle bolts). The threshold stress dependence curve for IASCC initiation of RVI materials as a function of neutron fluence recommended in the VERLIFE code is based on PWR data. Similar data on VVER materials are only limited available and mainly based on specimens irradiated in fast reactors.	**Status:** Open **Priority:** R0 - High

Table 3-3 (continued)
Assessment (AS) Gaps

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V-AS-06 – IASCC Characterization: Baffle Bolting (continued)

There are several additional factors that affects the failure process of the bolts, like low cycle fatigue that is caused by the thermal cycle, high cycle fatigue due to the fluid flow around the bolts. The radiation swelling of the basket leads to its deformation which causes increased stresses in the blots. This can accelerate the initiation and the propagation of IASCC cracking. There is a model for swelling in VERLIFE, and efforts have also been done to develop this model by Russian scientists, however the available input data are very limited, and there are several uncertainties in the model parameters, i.e. the dose and temperature distribution in the basket, the shielding plate and the bolt.

In addition, synergistic effects of irradiation creep, stress relaxation and swelling are anticipated, as is also the case for PWR internals where this aspect is considered relatively well characterized for non-stabilized stainless steels. These effects might also influence the IASCC process in the bolts. This topic represents an unexplored area for specific VVER RVI structural materials.

This issue is more important for VVER 440, than VVER 1000, because in VVER1000 the additional stresses in the bolt due to swelling is much less significant due to the different design. In some VVER 440 (Loviisa, Rovnó, Kola) there are dummies built-in, thus the doses are lower, thus the irradiation effect is smaller.

An R&D project to determine the initiation curve for 08Ch18N10T and its welds is needed. Synergistic interactions between the irradiation-induced degradation mechanisms are considered to need more research on VVER specific materials. Also, the effects of uncertainties of the input parameters for the numerical calculations should be investigated.

Key issues that need to be clarified include:

The relationships between stress and IASCC crack initiation times for bolting materials

There is a need to better develop mathematical relationships between imposed stress and crack initiation times for the materials used for baffle to former bolting at the dose for which the properties would be representative of those exhibited in long term operated plants.

Confirm the effects of dose on IASCC initiation for various bolting materials:

To reliably predict behavior at doses typical of bolts in extended life plants confirmation of this behavior for materials in the current fleet is needed.

Dynamic loading:

The effect of dynamic loading on the crack initiation in bolts must be determined in order to determine any potential accelerating effect on IASCC crack initiation, whether the cycling arises from transient plant operations or plate vibrations.

References:

EPRI Materials Degradation Matrix, Revision 4. (Notes Y-v4-7a, Y-v4-9a, Y-v4-13a) EPRI, Palo Alto, CA: 2018. 3002013781.

Materials Reliability Program: Analysis of IASCC Initiation Data for Irradiated Stainless Steels (MRP-224). EPRI, Palo Alto, CA: 2007. 1015480.

Materials Reliability Program: Characterization of US Pressurized Water Reactor Fleet Operational Transients (MRP-393). EPRI, Palo Alto, CA: 2014. 3002003085.

Materials Reliability Program: Hot Cell Testing of Baffle-to-Former Bolts Removed from US Pressurized Water Reactors (MRP-425). EPRI, Palo Alto, CA: 2017. 30020010528.

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Priority:

R0 – High

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Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-06 – IASCC Characterization: Baffle Bolting (continued) *Materials Reliability Program: Evaluation of the Effects of PWR Transients on Stresses and Crack Initiation in PWR Internals (MRP-434)*. EPRI, Palo Alto, CA: 2018. 3002013636. • IAEA-SSR-82, Rev. 1, "Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL)," International Atomic Energy Agency, Vienna, August 2020.AMP 115 Bolting Integrity Guidelines for Integrity and Lifetime Assessment of Components and Piping in WWER Nuclear Power Plant Nuclear Power Plant (VERLIFE) 2014. Ehrnstén, U., Pakarinena, J., Karlsen, W., Kei, H.,: " Investigations on core basket bolts from a VVER 440 power plant," in Engineering Failure Analysis, October 2013.Kalchenko, A.S., Bryk, V.V., Lazarevó, N.P., Neklyudov, I.M., Voyevodin, V.N., Garner, F.A,.: Prediction of swelling of 18Cr10NiTi austenitic steel over a wide range of displacement rates, Journal of Nuclear Materials, 399 (2010), 114 – 121. Margolin, B.Z., Sorokin, A.A., Buchatsky, A.A.: "Assessment of the Crack Growth Rate in the Material of the VVER-1000 Reactor Internals under Irradiation Creep," Radiation Material Science, Vol.7, No. 6, pp. 942-949, 2016.Margolin, B.Z., Sorokin, A.A., Smirnov, V., Potapova, V., Physical and mechanical modelling of neutron irradiation effect on ductile fracture. Part 1. Prediction of fracture strain and fracture toughness of austenitic steels, Journal of Nuclear Materials, 452 (2014), 595 – 606. Margolin, B. Z., Sorokin, A.A ., Smirnov, V., Potapova, V.: Physical and mechanical modeling of the neutron irradiation effect on ductile fracture. Part 2. Prediction of swelling effect on drastic decrease in strength. Journal of Nuclear Materials, 452 (2014), 607 – 613. Margolin B.Z. et al.:The radiation swelling effect on fracture properties and fracture mechanisms of irradiated austenitic steels. Part I: Ductility and fracture toughness. Journal of Nuclear Materials 480 (2016) 52 – 68.Margolin, B.Z. et al.: The radiation swelling effect on fracture properties and fracture mechanisms of irradiated austenitic steels. Part II: Fatigue crack growth rate. Journal of Nuclear Materials, 480 (2016) 15 – 24.	**Status:** Open **Priority:** R0 - High
V-AS-07 – Void Swelling of Stainless Steels **Issue:** There is a need to better understand the conditions contributing to void swelling and its effect on material properties of VVER RPV internals. Effort is needed to evaluate approaches of individual VVER utility operators to void swelling degradation mechanism and compare with approach in PWR area, analyze available calculating models, collect and critically assess the void swelling measurements on irradiated components from fast and thermal reactors and compile recommendations.	**Status:** Open **Priority:** R0 - High

**Table 3-3 (continued)
Assessment (AS) Gaps**

V-AS-07 – Void Swelling of Stainless Steels (continued)	Status: Open
Description: Void swelling is a concern for VVER internals as swelling of material would cause severe degradation of materials properties and also produce dilatational and distortional changes in reactor vessel internals components, that were not anticipated during design phase. Void swelling is modelled to be in units of percentage points after 40-60 years of operation and there are concerns that even such relatively small amounts could have significant impact on large structures that are not tolerant to dimensional changes. For some internal structures, small levels of differential swelling of one component could result in significant local stresses. This can change dimensions of the component as well as create additional stresses that were not accounted for during design calculations and can affect resistance to stress corrosion cracking. Also, the internals assembly is designed with relatively small tolerances and any positional changes in order of millimeters may have impact on their correct operation. Even relatively small amounts of swelling may have significant negative impact on lifetime of the internals. EPRI report 3002020892 contains the literature survey of void swelling of Type 321 steel. There are several models used for calculation of swelling for VVER plants (Russian central research institute Promethey, model Kalchenko and VERLIFE 2014). For some VVER operators the VERLIFE model or its local modification is required by the regulatory body. In general, calculations show some impact of void swelling on VVER 1000 internals, some results indicate values of up to 5% of void swelling in some locations after 60 years of operation. The potential consequence of this swelling is in closure of the gap between the core shroud and core basket used for cooling. Due to lower temperatures, gamma heating and different design of VVER 440 internals the impact of void swelling is expected to be more limited. Void swelling calculations strongly depend on the model as well as the input data, namely the temperature fields, gamma heating and the actual dose rate. These are usually not known directly from the measurements and must be calculated, depending on the fuel history and therefore may vary significantly unit to unit. All this adds to the uncertainty of the calculated values and possibly results in significant conservatism. There are limited data measured on material harvested from decommissioned reactors. Bolts from internals of Greifswald unit 2 after 15 years of operation were irradiated to 11.2 dpa. TEM measurements indicate the swelling is about 0.03 %, generally confirming lower impact of this degradation mechanism on VVER 440 units [Michalicka 2009]. Microstructural analysis of material harvested from a Russian VVER after 45 years of operation exhibited the maximum level of swelling to be 0.19 % [Kulosheva 2020]. Results from the fast reactors need to be analyzed thoroughly. One needs to take into account the fact, that the irradiation temperatures are usually very high with limited effect of gamma heating compared to operational reactors. Also, there is notable effect of dpa rate – the lower the displacement rate is, the higher amount of swelling is measured in material. For example, results from European TACIS project, with materials irradiated up to 40 dpa in fast reactor, indicate swelling rate of only 0.16 %. The reason for such a small value is probably relatively low irradiation temperature of 350 °C and high dpa rate. On the contrary material harvested from the fast reactor BN-350 vessel [Garner 2005] irradiated to 15.6 dpa at 340 °C, with relatively small dpa rate of 4.9×10^{-8} dpa/s, shows void swelling up to 1 %. Given the results and possible conservatism of calculations and the limited data from power generating reactor irradiated materials, some utility operators have already performed or are in the process of designing and constructing the rig for precise dimensional measurements of the internals. Measurement at a Ukrainian VVER plant after 30 years of	**Priority:** R0 - High

Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-07 – Void Swelling of Stainless Steels (continued)	Status: Open Priority: R0 - High
operation indicate that the dimensional changes are within manufacturing tolerances to ± 1 mm. However, one of the conclusions of the project was it would be appropriate to design a new measurement rig with improved precision and to install sensors for direct measurement of doses and temperatures inside the internals during unit operation. In the Czech Republic efforts are under way to construct a measurement rig for dimensional measurements of in-service reactor baffles. This rig is based on an optical-induction method and its precision should be ±0,1 mm. The first measurement is planned for 2022. Similar efforts are also ongoing in the Ukraine. The majority of VVER operators acknowledge void swelling as an issue that needs to be addressed; a number of different types of works are already being performed. Given the uncertainty and probable conservativism of available models, their benchmarking and assessment should be assessed and better validated. Void swelling measurements on the materials harvested from the decommissioned plants with higher doses are needed as well as critical assessment of all currently available data and models. The aim of the works should be to establish better correlation between calculation results and valid measurement data and provide VVER operators clear guidance on void swelling degradation mode and its effect on LTO. Calculations show that due to void swelling of the core baffle, sooner or later contact between core baffle and core barrel is possible. The calculations show that this contact is more probable than the contact between core baffle and fuel assembly. So, it is important to evaluate the possible influence of the contact of core baffle and core barrel on possible infractions of the design basis (disruption of the flow of coolant and of the heat balance, overflow of the coolant). This assessment is needed in order to determine whether continued operation of the plant would be possible in the presence of contact between core baffle and core barrel. **References:** *EPRI Materials Degradation Matrix, Revision 4*. (Notes v4-13a, v4-13b) EPRI, Palo Alto, CA: 2018. 3002013781. *EPRI Materials Reliability Program: Zorita Internals Research Project (MRP-440), Testing of Highly-Irradiated Baffle Plate Material*. EPRI, Palo Alto, CA: (2019). 3002016015. *Irradiation-Assisted Stress Corrosion Cracking (IASCC) of Ti-Stabilized Austenitic Stainless Steel IASCC in the Water-Water Power Reactor (VVER) Environment*. EPRI, Palo Alto, CA: 2018. 3002012411. Michalicka, J., et al.: Radiation damage of WWER 440 internals after long term operation, UJV report DITI 302/572, 2009. Garner, F.A., et al, Void swelling of austenitic steels irradiated with neutrons at low temperatures and very low dpa rates, in: Proceedings of the12th International Conference on Environmental Degradation of Materials in Nuclear Power Systems- Water Reactors, 2005. Garner, F.A., Kozlov, A. V., and Okita, T., The Competing Influences of Void Swelling and Radiation-Induced Precipitation on Dimensional Stability and Thermal-Physical Properties of Austenitic Stainless Steels in PWR and VVER internals, in: Proceedings of the 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, 2015. Kuleshova, E.A., et al., Microstructure degradation of austenitic stainless steels after 45 years of operation as WWER-440 reactor internals, Journal of Nuclear Materials, 533 (2020).	

Table 3-3 (continued)
Assessment (AS) Gaps

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V-AS-08 -Steam Generator Sludge Deposits and Scale Buildup

Issue:

Magnetite deposits on the secondary side of the horizontal SG tubing can cause clogging of the intertubing spaces and damage, in the form of SCC and corrosion product cracking to the tubes themselves. A guidance on assessment of these deposits is missing.

Discussion:

During operation, dissolved iron in feedwater originating from a corrosion of balance of plant is deposited on the VVER steam generator tubes. These deposits then crystallize into magnetite. Consequently, SCC defects develop at the sites of these deposits as a local chemistry in the crevices between deposits and heat exchange tube is more aggressive due to hide-out of soluble impurities.

There are two aspects related to this assessment gap for VVER units. The first one is related to determination of an amount of deposits on the heat exchange tube surface. The tools available include scraping of magnetite from a surface of topmost tubes. The method has questionable reproducibility and the tubes at the top of tube bundle are not heavily loaded with magnetite. A low frequency eddy current testing is still in development and VT inspections do not allow quantitative evaluation of magnetite deposits layer.

The second aspect is related to the lack of a basis for the limiting allowable value of deposit layer thickness. To date 150 g/m2 and 100 g/m2 are used for VVER440 and VVER1000 respectively, but no clear relation between amount of deposits and SCC intensity is available. The limit for triggering of mitigation measure like water lancing or chemical cleaning is missing as well.

References:

IAEA-SSR-82, Rev. 1, “Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL),” International Atomic Energy Agency, Vienna, August 2020.

- AMP116, Ageing Management for Nuclear Power Plants

Yurmanov, E.V., Yurmanov, V.A., Velikopolsky, S.V., “Behavior of Corrosion Products in Secondary Systems of WWER Plants,” 8th International Seminar on Horizontal Steam Generators. EDO. OKB, GIDROPRESS, Podolsk. 2010. (in Russian)

Vepsäläinen V.: Deposit formation in PWR steam generators. SAFIR2010, Finnish national research program on NPP safety. 2007-2010.

ÚJV-DITI 2302-615 EN: Database of water chemistry and SG tubes damage, Ver. 21.0

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Status:

Open

Priority:

R0 - Medium

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Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-09 Fluence Impact on Stainless Steel Mechanical Properties (Fracture Toughness, Tensile Strength)	Status:
Issue: A need exists to better understand the effects of high neutron fluence on the mechanical properties of stainless steels, including associated weld and HAZ. High fluence data are very limited for the titanium stabilized austenitic steels. **Description:** Mechanical property data after high fluence irradiation (above 5 dpa) of 316 and 304 steels are available, since these steels are studied for the western PWRs and fusion devices. The titanium stabilized austenitic steels (08Ch18N10T, similar to AISI 321) are widely used in the VVER reactors. According to the literature the tensile strength (and especially the yield strength) increases with the increased fluence, the ductility and the fracture toughness is decreases. At the level of 10-20 dpa saturation is observed, here the yield strength nearly reaches the ultimate tensile strength. The effect of further neutron irradiation of the Ti stabilized austenitic steels is not well studied. Most of the high fluence property are measured in air. The existing knowledge on the effect of 70-100 dpa neutron irradiation in high temperature water environment is very limited. Some of the reactor internal structures are degraded by fatigue or low-cycle fatigue in pressurized water environment. Temperature change causes extra fatigue stresses at high fluence. The swelling over 5 dpa irradiation also add further load after long term operation. Typical examples are the baffle bolts, similarly to the other PWRs. In the VVER units the occasionally used boron-acid in the coolant may increases the stress corrosion fatigue rate. Further study of on high fluence irradiation effect on the mechanical properties and stress corrosion resistance of the titanium stabilized austenitic stainless steels is recommended. **References:** *Materials Reliability Program: Fluence Effects on Stainless Steel Welds (MRP-451): Crack Growth Rate and Fracture Toughness Testing of Zorita Weld and HAZ Materials*. EPRI, Palo Alto, CA: 2020. 3002018250. IAEA-TECDOC-1557, "Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: PWR Vessel Internals," International Atomic Energy Agency, Vienna, June 2007.IAEA-TECDOC-1545, "Characterization and Testing of Materials for Nuclear Reactors," International Atomic Energy Agency, Vienna, March 2007.Ballesteros, A., Heid, A., and Luostarinen,.P.: Inspection and Replacement of Baffle Former Bolts in VVER-440 Reactor Type, 19th International Conference on Structural Mechanics in Reactor Technology (SMiRT 19). Toronto, Canada, 2007. Regidor, J.J., Ballesteros, A., Heid, A., and Luostarinen,. P.: Inspection and Replacement of Baffle Former Bolts in VVER-440 Reactor Type, 6th International Conference on NDE in Relation to Structural Integrity for Nuclear and Pressurized Components. Hungary. 2007.Pakarinen, J., et al., Microstructural Characterization of Irradiated Baffle Bolts Removed from a Finnish VVER and a French PWR, in: Proceedings of the 16th International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, 2013.Hojná, A., Košťál M.: Exploring Open Material Issues Before Extended Operation of Reactor Vessel Internals. Jaderné Energie 24(62), 2016. Ehrnstén U., Kytömäki P., Hietanen O. Investigations on Core Basket Bolts from a VVER 440 Power Plant, in: Proceedings of the Proceedings of the 15th International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors,2011. Grytsenko, O.V., Pugach, S.M., Diemokhin, V.L., Bukanov, V.N., Marek, M., and Vandlik, S.: "Exposure Conditions of Reactor Internals of Rovno VVER-440 Nuclear Power Plant Units 1 and 2," Journal of ASTM International, Vol. 9, No. 4 (2012).Namburi, H.K., Hojna, A., Hernandez Mayoral, M., Duchon, J., Halodova, P., Bublikova P.: Flux Effect on the Radiation Damage of Austenitic Steels. SOTERIA Midterm Workshop, Prague, 2018.	Open **Priority:** R0 - High

Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-10 - Low Temperature Crack Propagation (LTCP) Assessment	Status:
Issue: Low Temperature Crack Propagation is a form of the hydrogen embrittlement that can cause severe degradation of the fracture resistance of nickel-based alloys and cast austenitic stainless steels under specific operating conditions (low temperatures and high hydrogen concentrations in primary water). There is a need to assess the relevance of this experimental data to VVER operation. **Description:** The degrading effect of hydrogen on LTCP resistance of nickel-based Alloy 182, 82, 152 and 52 weld metals in low temperature water was clearly demonstrated. Alloy 182 is the most susceptible material to LTCP and exhibits a significant reduction of fracture resistance already with hydrogen content of 30 cm^3 H_2/kg H_2O. [Ahonen 2015] Testing of limited number of thermally aged CF-8 steel samples at 54 °C in PWR environment with 30 cm^3 H_2/kg H_2O has shown a dramatic drop in fracture toughness after fatigue pre-cracking. These conditions could be pertinent to start-up and shut-down operations and for lay-ups of plants or individual plant component. K_Q values measured after fatigue pre-cracking specimens are low but still not in the brittle regime for a structural material. [1020957] At VVER plants no hydrogen peroxide is generally added during operating cycle of the unit and the hydrogen concentrations remain between 20 – 40 cm^3 H_2/kg H_2O during operation and shutdown conditions. Use of CASS components in VVER plants is very limited; typically, only the body of main circulation pump is made from CASS. Therefore, the possible detrimental effect on the operation of the unit would be relatively limited. To date there is no OE from VVER plants indicating this degradation mechanism would take effect and as such it is currently not considered as relevant in the existing ageing management programs. However, it is not unequivocally determined whether this phenomenon could affect VVER materials in their operating conditions. No references of testing of LTCP performed on VVER materials have been found in the open literature. Additional work is needed to answer question whether LTCP could affect Type 321 titanium stabilized cast austenitic stainless steel used in VVER units, what are the parameters that affect the LTCP processes, namely combination of temperature and hydrogen concentration and whether there is any correlation with startup or shutdown conditions. **References:** *EPRI Materials Degradation Matrix, Revision 4.* (Notes p1-11d, 2-11h) EPRI, Palo Alto, CA: 2018. 3002013781. *Program on Technology Innovation: Scoping Study on Low Temperature Crack Propagation for 182 Weld Metal in BWR Environments and for Cast Austenitic Stainless in PWR Environments – Revision 1*. EPRI, Palo Alto, CA: (2010). 1020957. Demma A., et.al, Low temperature crack propagation evaluation in pressurized water reactor service, in: Proceedings of the 12th International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, 2005. Sakima, K., et el., Research and evaluation of low temperature crack propagation of Ni base alloys in actual plants, in: Proceedings of 15th International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, 2011. Ahonen, M., et.al., Factors affecting low temperature crack propagation (ltcp) susceptibility of nickel-based alloy 182, 82, 152 and 52 weld metals, in: Proceedings of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, 2015.	Open **Priority:** R0 - Low

Table 3-3 (continued)
Assessment (AS) Gaps

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V-AS-14 – Neutron Embrittlement of Reactor Pressure Vessel Steels

Issue:

Neutron embrittlement of RPV steels is a very closely monitored degradation mechanism by both operators and regulatory body. To date there appears to be no need for further research on this issue. The gap describes the process as it is set up in the Czech Republic [National Report 2017].

Description:

Two different types of steels are used for manufacturing of Reactor Pressure Vessels depending on the reactor power:

- Cr-Mo-V steel of 15Kh2MFA-A grade for RPVs of VVER-440 and their modifications for irradiation temperature 270 °C,
- Ni-Cr-Mo-V steel of 15Kh2NMFA-A grade for RPVs of VVER-1000 and their modifications with higher electrical outputs for irradiation temperature 290 °C.

Due to different chemical composition and heat treatment, irradiation temperatures and neutron fluences and especially their different designs of surveillance specimen programs, two different procedures are used for monitoring the effect of neutron embrittlement of RPV steels.

VVER-440

The highest anticipated fluences at the end of expected lifetime of VVER-440 reactors (currently 60 years) are about ~$3.5x10^{20}$ n/cm^2 (with energies larger than 0.5 MeV that is approximately equal to $2.5x10^{20}$ n/cm^2 with neutron energies larger than 1 MeV). The original Standard Surveillance Specimen Program developed for evaluation of irradiation degradation of the RPV materials by designer was in the Czech Republic and in Slovakia deeply modified into Supplementary Surveillance Specimen Program (DSP) and further into Extended Surveillance Specimen Program (PSP) to fulfill requirements of state of art and extended operational lifetime. The main purposes of the programs are:

- Monitoring of neutron flux at the RPV wall during the whole operation,
- Monitoring of radiation embrittlement of RPV base / weld / heat affected zone materials
- Monitoring of radiation embrittlement of austenitic cladding
- Monitoring of thermal ageing of the RPV materials

The results from the program are evaluated periodically once a year and time schedule of the program (removing containers and evaluation of surveillance specimens, temperature and fluence monitors) is updated according to results and needs of the operation. The PSP program is designed in such a way to monitor fluence as well as material behavior through the whole extended reactor lifetime.

According to existing results from the DSP and PSP programs the material of the RPV does not exhibit any unexpected embrittlement shifts up to the irradiation level $5x10^{20}$ n/cm^2. The changes of the material properties (yield strength, tensile strength, shift in critical temperature of brittleness from notch toughness tests and reference temperature from fracture toughness tests) are lower than expected according to normative Embrittlement Trend Curves for both notch toughness as well as fracture toughness transition temperatures.

Thermal ageing effects have been found as negligible even after about 20 years of operation at temperature 300 °C (outlet water temperature). Monitoring is still continuing by additional thermal ageing containers.

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Status:

Open

Priority:

R0 - Low

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Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-14 – Neutron Embrittlement of Reactor Pressure Vessel Steels (continued)	Status: Open Priority: R0 - Low
VVER-1000 This type of steel in VVER-1000 reactors has relatively high content of nickel in most of welds – between 1.5 and 1.88 mass % (welds of the Temelin RPVs contain only between 1.55 And 1.60 mass %) – while base metals are manufactured with nickel content between 1.2 and 1.5 mass % (Note that newer RPVs are being welded with content of nickel lower than 1.5 mass %). This high content of nickel in welds could be a source of potential “late blooming effects” but the expected neutron fluence for 60 years of operation is lower than $6x10^{19}$ n/cm^2 (with neutron energies larger than 0.5 MeV that is equivalent approximately to $4x10^{19}$ n/cm^2 with neutron energies larger than 1 MeV). This level is far below the expected threshold for the potential development of the “late blooming precipitates”. The main purposes of the program in Temelin NPPs are: • Monitoring of neutron flux at the RPV wall during the whole operation, • Monitoring of radiation embrittlement of RPV base / weld / heat affected zone materials • Monitoring of radiation embrittlement of austenitic cladding • Monitoring of thermal ageing of the RPV materials Evaluation and re-evaluation of Standard surveillance programs in the countries operating VVER-1000 several years ago lead to construction of normative Embrittlement Trend Curves for these materials up to neutron fluence of about $6x10^{19}$ n/cm^2 (with neutron energies larger than 0.5 MeV). The modified surveillance specimen program for NPP Temelin is designed to monitor neutron fluences and changes in material properties (yield strength, tensile strength, shift in critical temperature of brittleness from notch toughness tests and reference temperature from fracture toughness tests) with a relatively low lead factor (about 1.8), thus there are not yet specific experimental data for end-of-life fluences., However, the program is monitoring material changes after incremental exposures to provide a basis for predictions of whether unacceptable embrittlement may be produced within the vessel lifetime. . Current results well follow the normative Embrittlement Trend Curves with a good measure of conservatism as is necessary for the predictions to assure safe reactor operation. Some irradiation experiments have been performed with high lead factor to obtain necessary information about the potential development of the “late blooming effect” – no such effect has been found even for much larger fluences than expected for the LTO (i.e. up to $1x10^{20}$ n/cm^2 (with neutron energies larger than 0.5 MeV). All Surveillance programs in VVER-1000 RPVs contain also two to six replaceable thermal ageing containers. A large database of results has been collected with data even beyond exposures of about 150,000 hours at 320 °C (outlet water temperature) – under these conditions the transition temperature shift has been found to trend to an asymptotic value of about 30 °C. Currently, even though not all experimental data for the LTO of VVER-1000 reactors are available for all RPVs, the existing Surveillance program assures that they will be available in time. All existing data currently indicate that no unexpected degradation will arrive during the next, even extended operation of the reactors. In case of new reactors with newly designed materials used for Reactor Pressure Vessel steel it is expected that similar procedure will be applied. i.e. all materials used for a particular RPV vessel and its welds will be included in the surveillance program, that would be used as a tool for evaluation of material embrittlement. **References:** *EPRI Materials Degradation Matrix, Revision 4.* (Notes v1-12abc, v2-12abcd), EPRI, Palo Alto, CA: 2018. 3002013781. “Topical Peer-Review of Ageing Management,” The Czech Republic National Report, State Office for Nuclear Safety, Prague, 2017.	

Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-17 – Fracture Toughness Properties of Low Alloy Vessel Steels (Plates and Forgings)	**Status:** Open
Issue: Heterogeneity is not widely tested in VVER forgings. Macro heterogeneity is known from research projects studying the mechanical properties including fracture toughness of the same plate or forging taken the samples from different places. Carbon macro-segregation may occur. There is also a need to characterize and account for uncertainty in toughness properties (including heterogeneity) when performing direct fracture toughness test on a limited sample of specimens. **Description:** The pressure vessels of the VVER reactors are made from low alloyed CrMoV (VVER-440) or CrMoNiV (VVER-1000) forgings. Plate is not used for pressure vessels consequently no axial welding. The chemical composition of the forgings is generally satisfying the related standard requirements but inhomogeneity within the standard allowed range is typical. Testing of the materials cut from the retired Greifswald units shown considerable differences in the mechanical properties of the surface and middle section layers. The forgings quality and properties highly depend on the production date and on the manufacturer. Local inhomogeneity is also typical. The VVER-440 model V-230 are shutdown in Europe due to the high radiation embrittlement caused by high copper and phosphorus content. The reactors of V-213 model were built with limited copper and phosphorus content. The 15Ch2MFA steel is radiation tough, comparable or better than the A508 or A533. This is important since the VVER-440 pressure vessels are compact, the inner diameter is only 3840 mm and the wall thickness is only 140-190 mm. Due to the limited water gap the vessel wall got very high fluence (about $1.9 \cdot 10^{24}$n/cm^2 E>1 MeV during 30 years of lifetime and $3.4 \cdot 10^{24}$n/cm^2 E>1 MeV during 50 years of lifetime). The nearest circumferential weld is out of the dose maximum therefore the flux is about 60% of the maximum wall flux. Considering this high EOL fluence all European units had/have surveillance and extended surveillance program, most units use low leakage core configuration. Supporting research programs including fracture toughness specimens, and relative high amount of data available on as received and aged conditions. Interesting results obtained on the retired Greifswald units. The VVER-1000 pressure vessels are slightly bigger and thicker, and made of 15Ch2NMFA steel. This steel contains 1-2 % nickel. The Ni and Mn content reduces the as received transition temperature, but it increases the radiation and thermal ageing shift. The increased vessel diameter reduces the EOL fluence at the wall and at the nearest circumferential weld (about $3.4 \cdot 10^{23}$ n/cm^2 E>1 MeV during 40 years and $5.1 \cdot 10^{23}$ n/cm^2 E>1 MeV during 60 years of lifetime); it is in the range of western PWRs. Since the small number of VVER-1000 operated in the EU the available information on fracture toughness is less than in the case of VVER-440. European research projects (LONGLIFE, AGE-60) studied the ageing mechanisms of the VVER steels and welds, collected and analyzed the existing surveillance and research data. Also, efforts made to modelling and calculating the neutron radiation embrittlement (European research projects like PERFECT, PERFORM60, SOTERIA). The special guideline developed for the VVERs (VERLIFE) allows the direct use of fracture toughness data at lifetime (PTS, PT-curve) calculations. Master Curve and Charpy DBTT equally used at safety and lifetime analyses. The Russian Unified and Basic corves are not used in the Central European area. A new European research project STRUMAT (STRuctural MATerials research for safe Long Term Operation of LWR NPPs) started in late 2020 will focus on synergy effect of Ni, Mn and Si on mechanical properties and on validation of embrittlement trend equations. Further study on fracture toughness distribution in the low alloyed forgings to study heterogeneity is recommended.	**Priority:** R0 - Low

Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-17 – Fracture Toughness Properties of Low Alloy Vessel Steels (Plates and Forgings) (continued) **References:** IAEA-TECDOC-1442, "Guidelines for Prediction of Irradiation Embrittlement of Operating WWER-440 Reactor Pressure Vessels," International Atomic Energy Agency, Vienna, June 2005. IAEA-TRS-448, "Plant Life Management for Long Term Operation of Light Water Reactors, Principles and Guidelines," International Atomic Energy Agency, Vienna, 2006. Viehrig H.W., Houska M., Alstadt, E.: Mechanical Properties of RPV Beltline Welds in Dependence of the Thickness Location. Presentation of LONGLIFE final international workshop. Dresden, 2014. Viehrig, H.W., Alstadt, E., Houska, M., Mueller, G., Ulbright, A., Conheisser, J., Valo, M.: Investigation of Decommissioned Reactor Pressure Vessels of the Nuclear Power Plant Greifswald. HDZR, 2018. Lydman, J.et al.: Identification of Initial Microstructural Heterogeneities Promoting Brittle Fracture In Selected Reactor Pressure Vessel Base Materials And Weld Metals, presented at the SOTERIA Midterm Workshop, 2018. Viehrig, H.W.: Effects of initial material inhomogeneity on microstructure and mechanical properties at irradiated state, presented at the SOTERIA Midterm Workshop Prague, 2018. Timofeev, B., Brumovsky M., Von Estorff, U.: Certification Report of 15Kh2MFA/15Cr2MoVA Steel and its Welds for WWER Reactor Pressure Vessels. JRC, 2010. Acosta, B. et al.: WWER-1000 Base Metal Reference Steel And Its Characterization Nuclear Engineering and Design, Volume 235, Issues 17–19, pp. 1951-1957, 2005. Debarberis, L. et al.: Analysis of WWER-440 and PWR RPV Welds Surveillance Data to Compare Irradiation Damage Evolution. Journal of Nuclear Materials, 350 (2006) 173-181.	**Status:** Open **Priority:** R0 - Low
V-AS-18 - Environmental Effects on Fatigue Resistance: Pressure Boundary Components **Issue:** There is a need to incorporating the characteristics of Russian materials 08Ch18N10T and ST20 into the relevant MRP deliverables (or parallel products for VVERs) to take into account the effects of environment on the fatigue of pressure boundary components of VVERs for life assurance within the targeted LTO period of 60-70 years operations. **Description:** Approaches to Environmentally Assisted Fatigue (EAF) varies significantly between VVER operators. Some use the EPRI approach e.g. NUREG/CR-6909; ASME BPVC (exception: code fatigue curve), some use Russian normative documents (PNAE G-7-002-86; RD EO 1.1.2.05.0330) and others are developing own methodology combining different standards and guides (ASME B&PVC, Czech STD A.M.E. 4201-86 ÷ 4214-86, RCC-M, KTA, JSME S NF1-2009, VERLIFE).	**Status:** Open **Priority:** R0 - High

Table 3-3 (continued)
Assessment (AS) Gaps

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V-AS-18 - Environmental Effects on Fatigue Resistance: Pressure Boundary Components (continued)

Mechanisms contributing to environmental effects in stainless steel materials have not been fully clarified, therefore in a few VVER countries (Czech Republic, Finland) their own R&D activities are in progress for the Russian material grades of 08Ch18N10T, ST20. However, a better understanding to the EAF influencing parameters is needed to harmonize within international utilities. In this regard the development of an appropriate database for VVER specific materials is considered to be important.

Also, many experts consider that the available methods for accounting for environmental effects on low cycle fatigue evaluation are overly conservative. To introduce a more realistic approach to low cycle fatigue and EAF, further research work is needed. In this work the basis of NUREG/CR-6909 should be reviewed for taking into account the characteristics of Russian materials.

References:

EPRI Materials Degradation Matrix, Revision 4. (Notes b 1-9a, v1-9c,d; v2-9d,e; v6-9,a,b,c,d,e,) EPRI, Palo Alto, CA: 2018. 3002013781.

Environmentally Assisted Fatigue (EAF) Knowledge Gap Analysis: Update and Revision of EAF Knowledge Gaps. EPRI, Palo Alto, CA: 2018. 3002013214.

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Guidelines for Integrity and Lifetime Assessment of Components and Piping in WWER Nuclear Power Plant (VERLIFE), 2014.

