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Materials Reliability Program: Electric Power Research Institute (EPRI) Review of the Japanese Nuclear Operators' (JNOs') Aging Management Plan for Prolonged Shutdown Periods (MRP-435)

2018 TECHNICAL REPORT

Materials Reliability Program: Electric Power Research Institute (EPRI) Review of the Japanese Nuclear Operators' (JNOs') Aging Management Plan for Prolonged Shutdown Periods (MRP-435)

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Final Report, November 2018

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ABSTRACT

Nuclear power plant (NPP) operating periods are limited to 60 years in Japan. Most of the Japanese NPPs experienced, or are still experiencing, prolonged plant shutdown periods after the Great Japan Earthquake of 2011. These prolonged shutdown periods have significantly reduced the remaining licensed actual operating times for the NPPs. As a result, the eleven Japanese Nuclear Operators (JNOs) have completed technical assessments of aging-related degradation for NPP infrastructure originating from the long-term shutdown period. These assessments are documented in a Technical Report ("JNO Report") that justifies NPP operating life can be determined based on the total actual NPP operating years. The JNO Report asserts that a long-term shutdown period (i.e., approximately 10 calendar years) would not challenge the plant operating life from a technological viewpoint. The JNO Report asserts that the effect of the long-term shutdown on NPP aging degradation is negligible.

The JNOs requested the Electric Power Research Institute (EPRI) provide an independent review of their technical assessment report. EPRI, as an independent and nonprofit research institute, is well positioned to conduct this review. EPRI's staff has a depth of knowledge and experience in the technical bases for extended operation (to 60 calendar years), and subsequent license extension (to 80 total years). As such, EPRI's research and guidance supports the extended operation of other plants worldwide. EPRI has also been involved with developing many of the current standards and guidance accepted for use by the United States (U.S.) Nuclear Regulatory Commission (NRC) on plants that have applied for license extension. Many of the same NRC standards and guidance developed for the license extension process can be analogously utilized by a NPP seeking to justify recovery of operating time after a prolonged shutdown period.

The objectives of EPRI's review were to:

- Evaluate the technical assessments of aging-related degradation performed by the JNOs,
- Assess the overall concept that most of the aging phenomena are manageable by routine maintenance activities and component functions can be assured by replacement and other methods, as appropriate,
- Judge the logic that the four specific degradation phenomena will not progress during long-term shutdown, and
- Further judge whether electrical and instrument component insulation degradation, along with concrete component strength degradation, are minor and that sufficient margins are maintained.

This report documents the results of EPRI's review and includes recommendations for improvement of the JNO Report, which the JNOs can use to enhance their aging management programs for both prolonged shutdown period recovery and subsequently, long-term operation for 60 years.

Keywords
Aging Management
Low cycle fatigue
Reactor pressure vessel (RPV) integrity
Reactor vessel internals (RVI) aging management
Environmentally qualified (EQ) cables
Concrete structure degradation

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EXECUTIVE SUMMARY

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Product Type: Technical Report

Product Title: Materials Reliability Program: Electric Power Research Institute (EPRI) Review of the Japanese Nuclear Operators' (JNOs') Aging Management Plan for Prolonged Shutdown Periods (MRP-435)

PRIMARY AUDIENCE: The eleven Japanese Nuclear Operators (JNOs) considering recovery of the long-term nuclear power plants (NPPs) shutdown period.

SECONDARY AUDIENCE: Members of the public interested in the recovery of the long-term shutdown period for the Japanese NPPs.

KEY RESEARCH QUESTION

NPP operating periods are limited to 60 years in Japan. Most of the Japanese NPPs experienced, or are still experiencing, prolonged plant shutdown periods after the Great Japan Earthquake of 2011. These prolonged shutdown periods have significantly reduced the remaining licensed actual operating times for the NPPs. As a result, the eleven JNOs have completed technical assessments of aging-related degradation for NPP infrastructure originating from the long-term shutdown period. These assessments are in a Technical Report ("JNO Report") that justifies NPP operating life can be determined based on the total actual NPP operating years. The JNO Report asserts that a long-term shutdown period (i.e., approximately 10 calendar years) would not challenge the plant operating life from a technological viewpoint. The JNO Report asserts that the effect of the long-term shutdown on NPP aging degradation is negligible. The JNOs requested the Electric Power Research Institute (EPRI) provide an independent review of their technical assessment report and its conclusions. This report documents the results of EPRI's review.

RESEARCH OVERVIEW

EPRI performed detailed technical reviews of the aging-related degradation assessment completed in several technical areas for potential recovery of the long-term shutdown period by the JNOs. The objectives of EPRI's review were to evaluate the technical assessments of aging-related degradation performed by the JNOs; to assess the overall concept that most of the aging phenomena are manageable by routine maintenance activities, and component functions can be assured by replacement and other methods, as appropriate; to judge the logic that the four specific degradation phenomena will not progress during long-term shutdown; and finally, to further judge whether electrical and instrument component insulation degradation, along with concrete component strength degradation, are minor and that sufficient margins are maintained.

KEY FINDINGS

Overall, based on the review of the technical topics, EPRI did not identify any significant gaps or challenges that would preclude the JNO fleet from successfully recovering the prolonged shutdown period from a technical or a safety standpoint.

- Review of Overall Approach to Technical Evaluation of Aging Degradation
- Review of Aging Phenomena under Routine Maintenance Control

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EXECUTIVE SUMMARY

- Review of Four Specific Degradation Phenomena
    - Low Cycle Fatigue of Class 1 Components
    - Neutron Irradiation Embrittlement
    - Irradiation Assisted Stress Corrosion Cracking
    - Thermal Aging of Cast Austenitic Stainless Steels
- Review of Degradation of Electrical Instrumentation Equipment
- Review of Degradation of Concrete

WHY THIS MATTERS

This report documents the results of EPRI's review and includes recommendations for improvement of the JNO Report, which the JNOs can use to enhance their aging management programs for both prolonged shutdown period recovery and subsequently, long-term operation for 60 years.

HOW TO APPLY RESULTS

This report may be used by the JNOs to evaluate aging management and long-term operation plans for both their boiling water reactor (BWR) and pressurized water reactor (PWR) fleets selected for prolonged shutdown period recovery and operation for 60 years.

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PROGRAM: Materials Reliability Program 41.01.04 and Boiling Water Reactor Vessel and Internals Project 41.01.03

IMPLEMENTATION CATEGORY: Reference

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INTRODUCTION, REPORT OBJECTIVES AND SCOPE

Nuclear power plant (NPP) operating periods are limited to 60 years in Japan. Most of the Japanese NPPs experienced, or are still experiencing, prolonged plant shutdown periods after the Great Japan Earthquake of 2011. These prolonged shutdown periods have significantly reduced the remaining licensed operating times for the NPPs. As a result, the eleven Japanese Nuclear Operators (JNOs) have completed technical assessments of aging-related degradation for NPP infrastructure originating from the long-term shutdown period. These assessments are in a Technical Report ("JNO Report") that justifies NPP operating life can be determined based on the total actual NPP operating years. The JNO Report asserts that a long-term shutdown period (i.e., approximately 10 calendar years) would not challenge the plant operating life from a technological viewpoint. The JNO Report asserts that the effect of the long-term shutdown on NPP aging degradation is negligible.

The JNOs requested the Electric Power Research Institute (EPRI) provide an independent review of their technical assessment report. EPRI, as an independent and nonprofit research institute, is well positioned to conduct this review. EPRI's staff has a depth of knowledge and experience in the technical bases for extended operation (to 60 calendar years), and subsequent license extension (to 80 total years). As such, EPRI's research and guidance supports the extended operation of other plants worldwide. EPRI has also been involved with developing many of the current standards and guidance accepted for use by the United States (U.S.) Nuclear Regulatory Commission (NRC) on plants that have applied for license extension. Many of the same NRC standards and guidance developed for the license extension process can be analogously utilized by a NPP seeking to justify recovery of operating time after a prolonged shutdown period.

The objectives of EPRI's review were to:

- Evaluate the technical assessments of aging-related degradation performed by the JNOs,
- Assess the overall concept that most of the aging phenomena are manageable by routine maintenance activities and component functions can be assured by replacement and other methods, as appropriate,
- Judge the logic that the four specific degradation phenomena will not progress during long-term shutdown, and
- Further judge whether electrical and instrument component insulation degradation, along with concrete component strength degradation, are minor and that sufficient margins are maintained.

In the U.S., NPPs utilize the Standard Review Plan for License Renewal (SRP-LR), NUREG-1800, Revision 2 [1-1], and the Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Revision 2 [1-2], for generic regulatory guidance when applying for license extension to 60 years. Similarly, for a potential subsequent license renewal (SLR) to 80 years of plant operation, NUREG-2191 [1-3] details the GALL for SLR and NUREG-2192 [1-4] provides the SRP information.

These documents along with EPRI's technical reports were the primary bases for EPRI's technical assessments in this report. Specifically, EPRI made comparisons of the operating period extension methods, which are directly analogous to the recovery of a prolonged shutdown period used by the JNOs, to the methodology of license extension in the U.S. to ascertain if any gaps exist and to identify areas where the JNO approach may be non-conservative or in need of improvement, when compared to U.S. evaluations.

1.1 Report Objective

The scope of this study has several objectives:

- Review the age-related degradation technical assessments performed by the JNOs for the Japanese NPPs.
- Assess the overall concept that most of the aging phenomena are manageable by routine maintenance activities and component functions can be assured by replacement and other methods, as appropriate.
- Judge the logic that the four specific degradation phenomena will not progress during long-term shutdown, and further judge whether electrical and instrument component insulation degradation, along with concrete component strength degradation, are minor and that sufficient margins are maintained.
- Identify conservativisms between the Japanese technical approach and the approved methodologies of the NRC for license extension in the U.S., which are directly analogous to the recovery of a prolonged shutdown period.
- Identify any recommendations for improvement in the JNO technical assessments, and any significant differences and potential non-conservatisms between the Japanese approach and the approved methodologies used in the U.S.

1.2 Report Scope

This report documents the review of age-related degradation assessments of the JNO Report in the following technical areas:

Section 2: Review of Overall Approach to Technical Evaluation of Aging Degradation

Section 3: Review of Aging Phenomena under Routine Maintenance Control

Section 4: Review of Four Specific Degradation Phenomena

4-1. Low Cycle Fatigue of Class 1 Components

4-2. Neutron Irradiation Embrittlement

4-3. Irradiation Assisted Stress Corrosion Cracking

4-4. Thermal Aging of Cast Austenitic Stainless Steels

Section 5: Review of Degradation of Electrical Instrumentation Equipment

Section 6: Review of Degradation of Concrete

Additional detailed descriptions of the methodologies used, evaluations performed, and assessments made in the analysis of each respective technical area is included in the individual sections.

1.3 References

1-1. NUREG-1800, Revision 2, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," December 2010.

1-2. NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," December 2010.

1-3. NUREG-2192, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants," July 2017.

1-4. NUREG-2191, Volumes 1 and 2, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," July 2017.

2 REVIEW OF OVERALL APPROACH TO TECHNICAL EVALUATION OF AGING DEGRADATION

The JNO Report [2-1] evaluates whether some aging phenomena can be addressed by routine management using appropriate monitoring and inspection methods during the long-term shutdown period. For this situation, the JNO Report asserts that component functions can be recovered or assured by maintenance activities, such as repair and replacement, as necessary. Furthermore, the JNO Report describes that aging effects originating from the long-term shutdown period will not challenge the plant integrity since appropriate management is implemented, according to a special maintenance plan, during the long-term shutdown period.

As a precursor to more in-depth review, EPRI reviewed the overall approach of the JNO Report with regard to the following:

Evaluation of the potential aging phenomena that may occur in systems, structures, and components, as follows:

- Is the aging phenomenon manageable by routine maintenance activities?
- Is progression of degradation during a long-term shutdown period negligible?
- Is the aging phenomenon manageable by detailed evaluation considering the effects of long-term shutdown period?

2.1 Discussion

Based on a review of the JNO Report, EPRI generally agrees that the technical evaluation process used in the report is comprehensive. The known relevant aging mechanisms, analyses and operating experience are assessed. The review of existing maintenance programs, including inspection and monitoring for their adequacy in detecting relevant aging effects appears appropriate. Finally, the identified changes to plant maintenance plans, which account for the long-term shutdown period to ensure all relevant aging is detected or monitored, seems proper.

The JNO Report indicates that the Japanese utilities will modify their maintenance practices to address specific aging effects based on the results of the technical evaluations to assure proper management of aging effects. EPRI agrees that the approach taken by the JNOs represents an acceptable process to assess the potential age-related degradation mechanisms in NPPs during and after a long-term shutdown. The commitment to a continuous improvement process regarding the aging management practices and programs is proper, as this will assure the aging management program remains effective in managing aging issues and maintaining plant safety for 60 years of operation.

2.2 Searches of NRC ADAMS

EPRI performed a thorough search of the NRC's ADAMS electronic filing system for U.S. NPP submittals related to long-term shutdown period recovery. Historically, recapture of license time was an issue of interest for U.S. plants in the 1980s and 1990s that eventually led the NRC to release SECY-98-296 [2-2] that established the NRC's policy toward recapture. In addition, the restart of the Browns Ferry Unit 1 plant in the U.S. was considered relevant to EPRI's review since the plant was restarted after more than 20 years spent in shutdown[1]. EPRI's objective in searching ADAMS was to locate U.S. license recapture submittals and review any related technical matters that were addressed as part of U.S. submittals for comparison against the technical topics addressed in the JNO Report. EPRI concluded, after a review of all retrieved ADAMS documents, that no new technical topics were identified that are not already addressed by JNOs' planned programs.

In 1998, NRC staff developed a policy recommendation regarding recovery of operating periods other than periods of full power operation for NRC Commissioner approval. This was documented in SECY-98-296 [2-2]. The final policy approved by the Commission was documented in a Staff Requirements Memorandum (SRM) [2-3]. In the SRM, the Commission established a policy allowing plants to recover time spent in low-power testing before receiving a Full Power Operating License. It also agreed to continue to grant license amendment requests to recover time spent in construction in cases where the original 40-year license term began with the construction permit date. Subsequent to the issuance of these two NRC documents [2-2 and 2-3], three additional plant sites applied for extension to their operating license expiration dates. Those applications, which were for Palo Verde Units 1, 2 and 3, Diablo Canyon Units 1 and 2, and Seabrook Unit 1, were granted. Therefore, extension of U.S. NPP operating licenses have been granted by the NRC on a case-by-case basis with adequate justification consistent with the SRM.

As noted above, the restart of the Browns Ferry Unit 1 plant in the U.S. was considered relevant to EPRI's review since the plant was restarted after more than 20 years spent in shutdown. A review of the documents associated with the restart activities provides insight into the type and breadth of issues to consider in restarting a plant after a long-term shutdown. With regard to the JNO Report scope, review of the Browns Ferry documents did not identify any new technical issues that should be addressed. However, it is worth noting that there are numerous situations where components were replaced prior to the restart which is consistent with the JNO Report assertion that component functions can be recovered or assured by maintenance activities, such as repair and replacement, as necessary.

[1] Browns Ferry Unit 1 originally began low power testing in December 1973, and was licensed to operate through December 2013. However, the unit was shutdown for approximately a year after a cabling fire in March 1975 damaged the unit. The unit was subsequently repaired and returned to operation in 1976. It operated until March 1985, when all three Browns Ferry units were shutdown for operational and management issues. Unit 1 remained shutdown until May 2007, when it returned to operation after more than 20 years based on a significant restart effort to restore the unit to operational status. For the purposes of EPRI's review of the JNO Report, this lengthy plant shutdown was reviewed for possible technical issues that might be relevant to the JNO long-term shutdown period recapture effort.

2.3 Conclusions

EPRI agrees with the overall approach taken by the JNOs. The relevant aging and degradation mechanisms have been evaluated and the proper management processes put in place. The identified changes to plant maintenance plans, which account for the long-term shutdown period, to ensure all relevant aging is detected or monitored seems proper. As such, it is concluded there are adequate aging management practices and programs in place to effectively manage aging issues and maintain plant safety for 60 years of actual operation, in addition to the period of long-term shutdown. EPRI's review of each of the specific aging management aspects of the JNO Report is provided in the following sections.

2.4 References

2-1. Hokkaido Electric Power Co., Inc., et al., "Technical Report Regarding the Effects of Nuclear Power Plant Operating Period on Aging Degradation of Major Components/Structures," English Version, July 2018.