HAEA Guideline 3.25, “Review of the strength of operating pressure retaining equipment,” Hungarian Atomic Energy Authority, 2015.

Rosatom RD EO 1.1.2.05.0330, “Guidance on the strength analysis of equipment and pipelines of RBMK, VVER and EGP reactor plants at the operational stage, including operation beyond the design life,” Rosatom State Nuclear Energy Corporation, 2012.

STUK YVL E.4, “Strength analyses of nuclear power plant pressure equipment,” Finnish Radiation and Nuclear Safety Authority, 2020.

STUK YVL 3.5 “Ensuring the strength of pressure equipment,” Finnish Radiation and Nuclear Safety Authority, 2002.

- NP 306.2.099-2004: “General requirements to lifetime extension of NPP power units for operation beyond the design period based on the results of periodic safety reviews,” State Committee for Nuclear Regulation of Ukraine. Kyiv, 2004. (in Ukrainian)PL-D.0.03.126-10: “Regulation on the procedure for extending the life of equipment of systems important to

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R0 - High

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Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-18 - Environmental Effects on Fatigue Resistance: Pressure Boundary Components (continued) safety,"State Committee for Nuclear Regulation of Ukraine, Kyiv, 2010. (in Russian)SOU NAEK 080:2014 Operation of Technological System. Long-Term Operation of NPP Units. General Provisions /6/PM-D.0.03.222-14: "Standard program on ageing management of NPP structures and components," NNEGC Energoatom, Kyiv, 2014. (in Russian)NP 306.2.210-2017: "General Requirements for Ageing Management of Components and Structures and Long-Term Operation of NPP Units," State Committee for Nuclear Regulation of Ukraine. Kyiv, 2017. IAEA-SSR-82, Rev. 1, "Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL)," International Atomic Energy Agency, Vienna, August 2020.AMP 101 Low Cycle Fatigue Monitoring • TLAA 106 Environmentally Assisted Fatigue	
V-AS-19 - Environmental Effects on Fatigue Resistance: Reactor Internals **Issue:** Current calculations of the environmental effect based on the conservative material data show sufficient fatigue resistance of reactor internals for long term operation. As the environmental effect was assessed on unirradiated materials, the question remains of the effect of irradiation on fatigue properties. **Description**: Whereas there are some data that address the effects of VVER water environments on fatigue initiation and crack growth for pressure boundary and internals materials, there are presently very limited data that characterize the effects of both high neutron fluence and VVER water environments on the fatigue behavior of reactor vessel internals' components. For the fatigue lifetime evaluation of RVI VVER operators often use national approach using national standards, which were usually adopted from the VERLIFE approach (Unified procedure for lifetime assessment of components and piping in VVER NPPs during operation) developed within the 5. Framework Programme of the European Union in 2003 and later upgraded within the 6. Framework Programme 'COVERS - Safety of VVER NPPs' of the European Union in 2008. For RVI fatigue evaluation the most conservative values of material mechanical properties are used. Part of fatigue evaluation is crack growth rate analysis of postulated crack(s). Necessary experimental constants / formulas were supplied by Russian participants of the VERLIFE project as well as the base data, which are often not available in the open literature sources. The methodology is applicable up to level of 70 dpa. The fatigue load has been evaluated with respect to the currently applied principles, i.e. the base-load operation. In case of change to the load-following mode, the new analysis would have to be made. Currently, the fatigue life of reactor internals is not considered as an issue, since the calculations even with conservative material values show that the fatigue factors are sufficient for long term operation timeline. **References:** *EPRI Materials Degradation Matrix, Revision 4.* (Notes v3-6a) EPRI, Palo Alto, CA: 2018. 3002013781. Guidelines for Integrity and Lifetime Assessment of Components and Piping in WWER Nuclear Power Plant (VERLIFE), 2014.	**Status:** Open **Priority**: R0 - High

Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-20 – Flow Induced Vibration and Wear of Reactor Internals	**Status:** Open
Issue: There is a need to assess the potential significance of flow induced vibration processes and resulting wear of reactor internals components. **Description** Operational vibration monitoring of internals carried out in the last decade on the VVER 440 power units, as well as work on extending their service life, has significantly improved our understanding of the vibrations experienced and their effects on the internals. Measures to reduce the vibration load of the internals and fuel assemblies of unit 2 of the Kola nuclear power plant, e.g. by securing the shaft in the zone of the flow divider and reducing heat carrier consumption through the reactor, significantly reduced the swing of the shaft and fuel assemblies. Direct geometrical measurements of the internals attachment points revealed a relationship between the state of the internals attachment points and out-of-core neutron noise. The control rod drive mechanism' supper guide structure of VVER is simpler in design than PWR, for example, the guide tube assembly does not use guide plates. The VVER reactors also do not have the thermal shield flexures that are found in PWRs. VVER-440s do have baffle bolts however, which, due to irradiation enhanced void swelling, are susceptible to IASCC. The vibration caused by the flow can also cause stress on the baffle, which can further contribute to this cracking process. Recent VVER-440 examinations have not revealed cracks in the cross-section of the control rod guide tubes or lower core barrel. For VVER-1000, there is currently a lack of data to analyze possible degradation due to vibration. According to fatigue crack propagation calculations of the basket and lower core plate, the large cracks that lead to either stable or instable crack propagation are not caused by vibrations. The risk of baffle bolt failure in VVER-440 increases as neutron fluence increases. If failures occur, unconstrained sections of baffle plates potentially become more susceptible to flow induced excitation and vibration under high frequency conditions. These vibrations may add an additional element of fatigue damage to the primary processes of failure in baffle bolts. We therefore recommend R&D projects to analyze the potential for the failure of baffle bolts due to irradiation enhanced stress. **References:** *EPRI Materials Degradation Matrix, Revision 4.* (Notes Y-v4-5a, Y-v4-5c) EPRI, Palo Alto, CA: 2018. 3002013781. *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175, Rev 1).* EPRI, Palo Alto, CA: 2017. 30020010268. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A).* EPRI, Palo Alto, CA: 2011. 1022863. *Materials Reliability Program: Hot Cell Testing of Baffle-to-Former Bolts Removed from US Pressurized Water Reactors (MRP-425).* EPRI, Palo Alto, CA: 2017. 30020010528. Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operations Projections. WCAP-17451-P; Westinghouse Electric Company (2015).	**Priority:** R0 - Low

Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-20 – Flow Induced Vibration and Wear of Reactor Internals (continued) Ovcsrov, O.V. et al.: Development of Vibration Standards Inner Cases of VVER-440 Reactors. Proceedings of the 10th Conference -- Safety Assurance of NPP with VVER. OKB GIDROPRESS, 2017. (in Russian) Pecinka, L., Svrcek, M., Zeman, V.: Fretting wear of the WWER core barrel lower fixation against RPV. Proceedings of the 9th Conference -- Safety Assurance of NPP with VVER. OKB GIDROPRESS, 2015. (in Russian) Szávai, Sz., Dudra, J.,: Lifetime analysis of WWER Reactor Pressure Vessel Internals concerning material degradation. 20th International Conference on Structural Mechanics in Reactor Technology (SMiRT 20). Espoo, Finland, 2009. Usanov, A.I.,: *Vibration Studies VVER Inner Reactor Equipment at Different Stages of the Life Cycle in the Task of Managing the Service Life of a NPP*. Dissertation for a scientific degree. Obninsk-2009. (in Russian) IAEA-SSR-82, Rev. 1, "Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL)," International Atomic Energy Agency, Vienna, August 2020.	**Status:** Open **Priority:** R0 - Low
V-AS-26 - Lack of sharing information of latest updated and developed probabilistic fracture mechanics codes through benchmarking **Issue:** Each VVER operator works under different regulatory environment. This translates also to requirements of respective regulatory bodies concerning the use of probabilistic fracture mechanics (PFM) codes. In some countries they are used only as a supplementary information (Czech Republic) whereas in other countries the regulatory body requires calculations by PFM to be made additionally to deterministic FM (Ukraine). No formal work on benchmarking of individual codes used by VVER operators has been performed to date. **Description:** In the Czech Republic, the probabilistic approach is considered only as a supplementary information. The deterministic approach is used as a primary tool for RPV integrity assessment. The normative basis for both deterministic and probabilistic PRV integrity assessment is [NTD ASI]. A code PROVER is used for PFM calculations. It is based on the older version of the US code FAVOR, which has been officially provided by NRC. The code was modified to reflect VVER differences, different distribution of defects, different materials and methods for evaluation of ductile-brittle transition were used. In 2020 the European project APAL (Advanced PTS Analysis for LTO) was started within HORIZON 2020 program. This project includes also benchmarking of both deterministic and probabilistic RPV integrity assessment (including thermal-hydraulic part). Among 14 participants, 5 of them are from VVER operating countries, so comparison of codes used for VVER PFM can be expected. Moreover, FAVOR developing team from US (originally ORNL, currently OCI) will participate in APAL benchmarks.	**Status:** Open **Priority:** R0 – Medium

Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-26 - Lack of sharing information of latest updated and developed probabilistic fracture mechanics codes through benchmarking (continued) **References:** NTD-ASI, Section IV - Residual Lifetime Assessment of Components and Piping in WWER NPPs; Normative-technical documentation, Association of Czech Mechanical Engineers, 2020. Theoretical manual for software package PROVER, DITI 301/295, UJV Rez, 2004. (in Czech) Users guide for code VERLOAD, DITI 301/318/R1 UJV Rez, 2005. (in Czech) Users guide for code VERPFM, DITI 301/296/R1 UJV Rez, 2005. (in Czech) Pistora V. et al.: Probabilistic Assessment of Pressurised Thermal Shocks. 21st International Conference on Structural Mechanics in Reactor Technology (SMiRT 21), New Delhi, India, 2011.	**Status:** Open **Priority:** R0 – Medium
V-AS-27 – Fatigue Usage at J-Welds in CRDM Penetrations **Issue:** Welded joints of CRDM penetrations on the VVER reactor head showed service induced degradation (geometrical instability and corrosion, respectively) **Description** CRDM penetration failures on PWRs are quite well known. These led to leakages in many plants, and the massive boric acid corrosion as a consequence of the leakage in the US based Davis Besse plant was a serious event. The design of the CRDM penetrations including their structural materials in both VVER-440 and VVER-1000 differ from that of the PWRs. Into the penetration tube made of ferritic steel a stainless steel (08Ch18N10T, equivalent to AISI 321) sleeve is placed which is fixed by two welded joints. In some VVER-440 plants geometrical instability (buckling) of the sleeve was observed which was not an integrity problem but might cause serious safety issue, i.e. by getting stuck the control rod. The reason of this phenomenon was that low-cycle fatigue during heating up and cooling down due to the different thermal physical properties of the materials took place in the upper welded joint between the penetration and the sleeve. As a result of the fatigue process, primary water penetrates through the microcracks into the gap between the tube and the sleeve. In the course of heating up the microcracks close and the medium’s thermal expansion generates stresses in the sleeve which exceed the yield strength of the sleeve’s material and lead to buckling. The root cause of the degradation was a design deficiency (extreme high CUF values were calculated in the vicinity of the weld). Since the weld position and geometry do not allow a proper NDE, a specific procedure had to be introduced. With UT it is possible to detect the water in the gap, and thus to identify the increased risk for deformation. In Loviisa NPP, Finland, the CRDM penetrations are UT inspected periodically. When water is detected between the pipe fitting and the sleeve, the sleeve will be changed and possible defects repaired to avoid buckling and possible stuck of the control rod. Usually defects in the upper weld are the primary cause of the water leakage.	**Status:** Open (New) **Priority:** R0- Medium

Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-27 – Fatigue Usage at J-Welds in CRDM Penetrations (continued)	Status:
In the case of VVER-1000 reactor heads failure were observed in the lower welded joint between the penetration and the sleeve. Here a corrosion process in the penetration tube material beneath the sleeve was also detected which led to massive repair works. The root cause of the degradation was that inappropriate repair works of the welded joints were performed in the factory during manufacturing the reactor heads. Some spherical calibers of certain diameters are now used, which pass through the holes, to detect possible deformations of the sleeve. The R&D activity should revise the design in both reactor types. Also, the most suitable monitoring / NDE method and technique could be investigated. **References:** Miscellaneous Event Report WANO Moscow Centre, 2005-018; Defects in Reactor Vessel Head CRDM Nozzles Identified during Planned Maintenance (Novovoronezh Unit 5, 31 July 2005). Sorokin, A.A., Timofeev, B.T., Slivkin, N.P.: Investigation of the fatigue strength of CRDM penetration welded joints of Novovoronezh NPP, VVER-1000, Proc. 2nd Int. Scientific and Technical Conference on Safety Assurance of NPP with VVER, Podolsk, Russia, 2001. (in Russian) Petrov, V.A. et al.: Materials science foundation of service life extension of VVER-440 RPV head based on the results of Novovoronezh 3 after 45 operation years, Proc. 2nd Int. Scientific and Technical Conference on Safety Assurance of NPP with VVER, Podolsk, Russia, 2001. (in Russian)	Open (New) **Priority:** R0- Medium

Table 3-3 (continued)
Assessment (AS) Gaps

V-AS-28 – Methods for Determination of Locations of Class 1 Piping Susceptible to Thermal Fatigue	Status: Open
Issue: There is a need for comparison and evaluation of methods used by individual operators to assess the piping segments affected by thermal fatigue as well as guidance for repair or replacement. **Description:** There are some cases of thermal fatigue degradation at VVER plants; they are limited to locations where thin-wall, small-bore piping is connected to thick-walled components. Because the degradation occurred in high-risk locations, the degradation was examined by destructive testing, and the small fatigue cracks were detected. No thermal fatigue damage has been detected on Class 1 large bore piping at VVER plants to date. In some cases, regulators have requested complete evaluation of Class 1 piping for thermal fatigue as a condition for license renewal. Therefore, this work has high priority and may be of interest also for other operators of VVER plants. Due to high similarity of VVER design it is expected that similar approaches for determination of susceptible locations and monitoring tools may be used. One approach for determination of susceptible locations is based on the screening criteria. Only the piping locations with higher temperatures and fast temperature changes are taken into consideration. Data from measurements of temperature changes and operational conditions are then evaluated, and modifications of the measurement locations are recommended in order to obtain more data [Samohyl 2019]. The general position of the industry [NEA conference] states that "…there is the need to recognize the stratification locations, where thermal fatigue could be critical. Improvements in fatigue monitoring and knowledge of sensor locations should be developed based on more detailed assessment of the operating experience…". It should be added that the attention must not be limited to the locations where thermal stratification occurs but also to other locations with fast temperature changes. Thermal fatigue guidance should identify possible approaches for determination of susceptible locations, recommended methods for field measurements, and possible repair and replacement procedures (including weld overlay, installation of additional insulations and other viable solutions). In addition, effective technical approaches to justify a weld overlay (WOL) repair solutions are needed. Effects of the WOL and the processes used to deposit the WOL on adjacent base material need to be taken into consideration. **References:** P. Samohyl, et al: Procedure for evaluation of the current heat measurements for determination of critical piping locations susceptible to thermal fatigue at NPP Dukovany, DITI 2301/934, UJV report, 2019. Fatigue of Nuclear Reactor Components, Proceedings of the 4th International Conference, 28 September-1 October 2015, NEA, Seville, Spain NTD-ASI, Section III; Normative-technical documentation, Association of Czech Mechanical Engineers, 2020.	**Priority:** R0- High

Table 3-4
Inspection and Evaluation (IE) Gaps

Gaps related to component inspection capability limitations, NDE qualification, or development of new NDE technology to effectively detect and size indications in components.

V-IE-02 - NDE Accessibility for Reactor Internals **Issue:** The availability of NDE equipment capable of accessing the critical locations of the reactor internals for inspection and successfully completing required examinations remains an industry need. **Description:** In US the MRP-227-R1 provides detailed guidance regarding inspection and evaluation requirements for reactor internal components. To date, there has not been a comprehensive assessment to identify NDE technology and tooling needs to ensure that all of the primary and expansion locations are accessible for inspection as required by the guideline. Accessibility issues, however, have been identified during the implementation of MRP-227 inspections. As plants have implemented MRP-227 inspections approaches to resolve limitations on inspection accessibility have been developed on plant specific bases. Lessons learned have been implemented as inspections have progressed from plant to plant. There is still a potential for additional fleet-wide work to be needed resolve generic access issues. EPRI commissioned a study to assess the benefit and basis for performing B-N-1 VT-3 examinations (3002012966). The report concluded that the purpose of B-N-1 examinations is to detect foreign material and debris and that other industry requirements and guidance are sufficient for detecting foreign material or debris in the reactor vessel. The EPRI report is the technical basis used for ASME approval of Case N-885. In the case of VVERs the internals construction is slightly simpler compared with US design. Both VVER-440 and VVER-1000 have removable core support structures suitable for B-N-3 category examinations according to ASME BPVC Section XI. This means basically VT-3 and in some case VT-1 (maybe EVT-1) examinations. Performing these NDE has challenges similar or same as in case of PWRs. Surface examination such as ET is not usual in VVER internals. Volumetric examination (UT) of the baffle bolting had already been performed exclusively in the Finnish VVER plant (this issue is handled separately under V-IE-01). The issue is considered to be of low priority to the VVERs based on the fact that ISI experience does not show operation induced degradation / failure on reactor internals. LTO which is the common agenda for almost all VVER plants however requires a continued and focused attention so that similar research goals to those identified for PWRs can be set. **References:** *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines MRP-227-A*. EPRI, Palo Alto, CA: (2011). 1022863. *Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228).* EPRI, Palo Alto, CA: (2015). 1016609. *Evaluation of Basis for Periodic Visual Examination of Accessible Areas of Reactor Vessel Interior per Examination Category B-N-1 of ASME Section XI, Division 1*. EPRI, Palo Alto, CA: (2018). 3002012966. ASME Code Case N-885 - "Alternative Requirements for Table IWB-2500-1, Examination Category B-N-1, Interior of Reactor Vessel, Category B-N-2, Welded Core Support Structures and Interior Attachments to Reactor Vessels, Category B-N-3, Removable Core Support Structures Section XI, Division 1." Tsvetkov, E., Heinsius, J., ENIQ-Qualified Visual Examinations by Means of Remote Controlled Submarine, atw, Vol. 60 (2015) pp. 452-456. • IAEA-SSR-82, Rev. 1, "Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL)," International Atomic Energy Agency, Vienna, August 2020.AMP 102 Inservice Inspections/ Periodic Inspections	**Status:** Open **Priority:** R0 - Low

Table 3-4 (continued)
Inspection and Evaluation (IE) Gaps

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V-IE-03 – NDE Technology for Examination of CASS

Issue:
There is a need to develop suitable inspection technologies for CASS piping components capable of accurately detecting and sizing relevant flaws in CASS associated weldments and in CASS base materials.

Description:
Detection of flaws within cast austenitic stainless steel components continues to represent a significant challenge for the industry. Field examination of CASS materials using ultrasonic examination techniques are characterized by high attenuation and scattering of the acoustic energy due to coarse grain structure, and variety of grain structures. As a result, ISI / NDE techniques currently available for examination of CASS components and piping are not effective at identifying and sizing indications.
Although recent work indicates that flaw detection in CASS materials is possible using UT in the laboratory, grain structure differences can significantly affect success. Additionally, field implementation may prove to be a challenge. Even if flaw detection is possible, there are insufficient data to characterize flaw sizing capabilities. Finally, any technique that is to be applied in the field must be sufficiently matured so that the examination limitations are well understood and considered in evaluating the results.
The stainless steels of PWRs are mainly AISI 304 and 316 with their equivalent CASS Grades CF3 and CF3M and CF8 and CF8M. The “M” designation indicates higher Mo content, generally of the order of 2.5 wt %. The increased Mo content improves castability but has been found to increase loss of ductility on thermal and irradiation aging. The typical stainless steel grade in VVER plants is 08Ch18N10T (equivalent to AISI 321) – a Titanium stabilized austenitic stainless steel. Its grain size and grain structure including the dendritic structure does not differ from that of 304 and 316. Consequently, the flaw response to the ultrasonic waves in the VVER CASS material regardless the UT technique can be considered the same as for the Western grades.
Resolution of this R&D Gap includes either development of effective NDE processes for CASS components, development of effective techniques for assessing embrittled conditions, and / or assessment of CASS some piping locations to justify a conclusion that examinations are not required. These deliverables will be applicable for VVER operators, too.

References:
Plant Support Engineering: Flaw Tolerance Evaluation of Thermally Aged Cast Austenitic Stainless Steel Piping. EPRI, Palo Alto, CA: (2007). 1016236.
Nondestructive Evaluation : Cast Austenitic Stainless Steel Round-Robin Study – Summary of Results Rev. 1. EPRI, Palo Alto, CA: (2018). 3002010314.
Nondestructive Evaluation: Methods for Characterizing Residual Stress in Metals and Thermal Aging in Cast Austenitic Stainless Steel. EPRI, Palo Alto, CA: (2015). 3002005454.
NUREG/CR-6929: Diaz, A.A., Mathews, R.A., Hixon, J., and Doctor, S.R., “Assessment of Eddy Current Testing for the Detection of Cracks in Stainless Steel Reactor Piping Components,” NUREG/CR-6929, PNNL-16253, Pacific Northwest National Laboratory, Richland, WA, February 2007.
NUREG/CR-6933: Anderson, M.T., et al, “Assessment of Crack Detection in Heavy-Walled Cast Stainless Steel Piping Welds Using Advanced Low-Frequency Ultrasonic Methods,” NUREG/CR-6933, PNNL-16292, Pacific Northwest National Laboratory, Richland, WA, March 2007.
IAEA-SSR-82, Rev. 1, “Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL),” International Atomic Energy Agency, Vienna, August 2020.
• AMP 102 Inservice Inspections/Periodic Inspections
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Table 3-4 (continued)
Inspection and Evaluation (IE) Gaps

V-I&E-04 - Steam Generator Tubing Eddy Current Technology Improvements	Status:
Issue: There is a need for evaluation of eddy current examination methods used across VVER fleet, including improvements in eddy current capabilities for probability of detection and flaw sizing and in methods for effective evaluation of eddy current system performance. Reducing NDE uncertainties could help utilities justify increased inspection intervals by providing more precise data for decision making process. **Description:** Test blocks containing tubes with EDM/artificial SCC defects exist and are used for eddy current technology and personnel qualifications. Some VVER OE indicates that eddy current inspection is a complex process and there could be some limitations of the available technology. One of the reasons may be continuous development of ET equipment and probes with improved precision and detection limits, which may cause issues when comparing outage-to-outage inspection data. This may manifest for example in unrealistically high crack growth rates if the previously used technology did not locate smaller cracks. Also, there were some cases, when ET did not indicate through-wall defect of the tube, but the bubble test was positive. One of the causes of the stress corrosion cracking of heat exchange tubes is the excessive formation of oxide layers on their outer surface and accumulation of impurities. Eddy current examination is therefore required to provide information about the thickness and location of the oxide layers. This would justify the need for evaluation and benchmarking of different eddy current testing methods used across VVER fleet, comparison to PWR experience and creation of document summarizing recommendations and good practices. **References:** *Steam Generator Management Program: Tubing Material Equivalency Assessment for Examination Technique Specifications Sheets – Application of Examination Technique Specification Technique Sheets to Materials Other than Inconel 690*. EPRI, Palo Alto, CA: (2014). 3002002824. *Steam Generator Management Program: Eddy Current Array Probe Documentation for Examination Technique Equivalency*. EPRI, Palo Alto, CA: (2017). 3002010707. *Steam Generator Management Program: Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 8*. EPRI, Palo Alto, CA: (2016). 3002007572. *Steam Generator Management Program: Development of System Performance Examination Technique Specification Sheets.* EPRI, Palo Alto, CA: (2014). 3002002847. *Steam Generator Management Program: Effect of Eddy Current Noise on Sizing Steam Generator Tube Degradation*. EPRI, Palo Alto, CA: (2013). 3002000637. *EPRI Materials Degradation Matrix, Revision 4*. (Notes v5-7a). EPRI, Palo Alto, CA: (2018). 300202013781.	Open **Priority:** R0 - Medium

Table 3-4 (continued)
Inspection and Evaluation (IE) Gaps

V-IE-06 – Identification of Cracking of Steam Generator Collector Bridges **Issue:** Horizontal SGs of VVER units have history of collector bridges cracking due to combined effect of poor secondary side chemistry and excessive stress introduced on the material of the tubesheet in the area between heat exchange tubes. **Description:** VVER1000 units have a history of collector bridges cracking [Atlas of operational defects, 2007]. The degradation mechanism was assigned to SCC from secondary side, which was a result of poor secondary chemistry and high stresses introduced during manufacturing by explosive tightening of heat exchange tubes. Many VVER1000 SGs were replaced in the past for new ones made using hydraulic tightening. For VVER 1000 this issue is currently considered to be closed. For VVER 440 it is not possible to rule out the cracking of steam collector bridges. Extension of NPP lifetime means that this part of SG is exposed to secondary environment for a long time and provides a time for the cracks to develop. Also, the long term operation means a higher number of plugged tubes, when a plug welding introduces an additional stress on the collector and at the same time may introduce “cold spot” influencing temperature and stress fields in the collector body. To date there is no direct OE to support that the VVER440 collector bridges have developed cracks. However, no qualified NDE technique exist to perform testing of such locations. There is a need for NDE technology, which would be able to evaluate conditions of the collector material. **References:** IAEA-EBP-WWER-07, “WWER-1000 Steam Generator Integrity,” International Atomic Energy Agency, Vienna, 1998. Atlas of operational defects of heat exchange SG tubes in WWER1000 plants, Ukrainian National Academy of Sciences, 2007.	**Status:** Open **Priority:** R0 - Low

Table 3-5
Mitigation (MT) R&D Gaps

Gaps related to new technology development or technology implementation that result in the mitigation of the occurrence or progression of the degradation mechanisms.

V-MT-01 - Steam Generator Tubing ODSCC Mitigation via Water Chemistry Technologies **Issue:** A clear guidance on maintaining SG crevice chemistry within a range providing trouble free operation is missing. **Description:** There is a need to develop improved secondary water chemistry approaches and monitoring techniques in order to more aggressively mitigate SCC of titanium stabilized stainless steel steam generator tubing mostly in crevices, while at the same time minimizing BOP FAC issues and associated corrosion product transport and deposition in the steam generators. In addition, an optimized approach toward off-line steam generator treatments holds promise to improve steam generator tubing materials performance through elimination of detrimental oxides and sludge deposits, while simultaneously preventing excessive degradation of non-tubing steam generator components and BOP components. SCC is identified as one of the key degradation mechanisms in iGALL116 AMP and requires implementation of preventive measures namely mitigation of fouling, implementation of proper chemistry control and maintaining high water purity. Hide-out-return monitoring is emphasized in the document as well. There is a need for guidance on how to control SG crevice chemistry within acceptable limits **References:** *Pressurized Water Reactor Secondary Water Chemistry Guidelines: Revision 8*. EPRI, Palo Alto, CA: 2017. 3002010645. IAEA-EBP-WWER-08, “Guidelines on Pressurized Thermal Shock Analysis for WWER Nuclear Power Plants – Revision 1,” International Atomic Energy Agency, Vienna, 2006. IAEA-SSG-13, “Chemistry Programme for Water Cooled Nuclear Power Plants,” International Atomic Energy Agency, Vienna, January 2001. IAEA-NER-xx-WWER, “Water Chemistry of Nuclear Power Plants,” International Atomic Energy Agency, Vienna, 2008. IAEA-SSR-82, Rev. 1, “Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL),” International Atomic Energy Agency, Vienna, August 2020.AMP 116– Steam Generators (2018). Kaplan, J., Cuba, M.: Water chemistry of the secondary and auxiliary circuits of WWER 440 and WWER 1000 plants, UJV Řež, Czech Republic, 2018.	**Status** Open **Priority:** R0 - Medium

Table 3-5 (continued)
Mitigation (MT) R&D Gaps

V-MT-02 - Steam Generator Startup Chemistry Excursions after Major Component Replacement	**Status:** Open
Issue: Although this issue is currently not considered as a critical for VVER plants, PWR plant operating experience indicates it should be getting attention. Startup chemistry excursions attributed to contaminates from either fabrication or installation practices of a secondary system component have been reported. There is a need to develop strategies to prevent or mitigate / manage this phenomenon. **Description:** A number of PWR plants have been challenged with having to cleanup from a chemistry excursion during startup as a result of contaminants from fabrication or installation of a major secondary system component. For example, replacement of turbines has resulted in a chemistry excursion for silica and / or sodium as a result of the glass beads used during fabrication. Other plants have reported primary or secondary system chemistry excursions in the cycle following steam generator replacement. To date VVER plants have no formalized guidelines for chemical/mechanical cleaning after replacement of major component. Lesson learned from PWR experiences should be carefully assessed to develop guidance to mitigate occurrences of such degradation in the future. **References:** *Advanced Nuclear Technology: Major Component Replacement Sourcebook with Applicability to New Plant Water Chemistry*. EPRI, Palo Alto, CA: (2016). 3002007516. INPO Event Report (IER) 12-52; Large Component Replacements Affecting Chemistry (2012).	**Priority:** R0 - Medium

Table 3-5 (continued)
Mitigation (MT) R&D Gaps

V-MT-03 – Guidance for Extended Layup of SGs and BOP Systems	**Status:** Open **Priority:** R0 - Medium
Issue: Technical guidance for layup of SGs and BOP systems is needed to prevent degradation of components during unplanned extended outages.	
Description: Guidance for layup of steam generators (SGs) and balance-of-plant (BOP) systems is focused on preventing degradation of components during the length of time for typical outages. The length of time for typical refueling outages has decreased over the years, and the EPRI guidelines have been modified accordingly (e.g., by allowing lower concentrations of hydrazine and amine when steam generators are in layup for less than 7 days). However, plants are periodically required to shut down for extended periods of time, sometimes for several months or longer. OE from VVER plants confirms the need for such guidance. For example, during outage SG manholes were open for extended periods of time to allow personnel entry or performing required works inside. Opening of SG was monitored only from the point of view outage duration and its critical path. This type of manipulation introduced a lot of oxygen into SG, later causing severe degradation of collector venting pipe. A need for a formal approach to ageing management of systems, structures and components is addressed in a recent IAEA Techdoc for delayed construction periods, extended shutdown and permanent shutdown prior to decommissioning. A development of the EPRI guidance would support utilities to comply with these recommendations. General technical guidance for extended layup for SGs and BOP systems based on experience of both PWR and VVER operators is needed to address such issues and prevent unnecessary degradation.	
References: *Pressurized Water Reactor Secondary Water Chemistry Guidelines: Revision 8*. EPRI, Palo Alto, CA: (2017). 3002010645. *PWR Primary and Secondary System Layup and Restart Water Chemistry Control Sourcebook*. EPRI, Palo Alto, CA: (2015). 3002005335. *Sourcebook for Plant Layup and Equipment Preservation Rev 1*. EPRI Palo Alto CA: (1992). NP-5106. IAEA-SSG-13, “Chemistry Programme for Water Cooled Nuclear Power Plants,” International Atomic Energy Agency, Vienna, 2011. IAEA-NER-xx-WWER, “Water Chemistry of Nuclear Power Plants,” International Atomic Energy Agency, Vienna, 2008. Kaplan, J., Cuba, M.: Water chemistry of the secondary and auxiliary circuits of WWER 440 and WWER 1000 plants, DITI 2302/584, UJV Řež, Czech Republic, 2018. IAEA-TECDOC-1957, “Ageing Management of Nuclear Power Plants during Delayed Construction Periods, Extended Shutdown and Permanent Shutdown Prior to Decommissioning,” International Atomic Energy Agency, Vienna, May 2021.	

Table 3-5 (continued)
Mitigation (MT) R&D Gaps

V-MT-04 - Loss of Critical Chemical Supply for Required Chemical Mitigation Issue:	**Status:** Open
Issue: Alternatives to Hydrazine are needed to ensure that corrosion mitigation can be maintained if the use of hydrazine is no longer allowed. This is a problem due to hydrazine being identified as a potential carcinogenic hazard.	**Priority:** R0 – High
Description: Currently, steam generator tube corrosion is mitigated against ODSCC by maintaining a low electrochemical potential and low oxygen concentration in the secondary side using hydrazine. In addition, some plants use hydrazine in primary circuit to generate hydrogen by its decomposition by neutron flux, while other plants use ammonia. While very effective, hydrazine is also a known carcinogen and environmental restrictions could be placed on use of hydrazine. Alternative chemicals to hydrazine to maintain low electrochemical potential and effectively scavenge oxygen need to be investigated and qualified for use. This is a shared issue between PWR, VVER and CANDU. The EPRI project on Hydrazine Alternatives has been started in 2018 and is supposed to finish in 2021. It will continue as a collaborative project with MAI (2020-2023). The EPRI reports are: 3002010652 - Update on Hydrazine Alternatives for PWR Secondary Chemistry Control: PWR Chemistry Technical Strategy Group Report 3002007608 - Steam Generator Management Program: Alternatives to Hydrazine for PWR Secondary Chemistry Control: Evaluation of Diethylhydroxylamine (DEHA)”	
References: *Update on Hydrazine Alternatives for PWR Secondary Chemistry Control: PWR Chemistry Technical Strategy Group Report*. EPRI, Palo Alto, CA: (2018). 3002010652. *Steam Generator Management Program: Alternatives to Hydrazine for PWR Secondary Chemistry Control: Evaluation of Diethylhydroxylamine (DEHA)*. EPRI, Palo Alto, CA: (2016). 3002007608. Kaplan, J., Cuba, M.: Water chemistry of the secondary and auxiliary circuits of WWER 440 and WWER 1000 plants, DITI 2302/584, UJV Řež, Czech Republic, 2018.	