2-2. SECY-98-296, "Agency Policy Regarding Licensee Recapture of Low-Power Testing or Shutdown Time for Nuclear Power Plants," December 21, 1998.

2-3. Staff Requirements Memorandum, "SECY-98-296 - Agency Policy Regarding Licensee Recapture of Low-Power Testing or Shutdown time for Nuclear Power Plants," March 30, 1999.

3 REVIEW OF AGING PHENOMENA UNDER ROUTINE MAINTENANCE CONTROL

The JNO Report [3-1] states that some aging phenomena result in negligible aging effects during a long-term shutdown period because the degradation phenomena are manageable by routine maintenance. In such cases, the JNO Report asserts that degradation trends can be understood with the help of appropriate monitoring and inspection techniques, and that functions can be restored by carrying out repairs and replacements, as required. Furthermore, the JNO Report indicates that all forms of corrosion degradation phenomena, such as flow accelerated corrosion (FAC)/droplet impingement, stress corrosion cracking (SCC), etc., are also manageable by routine maintenance.

3.1 Discussion

EPRI generally agrees with the JNO Report conclusion that all forms of corrosion of vessels, reactor internals and piping can be adequately managed during long-term plant shutdown by routine maintenance and inspection activities. One item of note that EPRI identified is related to the severe corrosion experienced in the residual heat removal (RHR) service water and raw cooling water systems at the Brown Ferry Unit 1 plant in the U.S. after its prolonged (~20-year) shutdown. Section 3.1.3 of Reference [3-2] reviews this issue and identifies that Browns Ferry 1 had to replace the piping before startup. This serves as a key example where routine maintenance control replacement activities managed an aging phenomenon after long-term shutdown.

As described in the JNO Report, the Japanese Nuclear Industry has worked diligently to safely operate and maintain their power plants. The JNOs have constantly strived to improve their aging management methods and guidance based on updated science and operating experience. With the change in regulations in 1996, plants were required to perform aging management assuming the plants would operate for 60 years. Effective as of 2008, the regulations were revised to require "that aging management practices to be included in the plant's normal maintenance management activities and long-term maintenance management policies based on the result of aging management technical evaluation shall be approved according to the safety regulations for reactor facilities." During the same period the Atomic Energy Society of Japan (AESJ) developed the "Code on Implementation and Review of Nuclear Power Plant Aging Management Program" (AESJ-SC-P005E:2008) [3-3]. (Note the 2008 version with the 2012 Supplement is the latest version in English). The AESJ document provided the standard for how an owner would implement aging management measures. The AESJ document is thorough in its definition of scope, screening of aging phenomenon and aging factors, components to be evaluated, the evaluation processes and how issues should be managed.

EPRI has developed tools to assist members in defining and managing materials aging and degradation. The Materials Degradation Matrix (MDM) [3-4] has a compilation of the materials used in the primary pressure boundary systems and components, and identifies the modes of degradation that can affect the components. The report includes not only the degradation mechanisms but a high-level discussion of potential means to manage or mitigate the mechanisms.

The JNO Report shows that management of potential degradation and aging continues during shutdown periods. Figure 3-1 (Figure 3.1.1-1 from [3-1]) below shows the maintenance cycle for periods of operation and during shutdown. During shutdown, preventative maintenance continues that includes system walkdowns, routine periodic inspections (including component disassembly for interior examination), proactive replacements and plant modifications where necessary. The JNO Report notes multiple locations in boiling water reactors (BWRs) and pressurized water reactors (PWRs) where components and entire systems have been proactively replaced (Figures 3.1.1-3 through 3.1.1-5 [3-1].

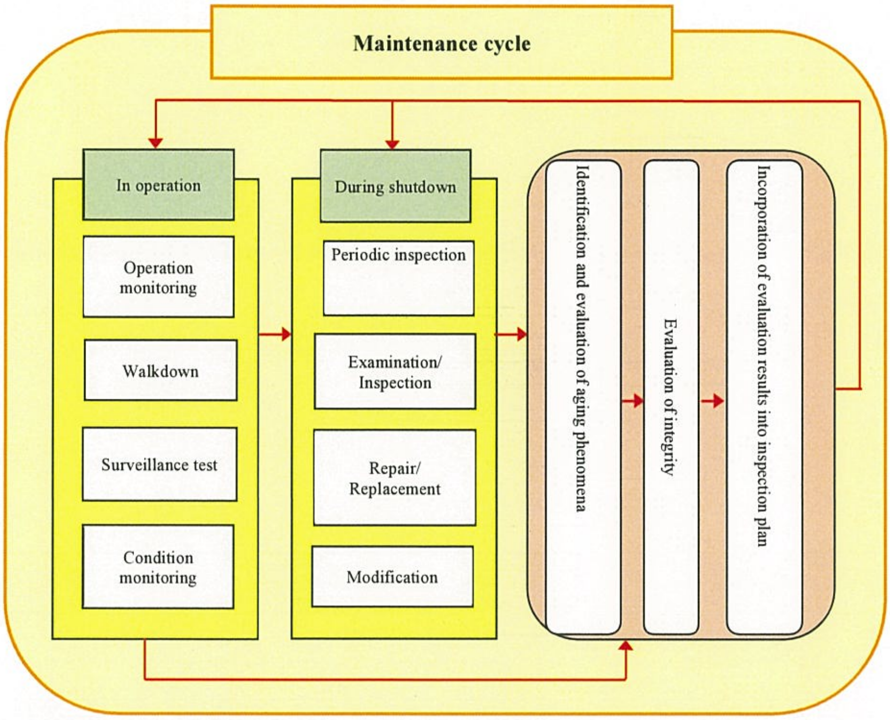


Figure 3-1
System of Maintenance Management Activities at a NPP

EPRI reviewed the definitions of general corrosion and the various forms of local corrosion in the JNO Report and agrees with them. The materials impacted by corrosion are identified, as are the various processes by which corrosion may occur. The differences between general corrosion and local corrosion and how they can work independently or together is clearly described. This information provides the necessary insights to choose mitigation methods that preclude or lessen degradation as well as the proper maintenance and inspection practices to manage aging during operation and the long-term shutdown.

The JNO Report identified the combination of measures that can be used to manage general and local corrosion. These include selection of the proper material, surface protection, and water chemistry controls, all in combination with the appropriate inspections. Also, where needed, preventative maintenance and replacement will be utilized. The report also describes additional measures taken during the extended shutdown including proper wet and dry layup processes, inspections, chemistry controls, and repairs such as re-application of protective coatings.

EPRI also reviewed the JNO Report pipe wall thinning bases and management. The report identifies FAC and liquid drop impingement (LDI) erosion as major contributors to pipe wall thinning and the mechanisms by which they work. The report identifies the variables that impact the rate of degradation such as fluid state, fluid velocity, piping configurations, etc.

The JNO Report defined the methods for use in managing pipe wall thinning. This involves control of water chemistry, routine pipe wall thickness measurements and, in some cases, replacement of piping systems with better materials or system reconfiguration to reduce susceptibility to degradation. The report also accurately notes that the wall thinning is dependent on fluid flow. As such, piping and tubing with no fluid flow during the long-term shutdown would not experience the wall thinning.

Finally, EPRI also reviewed the JNO Report's discussions on SCC. Its discussion is thorough. The detailed discussion is included Section 4.3 of this report. However, a synopsis is included here for completeness.

Section 3.2.2.1(3) of the JNO Report [3-1] asserts that long-term plant shutdown has no significant effects on the initiation and growth of SCC for the reactor coolant pressure boundaries and core internals when temperature decreases during plant shutdown below 100°C and when the reactor is not critical such that a neutron irradiation flux is not present. The JNO Report concludes that the possibility a new SCC initiation in a cooling water environment below 100°C during a shutdown period and the growth rate of existing SCC cracks is sufficiently lower than that during typical operation (288°C or 550°F) and, therefore, such SCC can be dealt with by normal routine inspection after the resumption of plant operation. The JNO Report also notes that there is some risk of SCC due to unusual chloride-containing atmospheres such that chloride-induced SCC could occur.

3.2 Conclusions

EPRI agrees with the JNO Report conclusions that corrosion, wall thinning and SCC can be adequately managed during long-term plant shutdown by routine maintenance, including component and system replacements and inspection activities. Furthermore, during the long-term shutdown, systems or portions of systems with no fluid flow will not experience wall thinning. As such, these issues do not represent a technical impediment to the recovery of the long-term shutdown period by the JNOs.