Table 3-5 (continued)
Mitigation (MT) R&D Gaps

V-MT-05 – Mitigation of SCC in Nozzle DM Welds – To avoid Stainless Steel Cracking Propagating into LAS Pressure Boundary Materials	**Status:** Open
Issue: OE indicates that DMWs frequently cracks, especially nozzle welds located on the steam generator shell. One of the remedial actions identified through root cause analysis is to protect the DMW surface from secondary water by nickel plating, see related gap V-RR-06 – Validation of Nickel Plating DMW Surfaces to Protect against SCC in Secondary Water. There is a need to evaluate other possible mitigation actions for this type of degradation. **Description:** At VVER plants Dissimilar Metal Welds are made between ferritic steel (carbon steels 22K, St20 or 12 022) and austenitic stainless steel (Type 321). They have different chemical composition as well as different material properties. The analysis of several welds removed from the service [Junek, Tonarová] indicated, that one of the causes of cracking is welding in the field. This results in defects in the welds, which act as the stress concentrators, further increasing the level of the local stresses. These are already relatively high due to 33% difference in thermal expansion coefficient of DMW materials. The cracks are located along the fusion line between the ferritic material and the weld metal and are caused by SCC. The weld metal Cr16Ni25Mo6 with ferritic steel 22K shows increased sensitivity of dendrites and decreased corrosion resistance due to local drop of Chromium concentration with simultaneous increase of Carbon, Sulphur and Phosphorus. These lead to formation of significant chemical and phase inhomogeneities, which serve as the crack paths. Based on the facts the solution may be to perform the dissimilar metal weld as a shop weld, allowing higher control of the process and more thorough nondestructive testing. Also, the different weld metal needs to be used with higher corrosion resistance (increased concentration of Chromium). Work is under way to evaluate the proposed mitigating actions. **References:** Tonarová, D., et al.: Evaluation of Dissimilar Metal Welds of SG Nozzles, UJV Report DITI 2302/437, 2017. Junek, L., et al.: Final Report on Experimental Program on Selected DMW on SG Nozzles, UAM Brno report ZP6062, 2017. Ducháček, P., et al.: Orbital Welding of DMW using Sv-07Ch25N13 Weld Metal. Conference on Increase of Lifetime of Energetic Equipment, Srní, Czech Republic, 2018.	**Priority:** R0 - Low

Table 3-5 (continued)
Mitigation (MT) R&D Gaps

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V-MT-07 – Mitigation of Steam Generator Sludge Deposits and Scale Buildup

Issue:

Magnetite deposits on the secondary side of the horizontal SG tubing can cause clogging of the intertubing spaces and damage, in the form of SCC and corrosion product cracking to the tubes themselves.

Discussion:

During operation, dissolved iron in feedwater originating from a corrosion of balance of plant is deposited on the VVER steam generator tubes. These deposits then crystallize into magnetite. Consequently, SCC defects develop at the sites of these deposits as a local chemistry in the crevices between deposits and heat exchange tube is more aggressive due to hide-out of soluble impurities.

While deposition occurs in both VVER-440 and VVER-1000 steam generators, it has been known to completely fill the space between the tubes on VVER-1000 steam generators.

There is therefore a need for guidance on mitigation of deposit accumulation

The deposit accumulation can be suppressed by better control of iron transport including lowering of iron ingress by FW flow and maximizing iron removal by blowdown at the same time. An application of dispersants (e.g. PAA) or film forming products (e.g. ODA) are used by some plants and addressed by several EPRI project within Water Chemistry program. Chemical cleaning and/or different types of water lancing or SG flushing are used across PWR/VVER/CANDU fleet to remove already existing sludge from the SG, but no comprehensive guidance is available for VVERs.

Extensive SG deposits are addressed by SG iGALL 116 as a factor affecting heat exchange tube integrity. The document also refers several sludge mitigations practices.

References:

EPRI Materials Degradation Matrix, Revision 4. (Notes Y-v5-4b,Y-v6-4d) EPRI, Palo Alto, CA: 2018. 3002013781.

Triton Steam Generator Thermal Hydraulics Code (TRITON) Version 1.0. EPRI, Palo Alto, CA: (2016). 3002005515.

Steam Generator Management Program: PWR Steam Generator Deposit Characterization Sourcebook. EPRI Palo Alto CA: (2014). 3002002794.

Loop Testing of Alternative Amines for All-volatile Treatment Control in PWRs. EPRI, Palo Alto, CA: (1992). TR-100756.

Dispersants for Fouling Control: Volume 2 Short Term Trial at ANO-2. EPRI, Palo Alto, CA: (2001). TR-1003144.

Identification and Testing of Amines for Steam Generator Chemistry and Deposit Control. EPRI, Palo Alto, CA: (2002). TR-1002773.

Kukushkin, A.N., Omelchuk, A.V., Chempik, E.: Film-Forming Amines in the Nuclear Industry -- Innovative Technology. Proceedings of the 10th Conference -- Safety Assurance of NPP with VVER. OKB GIDROPRESS, 2019. (in Russian)

Yurmanov, E.V., Yurmanov, V.A., Velikopolsky, S.V., “Behavior of Corrosion Products in Secondary Systems of WWER Plants,” 8th International Seminar on Horizontal Steam Generators. EDO. OKB, GIDROPRESS, Podolsk. 2010. (in Russian)

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Status:

Open

Priority:

R0 - Medium

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Table 3-5 (continued)
Mitigation (MT) R&D Gaps

V-MT-07 – Mitigation of Steam Generator Sludge Deposits and Scale Buildup (continued)	**Status:** Open
Mayboroda, E., Zaritsky N.: Analysis of Statistic Data and Recommendations on WWER Steam Generator Ageing Management Developed by WWER Regulators Co-operation Forum's Working Group. Energoatom. 6th International Seminar on Horizontal Steam Generators. EDO. OKB GIDROPRESS, 2010. (in Russian) Vepsäläinen V.: Deposit formation in PWR steam generators. SAFIR2010, Finnish national research program on NPP safety. 2007-2010. Ikäläinen, T., Jäppinen, E., Saario, T., Sipilä, K., Bojinov, M.: Mitigation of cracking through advanced water chemistry (MOCCA). Interim Report. SAFIR 2015 –2018. The Finnish Research Programme on Nuclear Power Plant Safety. VTT Technical Research Centre of Finland Ltd. ÚJV-DITI 2302-615 EN: Database of water chemistry and SG tubes damage, Ver. 21.0 IAEA-SSR-82, Rev. 1, "Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL)," International Atomic Energy Agency, Vienna, August 2020. • AMP 116 Steam Generators Inspection program	**Priority:** R0 - Medium

Table 3-6
Repair and Replacement (RR) Gaps

Technical gaps related to development, advancement or verification of the effectiveness of repair techniques.

V-RR-01 – (Fleetwide) Replacement Strategy for Baffle Bolts

Status: Open

Priority: Low

Issue:

Despite very limited OE concerning cracking of baffle bolts from VVER plants, experience from PWRs suggests a need for development of guidance to deal with possible multiple cracked bolts in a VVER plant. There is a need to summarize the operating experience, available NDE techniques and recommendations.

Description:

To date there is only one case of known cracked baffle bolt [Ehrnsten 2011] in VVER internals, moreover this particular bolt could have cracked due to improper alignment of the baffle hole. However, according to analysis performed in the Czech Republic, the baffle former bolts (BFBs) are the most susceptible subcomponent of reactor vessel internals. The BFBs are loaded by fatigue and IASSC (both degradation mechanisms are influenced and accelerated by swelling development).

The design of bolts is different for different VVER units, especially the design of bolt heads and locking bars. The possibility of the volumetric NDT performance strongly depends upon geometry of the bolt head geometry (shape of the cavity, filling of the cavity, type of safety pin, safety lock, presence of spot weld). For some designs it is not possible to perform volumetric inspection of the bolt with existing hardware (probes, robots). The approach to prolonged operation should be based upon:

- advanced computational evaluation based upon knowledge of real irradiated material properties,
- evaluation of clustering of damaged bolts, i.e. criteria for allowable clustering,
- development/qualification of sensitive, reliable NDT UT for specific design of VVER baffle bolts,
- development of a bolt replacement procedure.

There is a need for a guidance describing respective OE from individual plants, available NDE vendors and qualified techniques, detection limits and precision as well as summary recommendation.

References:

Ehrnstén, U., et al., Investigations on Core Basket Bolts from a VVER 440 Power Plant, in: Proceedings of the 15th International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, 2011.

Table 3-6 (continued)
Repair and Replacement (RR) Gaps

V-RR-03 – Welding Process for Repair of Irradiated Material of RPV Internals **Issue:** The weldability of irradiated reactor material decreases with increasing exposure to both fast and thermal neutrons. Welding on irradiated materials is problematic due to helium that develops within austenitic materials as fluence accumulates. If the material to be welded is highly irradiated, excessive heat input can result in cracking due to diffusion and coalescence of transmutated helium in the material. The development of repair technology for RPV materials for both VVER 440 and 1000 has been finished or is currently being performed. To date no technology is available for welding repairs of small flaws in VVER internals. There is a current need to address process developments and certification of the repair technology. **Description:** With continued aging of VVER units and the implementation of more detailed examinations as a part of the internals inspection and evaluation guidelines, additional attention toward development of irradiated materials weldability guidance specific to VVERs and certified welding processes for irradiated austenitic materials is warranted. The development of such processes may be built upon already existing results from the projects for development of weld repair technology for reactor pressure vessels (RPV). The repair technology for all types of VVER 440 RPV material has been developed in the framework of EU research project SMARTWELD. The repair technology is based on Inconel 52 material using automated TIG process without need for material preheating. The technology was tested and certified and it is currently owned by the original RPV manufacturer. It was successfully used in several cases for repairs of RPV flaws in VVER 440 units. Currently, the regional R&D project for development of a new method for repair welding of VVER 1000 RPV is being performed in the Czech Republic. The aim of the project is research, development and application of new method of repair welding of VVER 1000 reactor pressure vessel wall using high nickel-based alloy filler metal based on INCONEL FM 52 material and robotized GTAW technique. The new method should correspond to the currently available technology and requirements of utilities. The testing is performed for all three principal RPV materials (base metal, welds and cladding) as well as their fusion lines. Based on the successfully finished attestation-qualification program the repair technology will be included into NTD-ASI standards. No tested and qualified technology exists today for repair of flaws in RPV internals. The qualified repair procedure should be available as a crucial part of securing long-term operation needs. Development of such technology is a multi-year effort requiring a lot of resources and inherently linked with uncertainties. Knowledge obtained in projects for development of RPV repair solutions and EPRI Collaborative Research Program should be utilized. **References:** *Welding and Repair Technology Center Status: Report on Light Water Reactor Sustainability Collaborative Research Program on Irradiated Materials Weldability*. EPRI, Palo Alto, CA: (2019). 3002015849. European Commission SMARTWELD Project, CORDIS, https://cordis.europa.eu/project/id/G1RD-CT-2001-00490. Kasl, J., et al.: Structure Analyses of Experimental Welds Designed for Repair Welding of WWER 1000 Pressure Vessel, Key Engineering Materials (Volume 647), 2015. Konop, R., et al.: Corrective welding of reactor pressure vessels WWER 1000, Conference on increasing lifetime of powerplant components, Srní, Czech Republic, 2014. NTD-ASI, Section I - Welding; Normative-technical documentation, Association of Czech Mechanical Engineers, 2020.	**Status:** Open **Priority:** R0 - Medium

Table 3-6 (continued)
Repair and Replacement (RR) Gaps

V-RR-04 – Repair Guidelines for Reactor Internals	Status:
Issue: Standard guidance for the design of repairs to VVER reactor internals is needed to aid utilities in design and qualification of repairs for internals components. **Description:** Given an increased potential for detection of flaws in VVER reactor internals due to the potential for 60 to 70-year operating lives, development of standard guidance for evaluation of repair designs and fabrication is a gap. A mechanical repair guideline document would address general design acceptance criteria for temporary and permanent repair of internals and would provide a framework for plant owners to use in evaluating repair / replacement bids and would additionally allow for consistent regulatory review and approval of repair technologies. Since the specific configuration of future repairs is unknown, initial versions of the document would generically address mechanical replacements, with focus on general attributes (i.e. design acceptance criteria, materials selection, and materials processing and fabrication guidance) that are independent of specific repair design features. Concerning the irradiated baffle bolting repair see another, specific technical gap descriptions (dealing with both mechanical repair solutions and welded repairs). The standard material for VVER internals components is 08Ch18N10T (equivalent to AISI 321), a Titanium-stabilized austenitic stainless steel. Its resistance against SCC in normal operation circumstances is acceptable. Long term thermal stability may show certain degradation in the higher temperature range after long service (LTO), especially if the material has a typical coarse grain structure. In this case, thermal activation process leads to precipitation of grain boundary carbides and thus to sensibilization of the grain boundary area against intergranular attack. This phenomenon depends on various factors such as time, or C-content and Ni-content of the steel. As a result of the above those areas of the internals components may be potential candidates for failure where a coarse grain structure is situated. The major components are thick-walled components made of forged parts. The VVER-related research should focus on the effect of the grain size on gran boundary sensibilization taking into account metallurgical variables, dwell-time, and the potential repair technologies should concentrate on these areas of the internals components. **References:** *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines MRP-227 Revision 1-A*. EPRI Palo Alto CA: (2019). 3002017168. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines MRP-227-A*. EPRI Palo Alto CA: (2011). 1022863. Mamaeva, E.I., et al.: Investigation of 100,000 hours operation and additional ageing on the microstructure, mechanical and corrosion characteristics of the welded joints of VVER-440 reactor coolant pipes, Proc. 2nd Int. Scientific and Technical Conference on Safety Assurance of NPP with VVER, Podolsk, Russia, 2001. (in Russian)	Open **Priority:** R0 - Low

Table 3-6 (continued)
Repair and Replacement (RR) Gaps

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V-RR-06 – Validation of Nickel Plating DMW Surfaces to Protect against SCC in Secondary Water

Issue:

OE indicates that DMWs often cracks. One of the remedial actions identified through root cause analysis is to protect the DMW surface from secondary water by nickel plating. Initial testing indicates that the key factor affecting stability of the nickel layer is cleaning of the weld surface. Commonly used methods described in various standards are usually applicable only for base metal and are not sufficient for weld surfaces. There is a need to develop nondestructive cleaning method, which can be used for reliable cleaning prior to application of nickel plating procedure on welds.

Description:

Operating experiences show numerous cases of cracking of Dissimilar Metal Welds, majority of them from the secondary water side.

One of the first cases was cracking of DMW of SG collector on VVER 440. The root cause of this cracking was combination of SCC and anodic dissolution of weld metal along the fusion line. The repair solution for this particular location was developed and reliably applied in several cases. Thorough NDE process indicates that there is no degradation of these newly repaired DMW. Another possible solution is to use weld overlay technology developed by EPRI.

However, there are other DMWs, where the problem of weld degradation is persisting. They are among others SG nozzles for draining and sludge removal and cooling. These welds are degraded repeatedly, even though the repairs are done according to the best knowledge and following the current standards, there are still cases of repeated failures.

The root analysis of these defects [Junek] proposed several remedial actions. Among others it is application of different welding technology and different weld metal as well as utilization of nickel plating on the weld surface in contact with secondary water. The plating should prevent access of the water to the surface of the weld and thus protect it against degradation. The DMW is made in the shop, then the nickel plating is applied, and the assembly is welded into piping using homogenous welds.

In order to ensure proper protection of the surface by nickel plating, it needs to be thoroughly cleaned. ASTM standards describe procedures to clean base material before plating. The testing has shown that this method does not clean whole surface of the weld. Some oxides are not removed, and they locally prevent proper adhesion of the nickel layer on DMW surface.

Additional work would be needed to characterize the method for reliable nondestructive cleaning of DMW surface, which would ensure correct application of nickel plating process and would withstand operational parameters of piping.

References:

Tonarová, D., et al.: Evaluation of Dissimilar Metal Welds of SG Nozzles, UJV Report DITI 2302/437, 2017.

Junek, L., et al.: Final Report on Experimental Program on Selected DMW on SG Nozzles, UAM Brno report ZP6062, 2017.

ASTM B254-92(2020) e1, Standard Practice for Preparation of and Electroplating on Stainless Steel, ASTM International, West Conshohocken, PA, 2020.

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Status:

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Priority:

R0 - Medium

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4 SUMMARY OF RESULTS

There are 37 R&D (DM, AS, I&E, MT and RR) gaps to be opened in this VVER IMT. As noted earlier these gaps were derived from the listing of analogous gaps for western style PWRs in the 2020 revision 4 of the PWR IMTs, MRP-205. Note that in addition to these 37 gaps that were identified as being relevant to VVERs, another 17 gaps, derived from the PWR gaps, were considered by the workshop team but were deemed to be not relevant to the VVERs. For completeness these proposed gaps have been captured in this document, together with the rationalizations for considering the non-relevant The distribution of these gaps according to their priorities is given in Table 4-1. Section 4.1 provides a distilled summary of the high priority R&D areas for the MRP and SGMP in terms of a proactively managing materials degradation in PWRs consistent with the intent of the NEI 03-08 materials initiative.

Table 4-1
PWR IMT Gaps by Category and Priority

	DM	AS	IE	MT	RR	Total
High	1	8	1	1	0	11
Medium	2	5	1	4	2	14
Low	3	4	2	1	2	12
Total Active Gaps	**6**	**17**	**4**	**6**	**4**	**37**
Non-relevant/Closed Gaps	***1***	***11***	***2***	***1***	***2***	***17***

4.1 High Priority R&D Areas

The majority of the high priority gaps are Assessment gaps with a significantly lower number of Inspection and Examination gaps. There are then, in turn, fewer Mitigation and Repair / Replacement gaps. This distribution reflects the perspective that mechanisms of degradation in PWRs are relatively well understood and that currently, the key gaps relate to the ability to evaluate the extent of degradation and to predict the ability to continue operations without extraordinary actions. The lower number of Inspection and Examination gaps reflects the concerns for effective monitoring to obtain advanced indications of on-setting degradation. Finally, the Mitigation and Repair / Replacement gaps reflect the needs for remedial actions to be put in place to counteract aging effects or to “reset” the aging degradation process.

The high priority gaps mainly fall into eight of the 12 materials focus research areas (MRFA) that EPRI employs to characterize research programs. The MRFAs consider specific categories

Throughout the consideration of the gaps, a consistent effort was made to ensure that issues important to international members are identified so that R&D proposed for future years can be more effective in addressing international member needs.

4.1.1 Reactor Internals (MRFA 1)

The only identified high priority gap in reactor internals is the need to assess the significance of flow induced vibrations on the fatigue and wear of baffle assembly components. Research is needed to determine the factors that lead to accelerated wear in control rod guide cards and other sections of the internals. The effect of high frequency vibrations and other fatigue loadings on the accelerated failures of baffle bolting and thermal shields needs to be more quantitatively understood. (Note that research needs for the specific factors associated with the understanding and evaluation of internals aging degradation are considered in MRFAs for the specific materials and degradation modes as described below. For instance, IASCC of baffle bolting materials, while being key to the integrity of reactor internals is specifically considered in MRFA 2 Stainless Steels.)

As noted below in the MRFA 2 section, for VVERs there is a special need to understand, measure and predict potential void swelling effects in the thick section Baffle-Barrel forgings in the VVER-1000 reactor vessel internals.

4.1.2 Stainless Steels (MRFA 2)

The majority of the high priority gaps for stainless steel materials relate to the behavior of these materials after or under irradiation and in the presence of PWR coolant. These behaviors are considered to be driven by stress corrosion cracking processes (IASCC) although the possible additive effects of simultaneous fatigue processes need to be better understood and quantified. For IASCC processes there exists a significant gap between laboratory measurements, which are conducted under conditions which produce rapid crack initiation and growth, and in-reactor behaviors which occur much more slowly; for example, most IASCC crack initiation testing is conducted for (often much) less than 3000 hours whereas IASCC service failures occur at times generally above 100,000 hours. There are also limitations of the database with respect to the effects of different grades of austenitic stainless steels, the effects of dose and differences in water chemistry.

For VVER-1000 plants there is a need to develop a quantitative understanding of void swelling. Data are needed to predict void swelling rates as a function of neutron flux and local temperature. This need is driven by the identified potential for swelling in the thicker sections of the baffle-barrel forgings of the VVER-1000 where the potential for gamma-heating could drive local temperatures into the regime where swelling effects become exacerbated. In addition to laboratory studies there is a need to perform practical in-situ measurements of baffle geometries to determine if operating systems are experiencing such swelling.

4.1.3 Nickel Alloys (MRFA 3)

Since nickel alloys are not used in VVERs there are no gaps related to this EPRI MRFA. The only use of Nickel alloys is as an electrodeposit over steam generator tubing and weld. This topic is covered in the repair MRFA.

4.1.4 Low Alloy Steels (MRFA 4)

Neutron irradiation embrittlement of low alloy steels in reactor pressure vessels remains the most important concern for low alloy steels. Additionally, more information is needed on the environmental effects on the fracture and fatigue behavior of coolant on VVER steels is needed.

In contrast to the concerns of the western style PWRs dose rate effects, neutron embrittlement of nozzle sections and improved calculations of flux and fluence levels were not deemed by the workshop to be of immediate concern for VVERs and there is therefore no need to include such research activities in VVER materials support programs.

4.1.5 Fatigue (MRFA 5)

In an analogous manner to the research needs for western style PWRs, fatigue gaps relate principally to the ability to reliably predict the locations of and the expectations for the times to onset of cracking from fatigue damage within the primary system structures. The major group of gaps pertain to the ability to more accurately calculate fatigue damage in the presence of water. As lifetimes have been extended, previous calculations of fatigue damage in the presence of water, based on the environmentally impacted Cumulative Usage Factors ($\sum CUF_{en}$), which have previously been known to be conservative are now giving rise to calculations that predict in-environment fatigue damage approaching threshold levels within component's extended lives. For these cases the evaluation of the expected onset of deleterious effects must be more accurately determined. To accomplish this, better laboratory measurements of fatigue behavior in PWR coolant (or the effect of the presence of coolant on fatigue lives) need to be obtained. The study of the effect of environment on the properties of highly irradiated, i.e. baffle materials, as well as non-irradiated, i.e. piping, materials is encouraged.

Unlike the gap list for western style PWRS wherein a second set of technical concerns had arisen related to the incorporation of previously unexpected load cycles into fatigue damage evaluations due to unexpected seismic events, the workshop determined that there was no need to investigate this phenomenon for VVERs. Noting that this concern was effectively driven by the events of the Fukushima earthquake, the workshop noted that no VVERs are currently sited in areas of even potentially high seismic activity, and therefore these concerns, and the need for research into them would not be relevant to the VVER fleet.

4.1.6 Wear (MRFA 6)

There is still a need to better assess damage in steam generator tubing that occurs by wear and abrasion due to loose parts and foreign objects.

4.1.7 Water Chemistry (MRFA 8)

Improved understanding of the chemical effects on secondary side in steam generators is still needed for VVERs, particularly when these effects control sludge deposition and scale build up. Due to the horizontal positioning of the heat transfer tubing in VVER there are more concern for the effects of deposits on the secondary side of the tubing: chemical treatments and water chemistry controls are needed to mitigate against ODSCC of the stainless steel tubing.

Some research is needed into how more optimized water chemistry controls can be implemented to minimize effects during plan restarts, and, in particular, after the plant has undergone extensive downtime during extended layup. Similar work is also needed to develop optimum water chemistry practices during plant start up after major component replacement.

In a similar manner to the concern for chemical supplies for water chemistry control in western style PWRs, VVER plant need to search for alternative chemical approaches to potential replace those materials, such as Hydrazine, that are currently used but may in the future become of restricted use due to regulatory or cost concerns.

4.1.8 Repair and Fabrication (MRFA 9)

The high priority gaps for repair and fabrication technologies relate to the potential welding repair of irradiated materials and the more effective development of processes to apply Nickel plating as a protective layer over dissimilar metal welds.

Although PWR internals have not yet required weld repair, as plant lives are extended and the potential for aging driven cracking becomes greater, then local weld repair may present an economic alternative instead of complete replacement of internals. However, as stainless steels become increasingly irradiated, welding becomes more difficult. Low heat input welding based on laser or solid state joining offers the potential for overcoming the difficulties encountered when welding irradiated materials using conventional arc welding techniques. In order for such improved techniques to be available for use when plants reach the ages when weld repairs will be needed, it is recommended that development should be initiated now.

VVER plants employ plated nickel layers over dissimilar metal welds in order to exclude the water environment from the surface of the DMW which may be susceptible to SCC. Improved reproducibility and reliability processes are needed for plating deposition. To support these developments assessment of structures and performances of the deposited layers are needed from future research programs. Currently there appears to be no need to address alternative DM repair solutions.

In contrast to Western Style PWRs, VVER reactor systems do not include high Nickel-Chromium-Iron alloys like alloy 600 and alloy 690. There is therefore no driver to address performance and behavior of these alloys and their weld metals for VVERs.

4.1.9 Inspection (MRFA 11)

High priority gaps for inspection technology are related to the ability to inspect cast austenitic stainless steels and to more accurately interpret NDE signals from steam generator tube examinations. The complex structure of CASS continues to challenge UT inspections in their ability to discriminate real flaws from artifacts of the microstructure. For eddy current inspections of steam generator tubing, improved probes, more accurate modeling and faster analysis tools are needed to enhance the speed of inspections that can have critical impacts on the lengths of outages. Specifically, for VVER plants, improved technologies for detection of cracking in steam generator collector bridges is required.

5
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IMT Process References for Sections 1 and 2

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A
REVISION LOG

Table A-1 provides a summary of revisions for each version of the PWR IMT.

Table A-1
Revision Log

Revision	Summary of Changes
Revision 0 (2021)	Initial issue

B
GAP CLOSURE

Table B-1 will provide a summary list of the R&D gaps closed in subsequent revisions of this VVER IMT. In this revision, this table is being used to document those gaps that were proposed from the initial listing of the PWR IMT gaps but were deemed by the workshop team as being not relevant to the VVER units. Table B-2 provides the technical discussion of each of these non-relevant R&D gaps including rationales for why they have been identified as being "non-relevant".

Table B-1
List of PWR Gaps Identified as Non-relevant for VVERs

Gap ID	R&D Gap Title
V-DM-06	**Thermal Embrittlement of Low alloy Pressure Vessel Steels**
V-AS-02	**Improved Method of Calculating Accumulated Fluence in RPV and Reactor Vessel Internals Structures**
V-AS-11	**Cracking of Steam Generator Collector Bridges**
V-AS-12	**ODSCC of Stainless Steel Steam Generator Tubing at High Alkalinity**
V-AS-13	**Steam Generator Flow-Accelerated Corrosion Assessment**
V-AS-15	**Fluence Spectra and Dose Rate Effects on Low-Alloy Steels RPV Materials**
V-AS-16	**Neutron Embrittlement of Nozzle Forgings and Upper Shell Course**
V-AS-21	**High Cycle Fatigue Potential at branch Line Locations**
V-AS-22	**Outstanding Issues Associated with Thermal Fatigue of ASME Class 1 Piping**
V-AS-23	**Lack of appropriate methods for evaluating low-cycle fatigue of elbow pipes with welds under seismic loading**
V-AS-24	**Lack of methods for evaluating non-proportional multiaxial low cycle fatigue of pipes under seismic loading**
V-AS-25	**Lack of an advanced method for determining the number of fatigue cycles as part of the low cycle fatigue evaluation for piping under seismic loading**
V-IE-01	**NDE Technology for Detection and Characterization of Baffle and Former Assembly IASCC**
V-IE-05	**NDE Qualification for Reactor Internals Inspection (VT Evaluation)**
V-MT-06	**Develop Recommendations and Guidance for Mitigating Fatigue Failures in Piping**
V-RR-02	**Repair/replacement Guidance for Thermal Fatigue of ASME Class 1 Piping**
V-RR-05	**Alternative DM Weld Repair Solutions**

Table B-2
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

<table>
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V-DM-06 Thermal Embrittlement of Low alloy Pressure Vessel Steels

Issue:

Thermal exposure is known to embrittle low alloys steels. However, there is no issue relevant to VVERs

Description:

Carbon steel welded boilers operated since long time in the non-nuclear industry at elevated temperature. Thermal aging surveillance data available at reactor operating temperature for the VVER reactor pressure vessel steels, their weldment and HAZ. There is limited or no need to better characterize of the carbon and low alloy steels at higher temperature and after long term ageing except the HAZ.

Limited, but significant effect of thermal ageing of HAZ expected at the pressurized components of VVER units.

A U.K. modelling study shows that, at 290°C, steels with 0.25% Cu content suffer only about 5°C thermal degradation (DBTT shift) during 100 years; however, in the case of 0.5% Cu content, a 90°C thermal ageing shift is expected. This calculation agrees with industrial experience.

Research project result show that the middle section of the forged low alloyed 15Ch2MFA at 350°C suffered only limited or no thermal ageing. However, the near surface layers having much better DBTT than the middle section are affected, and after 3500 hours ageing the ductility and fracture toughness degraded near to the level of the middle section. At 350°C similar effect can be expected at the thick carbon steel vessels.

The VVER-440 units operated at relative low temperature; the maximum coolant temperature is below 300°C. In case of the VVER-1000 the coolant temperature can reach the 329 °C. Most of the pressurized vessels (pressurizer, steam generator) made from 22K type carbon steel and cladded inside by stainless steel. In case of low alloyed and carbon steel forgings and plates operated below 300 °C thermal ageing is not experienced in the industry if the copper content does not exceed 0.25%. The available information about the HAZ is few. Especially in the case of the HAZ between the welded austenitic cladding and the carbon base material is limited, however this HAZ is too thin to initiate dangerous cracks.

Note that while the technical issue seems to be resolved, operators may need to provide technical justification on this point to regulators for continuing operations. On that basis this gap would become a regulatory issue.

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Murthy, K., *Materials Ageing and Degradation in Light Water Reactors: Mechanism and Management*. Elsevier 2013 February 18 p.18-440.

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Status:

Closed/No Gap

Priority:

R0 – Closed

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Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-DM-06 Thermal Embrittlement of Low alloy Pressure Vessel Steels (continued)	**Status:** Closed/No Gap
Ondrouch J., Mechanical Properties of Dissimilar Weldment After Long-Term Ageing. Proceedings of Dissimilar Welded Joints in NPP Equipment and Piping: Problems and Means to Solve Them, St. Petersburg. 1995. p.70. Gillemot F., Gillemot L., Uri G., Pirfo S., Oszvald F., “Synergetic effects of irradiation and thermal embrittlement,” IAEA Specialist Meeting “Irradiation embrittlement and mitigation,” Vladimir, Russia. 1997. Kuleshova, E.A., et al., Degradation of RPV steel structure after 45 years of operation in the VVER-440 reactor. Journal of Nuclear Materials, 540 (2020). Sharma, T., Bonagani, S.K., Kumar, N., Mani, H.D., Krishna, K.V., Samajdar, I. : Effect of Thermal Aging on Embrittlement of Cr–Mo–V Pressure Vessel Steel. Journal of Nuclear Materials, 527 (2019). Gillemot F. et al.: “Review report on trend curves, flux effect, thermal ageing and late blooming including analyses of different databases.” LONGLIFE EU Research project deliverable D2.4. 2014. Beznosikov S.A., Mjasnjankin A.I., Veselovsky V.F., Timofeev B.T., Filchagin Yu. Assessment of State of Properties for Base Metal and Welds of NPP Pipe-Lines after Prolonged Operation. Proceedings of Workshop on Ageing of NPP Component Materials, St. Petersburg, 1995, p.221-232. Dauniene, S.A., Karzov, G.P, Timofeev, B.T., Timofeeva, A., Experimental Assessment of Sensitivity of Weld Metal to Strain Ageing. Sudostroitelnaja Promyshlennost, serja "Svarka", issue 20, p. 17-21. (1975). Gaponov, A. The State of the Du-500 Piping of the WWER Main Circulating Circuit after 100,000 hrs Operation. *Teploenergetika*, No.l, p.43-44. (1989). Karzov, G.P., Timofeev, B. Assessment of Degradation of Mechanical Properties of Steels and Weldments for Pressure Vessels and Piping. in: Proceedings of the Joint EC – IAEA Specialists Meeting on NDT Methods for Monitoring Degradation. Petten, 1994, p.94-103. Karzov, G.P., Timofeev, B.T., Filchagin, Yu.G. The Influence of Prolonged Exposure to Operating Temperatures and Strain Aging on the Material Properties of Nuclear Power Plant Reactor Vessels and Piping. International Journal of Pressure Vessel and Piping. Volume 53, No.2, 1993, p. 195-216. PNAE G-7-002-86*, Strength Calculations Norms for Nuclear Power Plant Equipment and Piping*. Energoatomizdat, Moscow, 1989. Timofeev, B., Shalygin, A. Assessment of Variation of Properties for Welded Joints of Piping from Carbon Steels during Design Lifetime. *Voprosy Materialovedenija*, No.3(39), p.54-61. (2004). Timofeev, B., Thermal Ageing of Russian RPV Materials Used for NPP with LWR. Proceeding of the 15th International Conference “Building Services -- Mechanical and Building Industry Days.” Debrecen, Hungary. Mechanical Engineering Section, p.55-66. 2009. Bazaras, Z. and Timofeev, B.: Effect of Long-Term Thermal Influence on Mechanical Properties of Welded Joints for Carbon Steels used in Power Engineering, Steam Generator Systems: Operational Reliability and Efficiency, Dr. Valentin Uchanin (Ed.), 2011.	**Priority:** R0 – Closed

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-AS-02 – Improved Method of Calculating Accumulated Fluence in RPV and Reactor Vessel Internals Structures **Issue:** More accurate methods of calculating accumulated fluence are needed for reactor vessel internals and for non-traditional RPV locations **Description:** The fast neutron fluence values are determined by calculations. The calculation results are accompanied by measurements of activities of the activation foils placed in the capsules behind the RPV at selected locations, namely in azimuthal profile and by measurement of induced activities of the activations monitors placed in capsules together with specimens of surveillance monitoring program. The measured experimental values are used for the verification and adjustments of calculated doses absorbed by RPV material if necessary. For RVI only calculated values are used. Both dpa and N/cm^2 values are determined. In case it is necessary to evaluate the fluence in nontraditional locations (for example supports of the RPV) temporary activation monitors are installed behind RPV wall to verify the dose calculation. Originally used 2D and 3D deterministic transport code approach for doses calculation has been replaced by MCNP code since 2019. So far there is a good agreement between calculations and measured values, and the need for more accurate methods is not imminent. Regulatory requirements may force utilities to calculate fluences in other locations, currently not covered. This would require further work but the reliable calculation tools are available and there are no problems expected. **References:** Association of Mechanical Engineers of Czech Republic, Design and Life Time Assessment of Components and Piping in VVER NPPs During Operation, Section IV, Appendix II, version 2020. Method for evaluation of radiation load of RPV and RVI (in Normative Technical Documentation) Guidelines for Integrity and Lifetime Assessment of Components and Piping in WWER Nuclear Power Plant (VERLIFE), 2014. • Appendix II Procedure for Determination of Neutron Fluence in Reactor Pressure Vessels and Reactor Vessel Internals.	**Status:** Closed/No Gap **Priority:** R0 – Closed
V-AS-11 – Cracking of Steam Generator Collector Bridges **Issue:** Horizontal SGs of VVER units have history of collector bridges cracking due to combined effect of poor secondary side chemistry and excessive stress introduced on the material of the tubesheet in the area between heat exchange tubes. **Description:** It has been determined that this gap is not relevant as an Assessment (AS) Gap; rather, it should be characterized as an Inspection and Evaluation (IE) Gap. Therefore, this issue is addressed under V-IE-06.	**Status:** Closed/Revised as V-IE-06 **Priority:** R0 – Closed