3.3 References

3-1. Hokkaido Electric Power Co., Inc., et al., "Technical Report Regarding the Effects of Nuclear Power Plant Operating Period on Aging Degradation of Major Components/Structures," English Version, July 2018.

3-2. EPRI Technical White Paper 2016-05, "Industry Operating Experience Related to Long Term Layup of BWRs, Support for Chubu Electric Hamaoka Unit 4," 2016.

3-3. "The Code on Implementation and Review of Nuclear Power Plant Ageing Management Program: 2015," AESJ-SC-P005:20150, March 2016.

3-4. *EPRI Materials Degradation Matrix, Revision 4.* EPRI, Palo Alto, CA: 2018. 3002013781.

4

REVIEW OF FOUR SPECIFIC DEGRADATION PHENOMENA

A summary of the overall technical approach of the JNO Report [4.1-1] and its conclusion are shown in Figure 4-1. As indicated in the red rectangle in Figure 4-1, the JNO Report identifies four specific aging phenomena that may be addressed by actions associated with aging management. Specifically, these four phenomena are:

- Low Cycle Fatigue (LCF) of Class 1 Components - Section 4.1
- Neutron Irradiation Embrittlement - Section 4.2
- Irradiation Assisted Stress Corrosion Cracking - Section 4.3
- Thermal Aging of Cast Stainless Steel - Section 4.4

EPRI reviewed the approach, logic, and associated conclusions in the JNO Report for these four specific degradation phenomena. EPRI's review of each of these specific areas and the associated conclusions that these phenomena will not progress significantly during long-term shutdown are detailed in the following subsections.

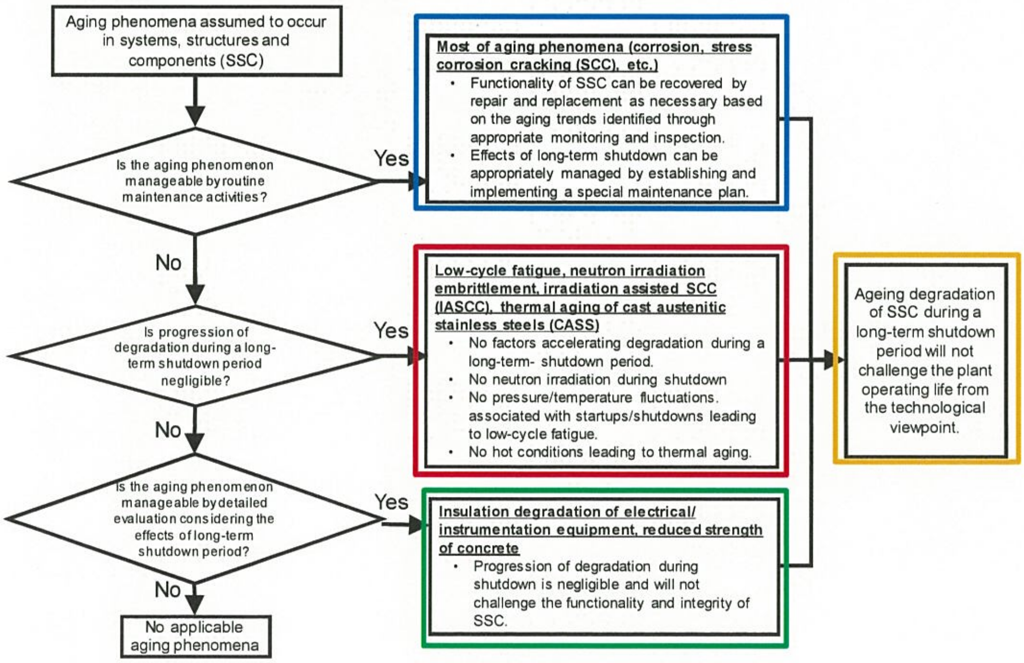


Figure 4-1
Overall Summary of the JNO Report Approach and Conclusions

4.1 LCF of Class 1 Components

The JNO Report discusses five topics related to LCF, as follows:

1. What is LCF?
2. Factors contributing to LCF;
3. Actions to deal with LCF;
4. Examples of LCF evaluations for actual units; and
5. Effects of long-term plant shutdown on LCF.

EPRI reviewed JNOs' overall approach described for these five topics. The details of EPRI's review the conclusions of the review are contained in the following subsections.

4.1.1 What is LCF?

The JNO Report identifies fatigue as a loading that is not large enough to cause failure of the material through one cycle of application, but causes local and small deformation in the material such that, if repeated, it may lead to crack initiation and subsequent growth. LCF is caused by relatively slow changes in stress. An example of LCF is temperature or pressure changes (transients) during plant startup, shutdown or other events.

EPRI agrees with the JNO Report definition of fatigue. For comparison, the standardized definition of fatigue that is followed in the U.S. and in some portions of the international community is from the American Society of Testing and Materials (ASTM), and is as follows [4.1-2]:

> *The process of progressive localized permanent structural change occurring in a material subjected to conditions that produce fluctuating stresses and strains at some point or points and that may culminate in cracks or complete fracture after a sufficient number of fluctuations.*

Another form of fatigue is high cycle fatigue (HCF) where the number of applications of repeated loading occur relatively rapidly such that a very high number of load cycles is accumulated rapidly. Examples of HCF include vibration or thermal mixing. HCF is distinguished from LCF in that the loading associated from HCF is considered to be infinite, so load levels must be kept below the fatigue endurance limit of the material; otherwise fatigue failures may occur quickly due to rapid load accumulation. In contrast, LCF loading is finite and may be addressed through aging management. Therefore, the damage due to LCF is relatively slow and may be monitored and, as a result, corrective action can be taken before fatigue failures occur.

Aging management of LCF for long-term plant operation is necessary to ensure the continued integrity for all Class 1 components, especially those components with discontinuities, such as nozzles (because of the increased loading caused by the presence of the discontinuity). All the Class 1 components comprising the reactor coolant pressure boundary of a NPP, such as the primary coolant piping, pressure vessels, steam generators and pressurizers, are subject to stringent LCF requirements as part of the component design that must be met throughout plant operation. Therefore, LCF is addressed in the JNO Report because it is identified in the aging management technical evaluation as one of the aging phenomena.

EPRI agrees with the JNO Report regarding the need to address LCF as an aging phenomenon for long-term plant operation. This approach is consistent with the U.S. approach for long-term plant operation that has been in practice for more than 20 years associated with license renewal (LR) for 60 years of operation and, more recently, for subsequent license renewal (SLR) for 80 years of operation. U.S. guidance for LCF for LR may be found in Chapter X.M1, "Fatigue Monitoring," of the Generic Aging Lessons Learned (GALL) Report [4.1-3], and for SLR in Chapter X.M1, "Fatigue Monitoring," of the GALL-SLR Report [4.1-4].

Also important to LCF evaluation for long-term operation is environmentally assisted fatigue (EAF), which is also addressed by the U.S. guidance for LR and SLR. EAF addresses the additional effects of a water environment that may not be fully addressed by fatigue curves derived from testing in air that were commonly used for component design for many of the worldwide operating nuclear power plants. EAF is also addressed in the JNO Report and is included in EPRI's review that follows.

Both sets of U.S. guidance with respect to LCF and EAF are very similar; the program description for LCF and EAF from the GALL Report, which is most appropriate for operation limited to 60 years in Japan, is as follows:

> ***Program Description***
>
> *Fatigue usage factor is a computed mechanical parameter suitable for gauging fatigue damage in components subjected to fluctuating stresses. Crack initiation is assumed to have started in a structural component when the fatigue usage factor at a point of the component reaches the value of 1, the design limit on fatigue. In order not to exceed the design limit on fatigue usage, the aging management program (AMP) monitors and tracks the number of critical thermal and pressure transients for the selected components. The program also verifies that the severity of the monitored transients are bounded by the design transient definition for which they are classified.*
>
> *The AMP addresses the effects of the reactor coolant environment on component fatigue life (to determine an environmentally-adjusted cumulative usage factor, or CUF_{en}) by assessing the impact of the reactor coolant environment on a set of sample critical components for the plant.*
>
> *Examples of critical components are identified in NUREG/CR-6260. Environmental effects on fatigue for these critical components may be evaluated using one of the following sets of formulae:*
>
> - *Carbon and Low Alloy Steels*
>   - *Those provided in NUREG/CR-6583, using the applicable ASME Section III fatigue design curve*
>   - *Those provided in Appendix A of NUREG/CR-6909, using either the applicable ASME Section III fatigue design curve or the fatigue design curve for carbon and low alloy steel provided in NUREG/CR-6909 (Figures A.1 and A.2, respectively, and Table A.1)*
>   - *A staff approved alternative*

- *Austenitic Stainless Steels*
    - *Those provided in NUREG/CR-5704, using the applicable ASME Section III fatigue design curve*
    - *Those provided in NUREG/CR-6909, using the fatigue design curve for austenitic stainless steel provided in NUREG/CR-6909 (Figure A.3 and Table A.2)*
    - *A staff approved alternative*
- *Nickel Alloys*
    - *Those provided in NUREG/CR-6909, using the fatigue design curve for austenitic stainless steel provided in NUREG/CR-6909 (Figure A.3 and Table A.2)*
    - *A staff approved alternative*

Any one option may be used for calculating the CUF_{en} for each material.

The purpose of citing the above guidance is not to imply that this guidance should be used in Japan to address long-term assessment of fatigue. Rather, this guidance forms the basis for EPRI's independent review of the JNO Report and the JNO technical evaluation of LCF for long-term shutdown. The U.S. guidance is comprehensive and has been applied to successfully manage LCF in U.S. nuclear power plants for more than 20 years, and it continues to be used to manage LCF for SLR. Consequently, comparisons between the U.S. and JNO approaches for managing LCF for long-term operation provide useful, independent content for EPRI's review of the JNO Report LCF technical assessment. The remaining LCF subsections therefore compare the U.S. practices for aging management of LCF to assess the JNO approach.

4.1.2 Factors Contributing to LCF

The JNO Report describes that the characteristics of LCF are assessed by testing that determines material-specific fatigue relationships specified as the range of strain, $\Delta\varepsilon$, to the fatigue life specified as the number of cycles to failure in air, N_{air}. An example ε-N fatigue curve for stainless steels is shown in Figure 4-2 (Figure 3.2.2.1 (1)-1 from the JNO Report). Fatigue curves are used to estimate LCF damage in components, as discussed in the next subsection.

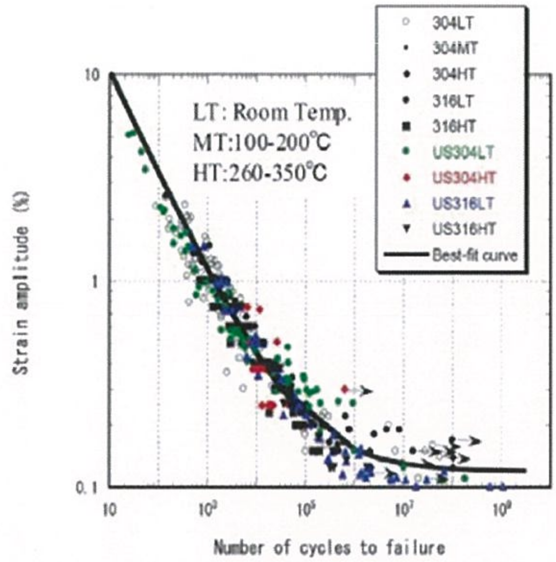
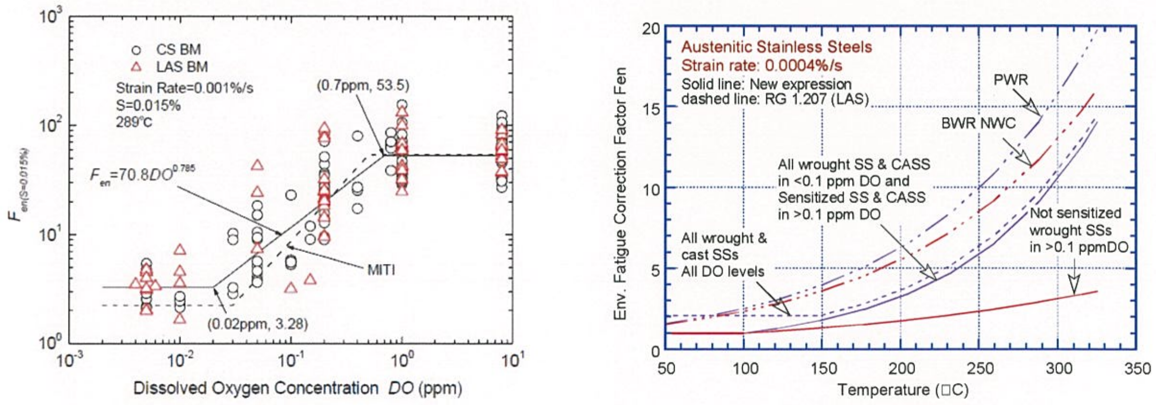


Figure 4-2
Example Fatigue ε-N Curve for Stainless Steel Material

The JNO Report also discusses EAF, defined by the ratio N_{air}/N_{en}, representing the fatigue life in air, N_{air}, divided by the fatigue life in a water environment, N_{en}. This ratio is also called the environmental effect correction factor, F_{en}. Research efforts into environmental fatigue have been actively pursued in Japan and the U.S. and similar methods for evaluating environmental fatigue have been developed. The Japanese research and development results were incorporated into JSME's Environmental Fatigue Evaluation Method [4.1-5], whereas the U.S. methods used for LR are documented in NUREG/CR-6909, Revision 0 [4.1-6]. Example environmental effect correction factors from Figure 3.2.2.1 (1)-3 of the JNO Report are shown in Figure 4-3.



(a) (b)

Figure 4-3
Example of the Environmental Effect Correction Factor (F_{en}) for Stainless Steels from (a) JSME's Environmental Fatigue Evaluation Method, and (b) NUREG/CR-6909, Revision 0

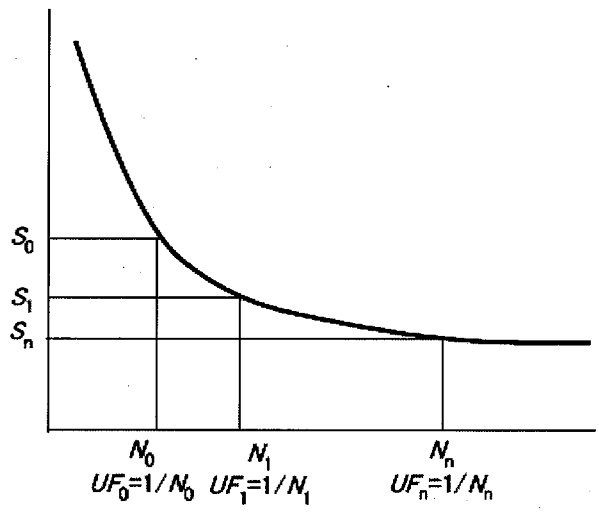
EPRI agrees with the JNO Report content regarding the factors that contribute to LCF. The factors are also consistent with the factors used in the U.S. for long-term plant operation, as identified in the fatigue aging management program description from the GALL Report cited in Section 4.1.1.

4.1.3 Actions to Deal with LCF

The JNO Report describes how LCF evaluation is performed during the design stage according to the JSME Code on Design and Construction using conservatively defined design transient conditions to confirm that no fatigue failure will occur during the operating period. Fatigue evaluation is also performed as part of the aging management technical evaluation by estimating the number of transients that will occur by the end of the evaluation life based on the actual number of transients that already have occurred up to the time of the evaluation. In addition, for systems or portions of systems subject to the environmental effects, fatigue evaluation is performed according to JSME's Environmental Fatigue Evaluation Method [4.1-5] taking into account the environmental effects.

In evaluating fatigue in air, a fatigue design curve similar to that shown in Figure 4-2 is used to compute a cumulative usage factor, *UF*, using the following formula and the method shown in Figure 4-4:

$$UF = \sum_{i=0}^{n} UF_i \qquad \textbf{Eqn. 4-1}$$



$UF=UF_0+UF_1+\ldots\ldots+UF_n$

Figure 4-4
Use of the Fatigue Design Curve to Determine the Cumulative Usage Factor, *UF*

Acceptable fatigue management and component integrity can be assured if *UF* is maintained to less than the allowed value of 1.0.