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-AS-12 – ODSCC of Stainless Steel Steam Generator Tubing at High Alkalinity	**Status:** Closed/No Gap
Issue: *The issue is a duplicate of V-DM-04, which is kept open with high priority. The V-AS-12 is therefore closed, because there is no need for duplication.* A guidance on crevice pH(T) control is needed. The aim of such guidance should be to prevent ODSCC cracking of Type 321 steam generator tubing. The steam generators of VVERs use titanium stabilized stainless steel tubing at both WWER440 and VVER1000 designs. This material is known to be susceptible to SCC initiated from secondary side (outer diameter) in acidic environment, but according to chromium oxide solubility curves the steel is susceptible to degradation in (highly) alkaline pH as well. **Description:** The titanium stabilized stainless steel is extensively used in WWER design and it is also a material used for steam generator. Although the material has an outstanding performance record for using in nuclear plants it is known to be susceptible to SCC in acidic environment. As a result, the secondary chemistry focuses on minimizing the chloride and sulfide ingress to mitigate ODSCC. Also, micro-dosing of alkali hydroxide is used at some plants to shift crevice pH(T) to higher values. To date only the acidic crevice pH(T) was a concern, but extensive retubing of VVER fleet condensers for stainless steel improved secondary chemistry and highly alkalic crevices (according to Hide Out Return calculations) are observed in some plants. The stainless steel is susceptible to corrosion attack in crevice environment and there is a need for developing the generally acceptable limits for the material to meet the requirement on long term safe operation. **References:** *EPRI Materials Degradation Matrix, Revision 4 (Notes v5-7a, v5-7b)*. EPRI, Palo Alto, CA: (2018). 3002013781. *Steam Generator Reference Book, Revision 1, Volume 1, (Table 24-31)*. EPRI, Palo Alto, CA: (1994). TR-103824-V1R1.	**Priority:** R0 – Closed

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-AS-13 - Steam Generator Flow-Accelerated Corrosion Assessment	Status: Closed/No Gap Priority: R0 - Closed
Issue: There is no issue relevant to VVERs **Description:** This issue was initially identified in heavily fouled PWR steam generators where the egg crates adjacent to the support wedges experienced mild to severe corrosion. Some egg crates were actually, found to be missing. A buildup of deposits on the tubes reduced the available flow area within the tube bundle and increased periphery flow. Increasing velocity gradients at the periphery-initiated vortices. Another concern is the potential for FAC of carbon steel moisture separator assembly components. OPEX at VVER steam generators shows fouling on the outer surface of the heat exchange tubing (material: 08Ch18N10T) by corrosion products from the secondary circuit. Sludge deposits within the tube bundles are found in the most VVERs. These deposits were also observed at the supports of the pipe bundles, which can cause SSC on the pipes. However, failure of the pipe support structure has not been reported in VVER SGs, yet. According to the VVERs design basis, the heat transfer thermal capacity of the SG (e.g. number of the heat exchanging tubes) assures at around 10-15% reserve, so a few % tubes plugging does not affect the operational safety of the SGs. On the other hand, in horizontal SGs of VVERs, the flow velocities are in the range of 2-3 m/s, which is significantly lower than the flow velocity in PWR, so no FAC is expected. This finding also applies to potentially fouled piping support structures as well as to the carbon steel moisture separator assembly components. FAC at Feedwater headers inside a steam generator has been observed to result in heavy degradation of this component in VVER440 units. A replacement of those carbon steel headers for stainless steel was made at all units and FAC is no longer a concern for them. Only a stub of a header upstream of DMW or flange is affected, but it is addressed by NDT. FAC of SG vessel is addressed by visual testing of square area at water level within In-Service-Inspection program and no FAC development has been observed so far. FAC is a well-known degradation mechanism so no additional R&D is needed specifically for VVER and appearance of the FAC is not expected in VVER steam generators. **References:** *EPRI Materials Degradation Matrix, Revision 4* (Note v5-3c), , EPRI, Palo Alto, CA: (2018). 300202013781. *Correlation of Flow Accelerated Corrosion of Steam Generator Internals with Plant Water Chemistry*. EPRI, Palo Alto, CA: (1998). 111113. *Steam Generator Management Program: Investigation of Steam Generator Secondary Side Degradation*. EPRI, Palo Alto, CA: (2011). 1022830. RD EO-0552-2004: Guideline for Application of System Approach to Integrity Control of SG Tubes at Operating NPPs with WWER-1000 and WWER-440 Type Reactors, Gosatomenergonadzor, 2004. IAEA-SSR-82, Rev. 1, “Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL),” International Atomic Energy Agency, Vienna, August 2020. IAEA-TECDOC-1577, “Strategy for Assessment of WWER Steam Generator Tube Integrity,” International Atomic Energy Agency, Vienna, December 2007. ÚJV-DITI 2302-615 EN: Database of water chemistry and SG tubes damage, Ver. 21.0	

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-AS-15 - Fluence Spectra and Dose Rate Effects on Low-Alloy Steel RPV Materials	**Status:** Closed/No Gap **Priority:** Closed
Issue: Additional evaluation might be required to assess the relative significance of a "flux effect" on the embrittlement of low-alloy steel RPV materials subjected to significant neutron fluence. Although the fluence spectra undoubtedly has an effect on material degradation levels, the data from the fast reactors are not used for RPV integrity evaluations in the Czech Republic. **Description:** RPV steel embrittlement data from high flux test reactors has been used to produce high fluence materials in order to generate data for embrittlement trend correlation. It has been shown that the higher flux has an effect on both microstructural evolution and can have a higher apparent embrittlement tendency at the same fluence levels as power reactor irradiated RPV steel. There would therefore be a concern if high fluence data from fast reactor exposures of materials were used to predict embrittlement trends for thermal reactor materials. While experiments are in progress in the USA to explore this phenomenon, data from fast reactor exposure are not used to predict embrittlement trends in VVER reactor pressure vessels. There is, therefore, no gap with respect to spectral and dose rate effects in predicting the embrittlement trend behaviors of materials in VVER pressure vessels **References:** *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines MRP-227-A*. EPRI, Palo Alto, CA: (2011). 1022863. *Materials Reliability Program: Static Tensile Testing of a Pressure Vessel Steel Irradiated to Assess Though-wall Radiation Embrittlement (MRP- 333).* EPRI, Palo Alto, CA: (2012). 1024798. 10 CFR 50 – Title 10, Code of Federal Regulations Part 50 , U.S. Nuclear Regulatory Commission, Washington, DC. Appendix G, "Fracture Toughness Requirements," 2013.	

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V AS-16 Neutron Embrittlement of Nozzle Forgings and Upper Shell Course	Status: Closed/No Gap
Issue: There is no issue relevant to VVERs **Description:** At the VVER reactors the upper shell ring is forged and the material is the same type as the beltline ring. The chemical composition is tested and known. During the PTS analysis for the service life extension up to 60 years the upper rings with the nozzles are considered. The cladding of the shell is made by submerged arc strip welding with austenitic electrode. The interface between the cladding and the base material and the heat affected zone are very narrow. The cladding of the nozzle corners are made by manual arc welding, therefore the interface and the HAZ here is larger and inhomogeneous, The VVER-440 reactor is a compact design, the vessel diameter is small, consequently the lifetime neutron fluence at the beltline vessel wall is high (about 2-4*10^{20} n/cm^2 E<0,5 MeV) compared with the other PWRs. The EOL neutron fluence of the VVER-1000 units are slightly higher than the other PWRs. The irradiation embrittlement of base material forgings and circumferential welds between the forged rings of the vessel are well known from the surveillance testing and from the extended surveillance programs. The transition material of the nozzle corners is near to the thermal ageing surveillance specimens, showing no degradation. The estimated EOL fluence exceeds the 10^{17} n/cm^2.value but it does not cause significant degradation. Nondestructive testing of the nozzle corners reducing the probability of cracking from the transition zone of the manual arc welding clad of the nozzle corner, but the irradiation effect of this area is not widely studied yet. **References:** IAEA-TECDOC-670, "Pilot Studies on Management of Ageing of Nuclear Power Plant Components, Results of Phase I," International Atomic Energy Agency, Vienna, 1992. IAEA-EBP-WWER-08, Rev. 1, "Guidelines on Pressurized Thermal Shock Analysis for WWER Nuclear Power Plants," International Atomic Energy Agency, Vienna, January 2006. Debarberis, L. Acosta, B, Sevini, F, Kryukov A., Gillemot, F. Valo, M. Nikolaev, V. Brumovsky, M. : Role of Nickel in a Semi-mechanistic Analytical Model for Irradiation Embrittlement of Model Alloys. Journal of Nuclear Materials, 336 (2005) 210-216. Debarberis, L., Sevini, F., Acosta, B., Kryukov, A., Nikolaev, Y., Amaev, A., Valo, M.: Irradiation Embrittlement of Model Alloys and Commercial Steels: Analysis of Similitude Behaviors. Pressure Vessel and Piping, 79 (2002) 637-642. Nikolaev, Yu., Nikolaeva, A., Shtrombakh, Y.: Radiation Embrittlement of Low-alloys Steels. Pressure Vessel and Piping, 79 (2002). Bohmert, J., Ulbricht, A., Kryukov, A., Nikolaev, Y., Erak, D.: Composition Effects on the Radiation Embrittlement of Iron Alloys. ASTM STR 1405, 2001. Bohmert, J., Viehrig, HV. Ulbricht, A.: Irradiation Effect on Toughness Behavior and Microstructure of VVER-type Pressure Vessels. Journal of Nuclear Materials, 297 (2001) 251-261. Miller, M.K., Russel, K.F., Kocik, J., Keilova, E.: Embrittlement of Low Copper VVER-440 Surveillance Samples Neutron-irradiated to High Fluences. Journal of Nuclear Materials 282 (2000) 83-88. Nikolaev Yu.: Radiation Embrittlement of Cr–Ni–Mo and Cr–Mo RPV Steels. Journal of ASTM International, Vol. 4, No. 8. Shtrombakh, Ya. and. Nikolaev, Yu.: Monitoring of Radiation Embrittlement of the First and Second Generation of VVER RPV Steels. Journal of ASTM International, Vol. 4, No. 5.	**Priority:** Closed

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-AS-21 – High Cycle Fatigue Potential at branch Line Locations **Issue:** There is no issue relevant to VVERs **Description:** High-cycle fatigue due to thermal cycling (turbulence penetration or thermal stratification) are design/location dependent phenomena and therefore are applicable to primary system piping components. At VVER plants there is only one event for high-cycle fatigue at reactor coolant system branch lines due to thermal cycling issue is reported. A fatigue life analysis for pressurizer surge line of Unit 3 Bohunice NPP, Slovakia showed high CUF value at an elbow (CUF=0,96). The elbow was replaced and investigated by metallography which did not prove irregularities of material structure giving evidence of advanced cyclic damage. The high cycle fatigue degradation mechanism is generally well known and characterized and also appropriately monitored by effective AMPs at VVER plants so no additional research is considered as necessary. **References:** *EPRI Materials Degradation Matrix, Revision 4*. (Notes v3-8a,b) EPRI, Palo Alto, CA: 2018. 3002013781. Guidelines for Integrity and Lifetime Assessment of Components and Piping in WWER Nuclear Power Plant (VERLIFE), 2014. NP 306.2.099-2004: “General requirements to lifetime extension of NPP power units for operation beyond the design period based on the results of periodic safety reviews,” State Committee for Nuclear Regulation of Ukraine. Kyiv, 2004. (in Ukrainian) PL-D.0.03.126-10: “Regulation on the procedure for extending the life of equipment of systems important to safety,” State Committee for Nuclear Regulation of Ukraine, Kyiv, 2010. (in Russian) SOU NAEK 080:2014 Operation of Technological System. Long-Term Operation of NPP Units. General Provisions /6/ PM-D.0.03.222-14: “Standard program on ageing management of NPP structures and components,” NNEGC Energoatom, Kyiv, 2014. (in Russian) NP 306.2.210-2017: “General Requirements for Ageing Management of Components and Structures and Long-Term Operation of NPP Units,” State Committee for Nuclear Regulation of Ukraine. Kyiv, 2017. IAEA-SSR-82, Rev. 1, “Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL),” International Atomic Energy Agency, Vienna, August 2020. • AMP 101 Low Cycle Fatigue Monitoring • TLAA 106 Environmentally Assisted Fatigue	**Status:** Closed/No Gap **Priority:** R0 - Closed

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-AS-22 – Outstanding Issues Associated with Thermal Fatigue of ASME Class 1 Piping **Issue:** There is no issue relevant to VVERs **Description:** High-cycle fatigue due to thermal cycling (turbulence penetration or thermal stratification) are design/location dependent phenomena and therefore are applicable to primary system piping components. At VVER plants there is only one event for high-cycle fatigue at reactor coolant system branch lines due to thermal cycling issue is reported. A fatigue life analysis for pressurizer surge line of Unit 3 Bohunice NPP, Slovakia showed high CUF value at an elbow (CUF=0.96). The elbow was replaced and investigated by metallography which did not prove irregularities of material structure giving evidence of advanced cyclic damage. High-cycle fatigue due to thermal cycling (turbulence penetration or thermal stratification) or mechanical vibration are design/location dependent phenomena and therefore are applicable to primary system piping components. The high cycle fatigue degradation mechanism is generally well known and characterized and also appropriately monitored by effective AMPs at VVER plants so no additional research is considered as necessary. Note: Only applicable to CASS material if cast fittings are installed in small bore lines having susceptibility to high-cycle fatigue. **References:** *EPRI Materials Degradation Matrix, Revision 4*. (Notes v3-8a,b) EPRI, Palo Alto, CA: 2018. 3002013781. Guidelines for Integrity and Lifetime Assessment of Components and Piping in WWER Nuclear Power Plant (VERLIFE), 2014. NP 306.2.099-2004: “General requirements to lifetime extension of NPP power units for operation beyond the design period based on the results of periodic safety reviews,” State Committee for Nuclear Regulation of Ukraine. Kyiv, 2004. (in Ukrainian)NP 306.2.099-2004: General requirements to lifetime extension of NPP power units for operation beyond the design period based on the results of Periodic Safety Reviews. PL-D.0.03.126-10: “Regulation on the procedure for extending the life of equipment of systems important to safety,” State Committee for Nuclear Regulation of Ukraine, Kyiv, 2010. (in Russian)PL-D.0.03.126-10: Provisions for lifetime extension of the safety-related equipment. SOU NAEK 080:2014 Operation of Technological System. Long-Term Operation of NPP Units. General Provisions /6/SOU NAEK 080:2014: Technological complex. Long-term operation of NPP power units. General provisions. PM-D.0.03.222-14: “Standard program on ageing management of NPP structures and components,” NNEGC Energoatom, Kyiv, 2014. (in Russian)PM-D.0.03.222-14: Standard program on ageing management of NPP power unit structures and components. NP 306.2.210-2017: “General Requirements for Ageing Management of Components and Structures and Long-Term Operation of NPP Units,” State Committee for Nuclear Regulation of Ukraine. Kyiv, 2017. NP 306.2.210-2017: General Requirements for Ageing Management of NPP Components and Structures and Long Term Operation. IAEA-SSR-82, Rev. 1, “Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL),” International Atomic Energy Agency, Vienna, August 2020.IAEA IGALL AMPs – IAEA Vienna Update 2021: • AMP 101 Low Cycle Fatigue Monitoring • TLAA 106 Environmentally Assisted Fatigue • TLAA 119 High Cycle Thermal Fatigue	**Status:** Closed/No Gap **Priority:** R0 - Closed

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-AS-23 - Lack of appropriate methods for evaluating low-cycle fatigue of elbow pipes with welds under seismic loading **Issue:** There is no issue relevant to VVERs **Description:** The sites accommodating VVER units are of low-to medium seismicity areas where even the beyond design basis earthquakes are of magnitude less than Mw=7. The number of load cycles depends mainly on the magnitude, see e.g. Hancock, J., and Bommer, J.: The effective number of cycles of earthquake ground motion. Earthquake Engineering, 34 (2005) 637-664. Or see Wenqi Du, Xiaohui Yu, Gang Wang, "Prediction equations for the effective number of cycles of ground motions for shallow crustal earthquakes", Soil Dynamics and Earthquake Engineering 125 (2019) 105759 In contrast, under Paks site tectonic conditions, the maximum probable earthquake magnitude is approximately 6. The expected strong motion duration is slightly longer than 10 seconds, but anyway, not comparable with the strong motion duration of Great Tohoku Earthquake 11. 03. 2011. According to the above, impacts of larger seismic events on the low-cycle fatigue damage summation as it is considered for VVERs is of more theoretical interest than practical importance for existing plants. **References:** Wen, J. and Adams, T., "Technical Basis of a Code Case to Provide a Strain-Based Acceptance Limits for Service Level D Evaluation of Piping Systems under Section III of ASME Boiler and Pressure Vessel Code," Proceedings of the ASME 2019 Pressure Vessels and Piping Conference, PVP2019-93119 (2019). ASME Boiler and Pressure Vessel Code (BPVC), Sections III and XI, ASME New York, 2019. Hancock, J., and Bommer, J.: The effective number of cycles of earthquake ground motion. Earthquake Engineering, 34 (2005) 637-664. Wenqi, Du, Xiaohui, Yu, and Gang, Wang, Prediction equations for the effective number of cycles of ground motions for shallow crustal earthquakes, Soil Dynamics and Earthquake Engineering, 125 (2019) 105759.	**Status:** Close/No Gap **Priority:** R0 - Closed