EAF is evaluated using F_{en} and the following formula:

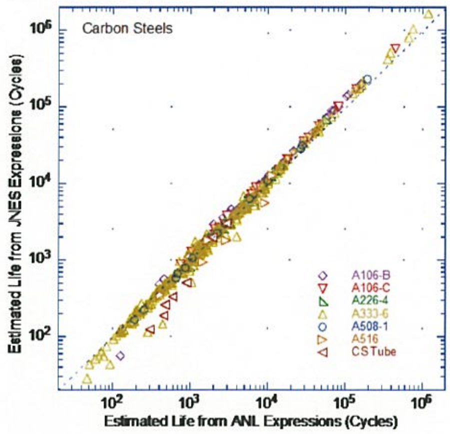
$$UF_{en,i} = F_{en,i} \times UF_i \qquad \textbf{Eqn. 4-2}$$

Acceptable EAF management and component integrity can be assured if the total UF_{en} ($\sum_{i=0}^{n} UF_{en,i}$) is maintained to less than the allowed value of 1.0.

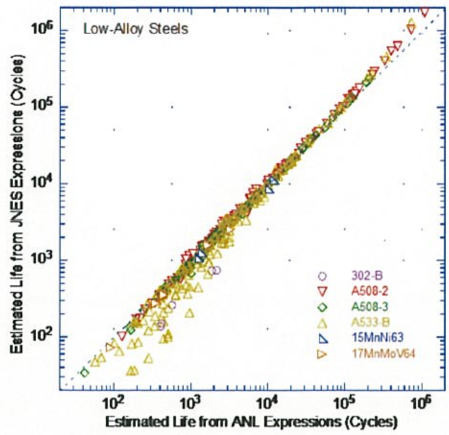
EPRI agrees with the JNO Report content regarding the actions to deal with LCF. Specifically, EPRI notes the following three aspects of the review:

1. **Evaluation of Fatigue in Air (*UF*)**: EPRI performed a detailed comparison of the JSME methods for evaluating fatigue (as used in Japanese NPPs) to the ASME methods for evaluating fatigue (as used in U.S. NPPs) as a part of EPRI's review of the long-term operation aging evaluation performed for Kansai for Takahama Units 1 and 2 in MRP-429 [4.1-7]. EPRI concluded that Kansai's methods are appropriate for use in evaluating long-term operation and will yield conservative results. Those methods are consistent with the methods described in the JNO Report, so EPRI's previous conclusions are valid for the JNO Report. Figure 3.2.2.1(1)-6 of the JNO Report shows consistency in *UF* evaluation for three JNO PWR plants for eight components.
2. **Evaluation of F_{en}**: EPRI performed a comparison of the JSME and U.S. F_{en} methods in Section 5.0 of MRP-429 [4.1-7]. The comparisons are shown in Figures 5-9 through 5-11 of MRP-429 for carbon and low alloy steels, stainless steels, and nickel-alloy steels, respectively. The comparison showed that, in some cases, the U.S. methods were conservative compared to the JSME methods. However, since MRP-429 was published, the NRC published NUREG/CR-6909, Revision 1 [4.1-8]. The results of the NRC's latest EAF research indicate that the Japanese and U.S. approaches to F_{en} are in very good agreement, as identified in Figure 4-5. EPRI therefore agrees with the JNO Report approach to evaluating EAF, and finds the F_{en} approach used in the JNO Report to be valid and consistent with methods employed in the U.S. for long-term operation.
3. **Evaluation of UF_{en}**: EPRI performed an independent assessment of the UF_{en} calculations performed by Kansai for the Takahama Long Term Operation Assessment. The approach used by Kansai in that assessment is identical to the approach described in the JNO Report. In order to assess the conservatism of the Kansai evaluation, EPRI performed an assessment of the F_{en} multipliers used by Kansai, as described above, as they are important for calculating UF_{en}. EPRI considers the F_{en} values obtained using NUREG/CR-6909, Revision 0 to be conservative, and they are accepted for use in the U.S. by the NRC to evaluate EAF for 60 years of plant operation. Based on that detailed F_{en} comparison, EPRI concluded that Kansai's UF_{en} calculations were consistent with U.S. methods, and the UF_{en} values calculated by Kansai using the methods of JSME S NF 1-2009 were sufficiently conservative. Because the JNO Report uses the same methods as those used by Kansai, EPRI's conclusions from the Kansai study are also applicable to the JNO Report.

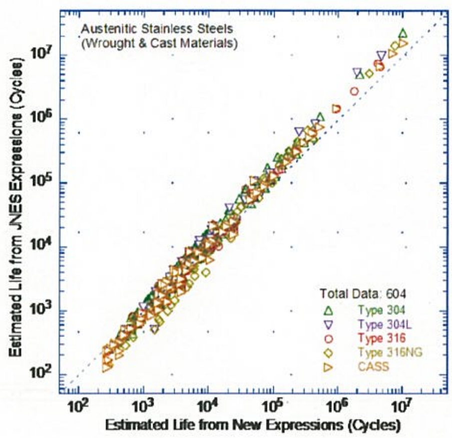
The actions to deal with LCF in the JNO Report are also consistent with the factors used in the U.S. for long-term plant operation, as identified in the fatigue aging management program description from the GALL Report cited in Section 4.1.1.



(a)

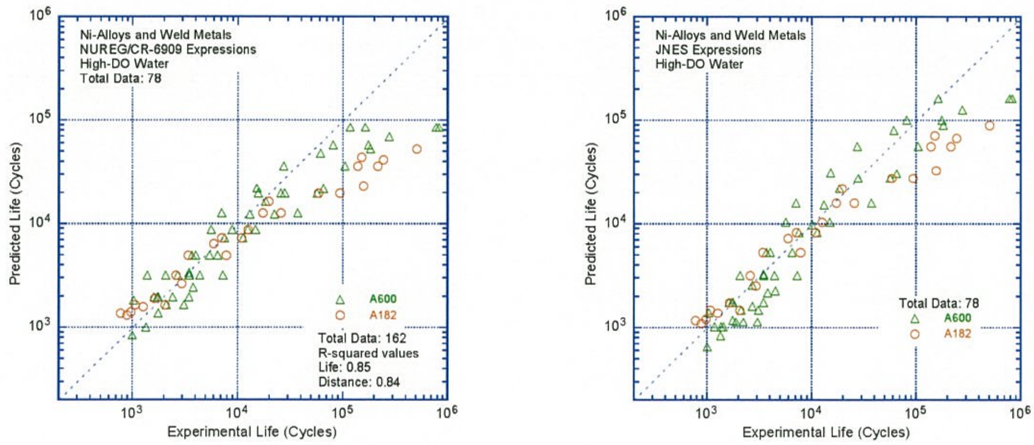


(b)



(c)

Figure 4-5
Comparison of Japanese (JNES) and U.S. (ANL) F_{en} Methods (from [4.1-8]) for (a) carbon steels, (b) low alloy steels, (c) stainless steels, and (d) nickel-alloy steels.



(d)

4.1.4 Examples of LCF Evaluations for Actual Units

The JNO Report described that all evaluations performed based on actual operation to-date of all Japanese PWR and BWR plants demonstrated the integrity of components and structures could be sufficiently assured for a period of evaluation life (for example, 60 years). Estimated cumulative usage factors for RPV components at Takahama Units 1 and 2 and Mihama Unit 3 (PWRs) and Tokai 2 (BWR), all of which have filed an application for extending the operating period, are shown in Figures 3.2.2.1(1)-6 and 3.2.2.1(1)-7 of the JNO Report. The LCF evaluation for these plants was conducted using conservative estimates of transient frequencies for future operation, which was based on the actual number of transients accumulated during past operation, as shown in Figure 3.2.2.1 (1)-8 of the JNO Report. The JNO Report concludes that, "*The low-cycle fatigue evaluation for these plants was conducted giving sufficient margins to the evaluation conditions, including conservatively estimated transient frequencies in future, which are based on the actual number of past transient times.... [T]he cumulative usage factor is below the allowable limit with sufficient margins even at the end of the evaluation life.*"

EPRI agrees with the JNO Report content regarding the examples of LCF evaluations for actual units. In EPRI's previous investigations for Takahama Units 1 and 2 [4.1-7], EPRI performed three assessments regarding the approach used to evaluate transients as part of the LCF evaluations for the Takahama units:

1. Assessment of the transient projection scheme
2. Assessment of the transient selection process
3. Comparison of the Takahama Unit 1 transients to similar U.S. PWRs:

EPRI made the following conclusions regarding the above three transient assessments:

1. **Transient Projection Scheme**: EPRI determined that the method for projecting transients described in the JNO Report, which is the same as the methods in the Takahama evaluation, is conservative for two reasons. First, the future rate of transient accumulation considers the average rate of actual transient accumulation for all plant operation to-date. This is conservative because world-wide experience with nuclear plant operation indicates that transient accumulation rates are lower the longer plants operate. Generally, this is due to improved operating practices and modified fuel design and loading practices that allow longer operating cycles. A graphical illustration of this conservatism is shown in Figure 4-6, where different methods of projections for the Takahama Unit 1 Startup Event are shown. Use of the average rate of transient accumulation is conservative compared to a projection that is based on more recent plant operating practices. The second reason that the transient projection method is conservative is the application of an increased rate of transient accumulation for future operation. Figure 3.2.2.1(1)-8 of the JNO Report shows this increased rate (the slope of the "Estimated transients" line is steeper than the "Past records" line). Based on this information, EPRI concludes the transient projection scheme is appropriate and conservative for evaluating long-term operation.