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-AS-24 Lack of methods for evaluating non-proportional multiaxial low cycle fatigue of pipes under seismic loading **Issue:** There is no issue relevant to VVERs **Description:** The sites accommodating VVER units are of low-to medium seismicity areas where even the beyond design basis earthquakes are of magnitude less than Mw=7. The number of load cycles depends mainly on the magnitude, see e.g. Hancock, J., and Bommer, J.: The effective number of cycles of earthquake ground motion. Earthquake Engineering, 34 (2005) 637-664. Or Wenqi Du, Xiaohui Yu, Gang Wang, Prediction equations for the effective number of cycles of ground motions for shallow crustal earthquakes, Soil Dynamics and Earthquake Engineering 125 (2019) 105759 For example, under Paks site tectonic condition the maximum probable earthquake magnitude is approximately 6. The expected strong motion duration is slightly longer than 10 seconds, but anyway not comparable with the strong motion duration of Great Tohoku Earthquake 11. 03. 2011. According to this, the low-cycle fatigue as it is considered below can be more theoretical rather than practical issue for VVER. **References:** Wen, J. and Adams, T., “Technical Basis of a Code Case to Provide a Strain-Based Acceptance Limits for Service Level D Evaluation of Piping Systems under Section III of ASME Boiler and Pressure Vessel Code,” Proceedings of the ASME 2019 Pressure Vessels and Piping Conference, PVP2019-93119 (2019). ASME Boiler and Pressure Vessel Code (BPVC), Sections III and XI, ASME New York, 2019. Hancock, J., and Bommer, J.: The effective number of cycles of earthquake ground motion. Earthquake Engineering, 34 (2005) 637-664. Wenqi, Du, Xiaohui, Yu, and Gang, Wang, Prediction equations for the effective number of cycles of ground motions for shallow crustal earthquakes, Soil Dynamics and Earthquake Engineering, 125 (2019) 105759.	**Status:** Closed/No Gap **Priority:** R0 - Closed

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-AS-25 Lack of an advanced method for determining the number of fatigue cycles as part of the low cycle fatigue evaluation for piping under seismic loading **Issue:** There is no issue relevant to VVERs **Description:** The sites accommodating VVER units are of low-to medium seismicity areas where even the beyond design basis earthquakes are of magnitude less than Mw=7. The number of load cycles depends mainly on the magnitude, see e.g. Hancock, J., and Bommer, J.: The effective number of cycles of earthquake ground motion. Earthquake Engineering, 34 (2005) 637-664. Or Wenqi Du, Xiaohui Yu, Gang Wang, Prediction equations for the effective number of cycles of ground motions for shallow crustal earthquakes, Soil Dynamics and Earthquake Engineering 125 (2019) 105759 For example, under Paks site tectonic condition the maximum probable earthquake magnitude is approximately 6. The expected strong motion duration is slightly longer than 10 seconds, but anyway not comparable with the strong motion duration of Great Tohoku Earthquake 11. 03. 2011. According to this, the low-cycle fatigue as it is considered to be of more theoretical interest than of practical importance for VVERs. **References:** Wen, J. and Adams, T., “Technical Basis of a Code Case to Provide a Strain-Based Acceptance Limits for Service Level D Evaluation of Piping Systems under Section III of ASME Boiler and Pressure Vessel Code”, Proceedings of the ASME 2019 Pressure Vessels and Piping Conference, PVP2019-93119 (2019). ASME Boiler and Pressure Vessel Code (BPVC), Sections III and XI, ASME New York, 2019. Hancock, J., and Bommer, J.: The effective number of cycles of earthquake ground motion. Earthquake Engineering, 34 (2005) 637-664. Wenqi, Du, Xiaohui, Yu, and Gang, Wang, Prediction equations for the effective number of cycles of ground motions for shallow crustal earthquakes, Soil Dynamics and Earthquake Engineering, 125 (2019) 105759.	**Status:** Closed/No Gap **Priority:** R0 - Closed

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-IE-01 – NDE Technology for Detection and Characterization of Baffle and Former Assembly IASCC **Issue:** There is no issue relevant to VVER. **Description:** In the US UT and visual inspection methods are available for baffle bolt examination. The development of techniques is now the responsibility of vendors and the PWROG. PWROG WCAP 17096-NP-A which is an acceptance criteria document updated in 2016 provides methodologies to be implemented by vendors and was approved by NRC safety evaluation report (SER). Previously MRP-22 guidance specified EVT-1 visual examination of bolting in baffle plates. Subsequently UT methods were developed and validated. Operation experience of VVERs do not justify the degradation effect of IASCC to that extent which could lead to baffle bolt failure (material: AISI 321). At the same time the acceleration of the degradation process cannot be excluded during LTO. In the case of VVER reactors, full scope NDE of baffle bolts was performed only in the Finnish Loviisa NPP (VVER-440 model 213), twice so far. The first examinations (2006 and 2008 in unit 2 and unit 1, respectively) were rather preventive. Totally 9 bolts were replaced in the two units but only three of them had UT indications. The reason for the indications in two bolts was incomplete filling of the flat slot of the bolt head when the bolt was welded to the washer during installation. Only one bolt, having an accumulated dose of 2.9 dpa, had suffered from IASCC. Here the washer had been welded to the shielding plate unintentionally, which had resulted in high additional stresses during start-up and operation, and thus intensified the cracking process. This degradation was clearly triggered by manufacturing failure and as such it cannot be considered as same as had been found in PWRs. The second Loviisa inspections were performed in 2018 and 2020 the scheduling of which are in line with EPRI guidance. During the repeated inspections no defected bolts were found. No other VVER plants had baffle bolt inspections. Also defected or missing bolts were not reported in the VVERs. This is a gap in commercial implementation / application of existing technology rather than the technology per se being lacking. Some of the European ISI / NDE vendors are suitable to carry out complete NDE on VVERs both VT (EVT) and UT. The UT solutions apply both conventional and Phased Array and TOFD techniques. The technique can be fitted to the bolt construction. The capability of the NDE system is validated; for the Finnish plant it was done according to the European inspection qualification methodology. The remote-controlled manipulators used in PWRs are applicable for the VVERs. **References**: Ehrnstén, U., Pakarinena, J., Karlsen, W., Kei, H.,: "Investigations on core basket bolts from a VVER 440 power plant," in Engineering Failure Analysis, October 2013. Regidor, J.J. et al, Inspection and Replacement of Baffle Former Bolts in VVER-440 Reactor Type, Proc. 6th International Conference on NDE in Relation to Structural Integrity for Nuclear and Pressurized Components (ed. M. Bieth), Budapest, Hungary, pp. 926-929, 2007. ENIQ Methodology Document No. 61, "European methodology for qualification of non-destructive testing – Issue 4," Brussels, March 2019. IAEA-SSR-82, Rev. 1, "Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL)," International Atomic Energy Agency, Vienna, August 2020. • AMP 113 PWR Reactor Pressure Vessel Internals *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines MRP-227-A.*EPRI, Palo Alto, CA: (2011). 1022863. *Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228)*. EPRI, Palo Alto, CA: (2009). 1016609.	**Status:** Closed/No Gap **Priority:** R0 - Closed

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-IE-05 – NDE Qualification for Reactor Internals Inspection (VT Evaluation)	**Status:** Closed/No Gap **Priority:** R0 - Closed
Issue: NDE procedures for inspection of reactor internals must be demonstrated to have reliability and reproducibility of performance in order to be applied. While physical viability of inspection methods has been demonstrated, implementation by vendors and, hence, availability to plant operators, is lacking in some situations. **Description**: Implementation of MRP-227 requires remote visual inspection of reactor internals using VT-1 and / or EVT-1. EVT-1 makes use of high-resolution camera systems to routinely examine nuclear plant internals components. Techniques and methods for performing EVT-1 examinations were developed in the field at the time of the first core shroud examinations (1993-1995 time frame). As a result, EVT-1 did not initially have a formal laboratory-derived technical basis. The methods and techniques evolved as lessons were learned about the necessary camera distance, lighting, and surface-condition requirements. During the course of the past few years, increased attention on remote EVT-1 capability has occurred as a result of concerns expressed by U.S. NRC. In response, EPRI initiated a multi-year R&D program which resulted in some improvements to EVT-1 requirements, including resolution, angle, and scan speed requirements. The studies also resulted in recommendations regarding examiner qualification and equipment selection. Based on this work and the associated improvements made to EVT-1 requirements, EPRI believes that the significant technical questions regarding EVT-1 reliability have been resolved. In particular, the current remote visual examination practices for vessel internals inspections were found to be capable of detecting all but the smallest of flaws. However, because of the persisting concerns expressed by some regulatory bodies, this gap remains open. The EPRI NDE Program has ongoing projects related to enhancing the capability of EVT-1 examinations. The MRP will continue to evaluate the results of these NDE Program projects. The VVER inspection practice does not make an explicit differentiation between VT-1 and EVT-1. It can be stated that majority of the routine visual inspections on reactor internals components are VT-1. Inspections are normally performed by the plant personnel with own inspection equipment but in some cases done by ISI vendors available on the market. Qualification of remote controlled visual inspections (according to ENIQ methodology) is exceptional. There is therefore no technical issue relevant to VVERs. VVER specific research cannot be identified and EPRI research deliverables can be used for the VVER reactors, too. **References**: Tsvetkov, E., Heinsius, J., ENIQ-Qualified Visual Examinations by Means of Remote Controlled Submarine, atw, Vol. 60 (2015) pp. 452-456. ENIQ Methodology Document No. 61, “European methodology for qualification of non-destructive testing – Issue 4,” Brussels, March 2019. IAEA-SSR-82, Rev. 1, “Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL),” International Atomic Energy Agency, Vienna, August 2020.	

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-MT-06 - Develop Recommendations and Guidance for Mitigating Fatigue Failures in Piping **Issue:** There is no issue relevant to VVERs **Description:** All VVERs were originally designed according to the requirements of the Russian normative documents (e.g. PNAE G7-002). The design included the fatigue life calculation for the piping, too. At Paks NPP the design basis had been changed to ASME BPVC Section III after 30 years operation as part of the technical justification of the LTO (e.g. metal fatigue TLAA compliance with HAEA 3.25 Guide & MSZ 27003 Hungarian national standard). At some other VVERs (e.g. Bohunice and Dukovany NPPs) the LTO is also based on metal fatigue TLAA, where the Czech codes (STD A.M.E. 4201-86 ÷ 4214-86) and VERLIFE recommendations were applied. All VVER TLAA analyzes comply with IGALL TLAA 101 document (low-cycle fatigue usage), TLAA119 (High-Cycle Fatigue) and TLAA106 (Environmentally-Assisted Fatigue) programs The monitoring and control of aging effects due to fatigue at piping is obligatory at VVER plants and that is based on national regulations. At all VVERs the consequence of the fatigue (e.g. crack appearance) is consequently monitored by appropriate ISI program (surface and volumetric examinations, etc.) according to the national codes and standards. Also at some VVER countries (e.g. Finland, Ukraine, Russia, Bulgaria, Slovakia) an online fatigue monitoring system (FAMOSi) is applied. All VVER operators developed their own AMPs for fatigue monitoring which are comply with IGALL AMP 101, 161 programs. These programs are continuously updated according to the VVER OPEX and references. At some VVERs there were cracks found due to fatigue at the Steam Generator emergency feedwater piping nozzle. The detected cracks demonstrate the effectiveness of the fatigue monitoring applied (discovered timely). Note that the nozzle belongs to the Steam Generator (not to the piping) so this type of failure does not part of the piping issues. Vibrational fatigue failures in small-bore piping and thermal fatigue failures due to thermal stratification, cycling, and striping (TASCS) represent the most common fatigue-related problems appearing in PWR power plant equipment (see MRP-235R2, MRP-408, MRP-409). The MRP-235R2 is a generic handbook which can be applied for PWR and VVER plants too, for helping utility engineers at operating plants understand and manage fatigue issues. The fatigue life of piping is well characterized and monitored by appropriate AMPs and fatigue monitoring systems (e.g. ISI and FAMOSi, FATI) at VVER plants and the available EPRI research deliverables are applicable to VVER, too, so no additional research is considered necessary. **References:** *EPRI Materials Degradation Matrix, Revision 4.* (Notes p1-8c, p1-8d, p1-8e, v3-8a,b), EPRI, Palo Alto, CA: (2018). 3002013781. *Materials Reliability Program: Management of Thermal Fatigue n Normally Stagnant Non-Isolable Reactor Coolant Systems Branch Lines (MRP-146, Revision 2).* EPRI, Palo Alto, CA: (2016). 3002007853. *Materials Reliability Program: Assessment of Residual Heat Removal Mixing Tee Thermal Fatigue in PWR Plants (MRP-192, Revision 3)*. EPRI, Palo Alto, CA: (2018). 3002013266. *Materials Reliability Program: Fatigue Management Handbook (MRP-235, Revision 2).* EPRI, Palo Alto, CA: (2009). 3002005510.	**Status:** Closed/No Gap **Priority:** Closed

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-MT-06 - Develop Recommendations and Guidance for Mitigating Fatigue Failures in Piping (continued) *Materials Reliability Program: Summary of JSME Thermal Fatigue Assessment Guideline and Comparison with MRP Management Guideline (MRP-408).* EPRI, Palo Alto, CA: (2016). 3002007851. *Materials Reliability Program: EdF Assessment of US Thermal Fatigue Management and Operating Experience and Development of Recommendations for Guideline Improvement (MRP-409)*. EPRI, Palo Alto, CA: (2016). 3002007852. Guidelines for Integrity and Lifetime Assessment of Components and Piping in WWER Nuclear Power Plant (VERLIFE), 2014. IAEA-SSR-82, Rev. 1, “Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL),” International Atomic Energy Agency, Vienna, August 2020. • AMP 101 Low Cycle Fatigue Monitoring • TLAA 101, Low Cycle Fatigue Usage • TLAA 106 Environmentally Assisted Fatigue • TLAA 119 High Cycle Thermal Fatigue	**Status:** Closed/No Gap **Priority:** Closed

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-RR-02 – Repair / Replacement for Thermal Fatigue of Piping The workshop identified that this is more appropriate as an AS GAP – **"Methods for Determining Locations of Class 1 Piping Susceptible to Thermal Fatigue (and Potential for Repair and Replacement) – The technical concept is now included in Gap V-AS-28** **Issue:** There is a need for comparison and evaluation of methods used by individual operators to assess the piping segments affected by thermal fatigue as well as guidance for repair or replacement **Description:** There are some cases of thermal fatigue degradation at VVER plants; they are limited to locations where thin wall small bore piping is connected to thick walled components. As these locations were evaluated as high risk, they were examined by destructive testing and the small fatigue cracks were detected. No thermal fatigue damage has been detected on Class 1 large bore piping at VVER plants to date. In some cases, regulators requested complete evaluation of Class 1 piping for thermal fatigue as a condition for license renewal. Therefore, this work has high priority and may be of interest also for other operators of VVER plants. Due to high similarity of VVER design it is expected that similar approaches for determination of susceptible locations and monitoring tools may be used. One approach for determination of susceptible locations is based on the screening criteria. Only the piping locations with higher temperatures and fast temperature changes are taken into consideration. Data from measurement of temperature changes and operational conditions are then evaluated and modifications of the measurement locations are recommended in order to obtain more data [Samohyl 2019]. General position of the industry [NEA conference] state that "…there is the need to recognize the stratification locations, where thermal fatigue could be critical. Improvements in fatigue monitoring and knowledge of sensor locations should be developed based on more detailed assessment of the operating experience…". It should be added that the attention must not be limited to the locations where thermal stratification occurs but also to other locations with fast temperature changes. EPRI guidance should identify possible approaches for determination of susceptible locations, recommended methods for field measurements as well as possible repair and replacement procedures, including weld overlay, installation of additional insulations and other viable solutions. Effective technical approaches to justify weld overlay (WOL) repair solutions are needed. Effects of the WOL and the processes used to deposit the WOL on adjacent base material need to be taken into consideration. **References:** Samohyl, P., et al: Procedure for evaluation of the current heat measurements for determination of critical piping locations susceptible to thermal fatigue at NPP Dukovany, UJV Report DITI 2301/934, 2019. NTD-ASI, Section III; Normative-technical documentation, Association of Czech Mechanical Engineers, 2020 UJV DITI 2302/584.	**Status:** Closed/No Gap **Priority:** Closed

Table B-2 (continued)
Non Relevant/Closed Gaps (from PWR Gap Listing) – Gaps and Closure Bases

V-RR-05 – Alternative DM Weld Repair Solutions Recognize that for VVERs this concept is covered by **V-RR-06 – Validation of Nickel Plating DMW Surfaces to Protect against SCC in Secondary Water and V-MT-05 – Mitigation of SCC in Nozzle DM Welds – To avoid Stainless Steel Cracking Propagating into LAS Pressure Boundary Materials**	**Status:** Closed/No Gap **Priority:** Closed

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