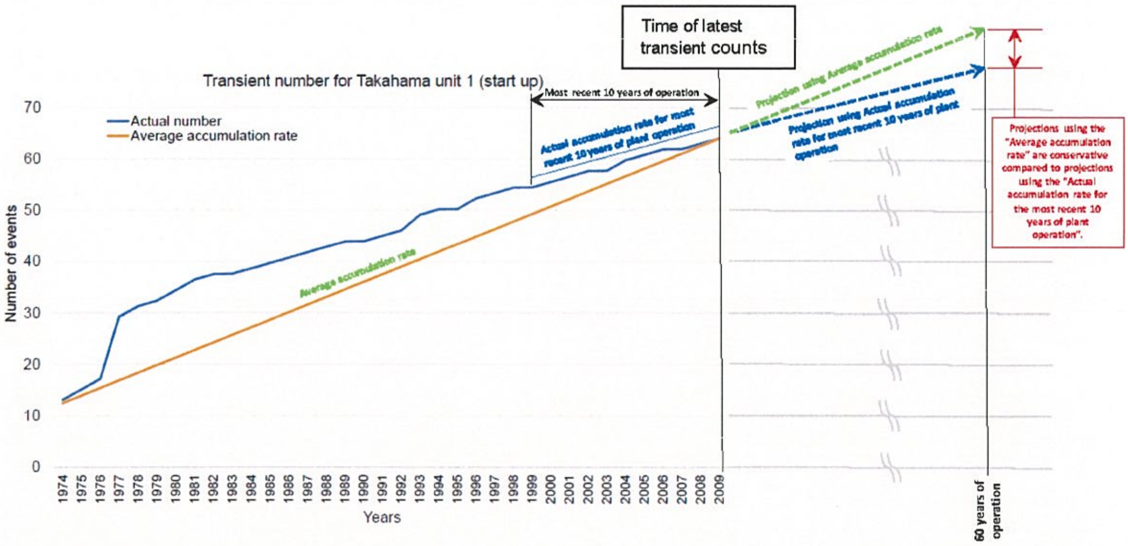


Figure 4-6
Transient Projection Methods

2. **Transient Selection Process**: The transient selection process is not described in the JNO Report. However, EPRI assumes this process is identical to the process used by Kansai to select transients for Takahama Units 1 and 2 [4.1-7]. This is a reasonable assumption based on a comparison of the transients listed in Table 3.2.2.1(1)-1 of the JNO Report for Mihama Unit 3 to those listed in Table 5-1 of the EPRI report [4.1-7]. The EPRI review indicates those transients represent all transients evaluated for fatigue in the component design. The transients are divided into Operating Conditions I and II in Table 5-1, which is similar to Service Levels A and B (or Normal and Upset conditions) used by Section III of the ASME Code for U.S. plants. EPRI concluded the transient selection process is consistent with U.S. practices for extended plant operation and is appropriate for long-term operation. Because the JNO Report describes a transient selection process that is the same as the process used by Kansai, EPRI's conclusions from the Kansai study are also applicable to the JNO Report.

3. **Comparison of Transients to Similar U.S. PWRs**: To provide further insight into the transient selection and projection processes reported in the JNO Report, EPRI compared transient types and projections for 13 similar U.S. PWRs for 60 years of operation to the transient projections for Takahama Unit 1 in MRP-429 [4.1-7]. That comparison is shown in Table 5-3 of MRP-429. From that comparison study, EPRI concluded that the transient projections for Takahama are reasonable and slightly better than average vs. the U.S. PWRs. Table 3.2.2.1(1)-1 of the JNO Report provides similar projections of Operating Mode I transients for Mihama Unit 3. The Startup transient for Mihama Unit 3 is compared to the Takahama Unit 1 and U.S. PWR Startup transient projections from Table 5-3 of MRP-429 in Table 4-1 and Figure 4-7[2]. For transient projections, plants in the U.S. have successfully used both of projection methods shown in Figure 4-6. The U.S. PWRs tend to use the "Average Accumulation Rate" which is less conservative than what the JNOs used for

[2] Note that the average number of events shown in Figure 4-7 (128) differs from the average number of events shown in Figure 5-4 of MRP-429 (131) because of the inclusion of Mihama Unit 3 in this study, which was not considered in MRP-429.

Mihama Unit 3 and Takahama Unit 1 (because the JNO Report includes the application of an increased rate of transient accumulation for future operation). Based on this information, EPRI finds the JNO Report treatment of transient projections is conservative for evaluating long-term operation.

Table 4-1
Comparison of Operating Mode I Transient Projections for Mihama Unit 3 [4.1-1] and Takahama Unit 1 [4.1-7]

Event No.	Transient	Mihama Unit 3		Takahama Unit 1	Mihama vs. Takahama (%)
		Past # of Transients (at end of FY 2010)	Estimated # of Transients (60 Years)	Estimated # of Transients (60 Years)	
1	Startup (temperature increasing rate: 55.6°C/h)	46	78	99	79%
2	Shutdown (temperature decreasing rate: 55.6°C/h)	44	78	99	79%
3	Load increase (load increasing rate: 5%/min)	368	706	710	99%
4	Load decrease (load decreasing rate: 5%/min)	353	691	687	101%
5	Step-wise load increase from 90 to 100%	2	3	5	60%
6	Step-wise load decrease from 100 to 90%	2	3	6	50%
7	Large step-wise load decrease from 100%	4	7	4	175%
8	Fluctuation during steady load operation	---	---	---	---
9	Refueling Operations	23	52	55	95%
10	Load increase from 0 to 15%	50	84	112	75%
11	Load decrease from 15 to 0%	37	69	86	80%
	1 loop shutdown/1 loop startup				
12	I) Shutdown	0	1	1	100%
13	II) Startup	0	1	1	100%

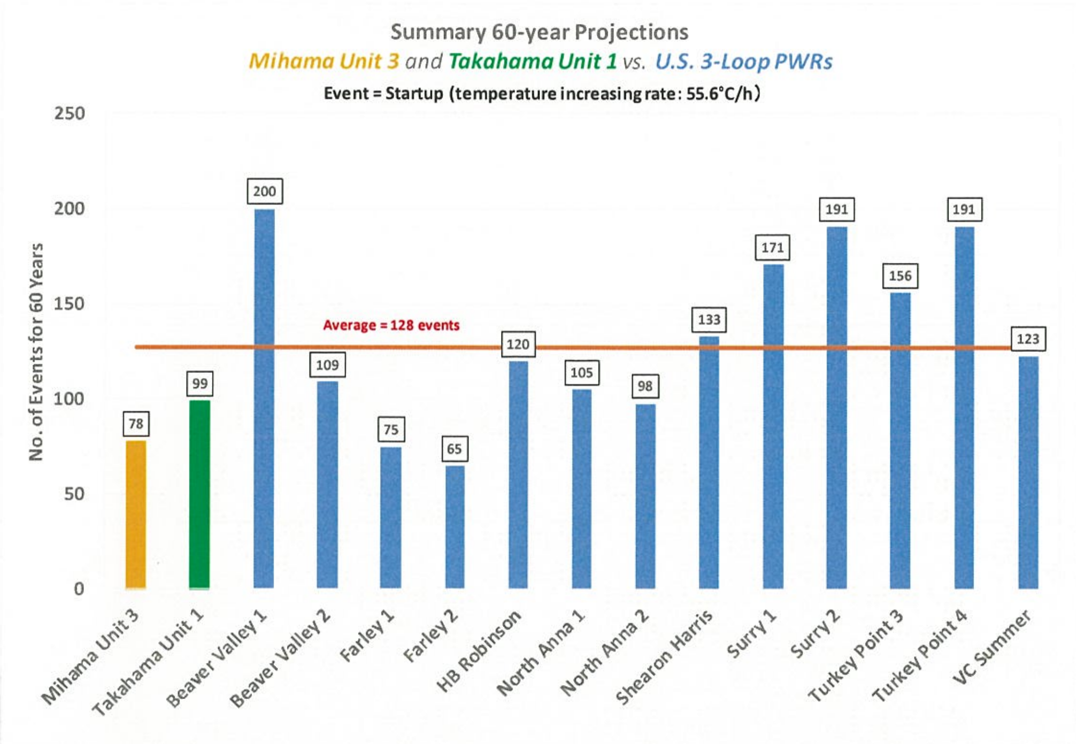
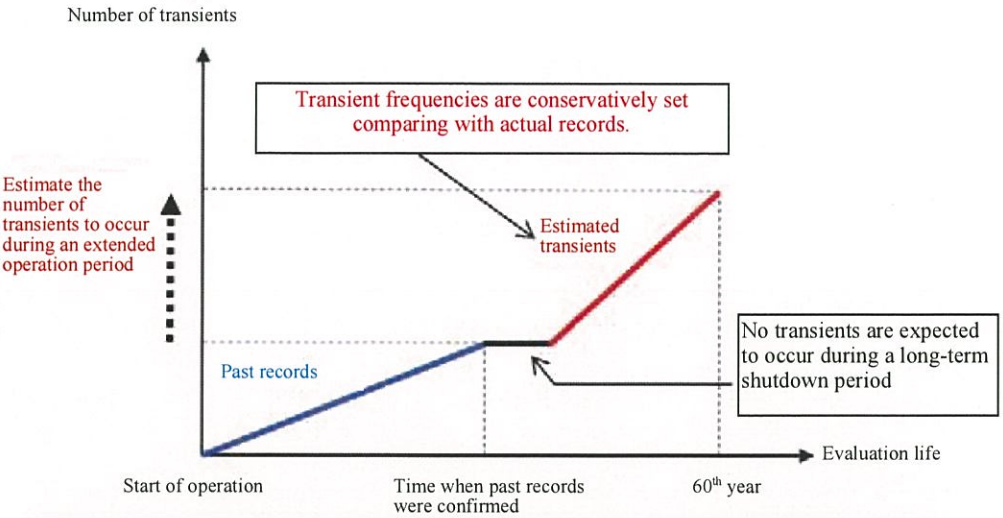


Figure 4-7
Comparison of Mihama Unit 3 and Takahama Unit 1 Startup Event 60 Year Projections to 3-Loop Westinghouse U.S. PWRs

Therefore, EPRI concludes the JNO Report approach to LCF evaluation for actual units is valid, and that the JNO Report approach to evaluate LCF in actual units is consistent with the approach used in the U.S. to manage fatigue for long-term plant operation.

4.1.5 Effects of Long-Term Plant Shutdown on LCF

The JNO Report states that, in performing the LCF evaluation for individual RPV components, transient events such as those listed in Table 4-1 were considered. However, the JNO Report concludes that, "*...since no transient events are expected to occur during long-term shutdown, plant long-term shutdown will cause no significant effects on LCF.*" Consideration of this effect on LCF is reflected in Figure 3.2.2.1(1)-8, which is duplicated in Figure 4-8.



The number of transient times for a period of evaluation life is calculated based on the past operating records (in a blue line) while conservatively estimating transient frequencies in future (in a red line).

Figure 4-8
Image of Calculation of Number of Transients for Evaluation

EPRI agrees with the JNO Report conclusion that there are no significant LCF effects during long-term plant shutdown. EPRI's agreement is based on the fact that there are no significant transients during plant shutdown periods. LCF transients, such as those listed in Table 4-1 that are typical for 3-loop PWRs, are caused by changes in temperatures, pressures and flow rates. However, during plant shutdown:

- There is no system pressure, so there are no pressure excursions that lead to changes in component stresses.
- System temperatures are very low and constant, approaching ambient conditions, so there are no significant temperature gradients that will lead to significant changes in component stresses.
- System flow rates are low (for systems that must run during shutdown to filter reactor coolant) or zero (for most systems that are turned off). This lack of flow, combined with the low temperatures described above, leads to little or no potential for fluids at different temperatures to mix and cause thermal cycling-induced stresses.

Even for small temperature differentials that may exist between different system fluids, under the assumption that those fluids could mix if cooling system pumps are operated and system valves were opened, LCF impacts would be insignificant. Such mixing is inconsequential during plant shutdown because available temperature differentials during plant shutdown are well below the thresholds that lead to LCF. EPRI has performed extensive research for thermal fatigue [4.1-9 through 4.1-23]. EPRI's research has been primarily used to identify plant locations susceptible to HCF.

As mentioned previously, HCF situations must be avoided because the rapid accumulation of cycles can lead to premature component failure. Therefore, EPRI's thermal fatigue screening criteria for HCF is based on the fatigue endurance limit for materials, i.e., stresses below the tail end of the fatigue curve shown in Figure 4-2. Transients that cause stresses below the fatigue endurance limit have no consequence to the *UF*; an infinite number of such load cycles can be applied to a component without causing a fatigue failure.

The HCF thermal screening criteria can be applied to long-term shutdown conditions to further support the conclusion that long-term shutdown does not contribute to LCF. As mentioned above, these screening limits are based on the temperature differential required to cause a thermal shock stress in piping components equal to the material fatigue endurance limit. There are no temperature differences of this magnitude between plant systems during plant shutdown. Therefore, even in a case where fluids with different flow rates available during a plant shutdown were mixed together, the resulting transients would be well below the threshold to cause LCF.

4.1.6 Conclusions Regarding LCF

EPRI agrees with the content of the JNO Report for the five LCF topics addressed in Section 3.2.2.1(1) of the JNO Report and the overall approach described in the JNO Report to assess the impact of long-term shutdown on LCF. EPRI's detailed review of the five LCF topics are described in Sections 4.1.1 through 4.1.5.

With respect to LCF, the JNO Report reaches three important conclusions:

1. As discussed in Section 4.1.4, the JNO Report identifies that, "*The low-cycle fatigue evaluation for these plants was conducted giving sufficient margins to the evaluation conditions, including conservatively estimated transient frequencies in future, which are based on the actual number of past transient times.... [T]he cumulative usage factor is below the allowable limit with sufficient margins even at the end of the evaluation life.*"
2. As discussed in Section 4.1.5, the JNO Report concludes that, "*Since no transient events are expected to occur during long-term shutdown, plant long-term shutdown will cause no significant effects on low-cycle fatigue.*"
3. Table 3.3-1 of the JNO Report identifies that, "*[it is] Unnecessary to consider the accumulation of fatigue since there are no significant pressure and temperature fluctuations during plant shutdown.*"

EPRI agrees with these three important conclusions. In addition, EPRI notes that the LCF aging management program at all plants requires that every thermal transient be identified, tracked, and evaluated on a continuous basis. This is identified in Line No. 4 of Table 3.1.2-2 in the JNO Report. Therefore, even in the unlikely event that a transient of significance was to occur during any plant shutdown (short or long), existing aging management programs would capture and evaluate the impact of the event.

Based on the reviews contained in Sections 4.1.1 through 4.1.5, EPRI agrees with the main JNO Report conclusion regarding LCF that, since no transient events are expected to occur during long-term shutdown, long-term plant shutdowns will not cause significant effects on LCF.

4.1.7 References

Note: References highlighted in yellow are available for purchase from EPRI; they are available at no charge to EPRI members.

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4.1-3. NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," December 2010.

4.1-4. NUREG-2191, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," July 2017.

4.1-5. JSME SNF1-2009, "Environmental Fatigue Evaluation Method for Nuclear Power Plants," Japan Society of Mechanical Engineers, 2009.

4.1-6. NUREG/CR-6909, Revision 0, ANL-06/08, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," February 2007.

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4.1-8. NUREG/CR-6909, Revision 1, "Effect of LWR Water Environments on the Fatigue Life of Reactor Materials," May 2018.

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4.1-10. *Materials Reliability Program: Fatigue Management Handbook, Revision 1 (MRP-235)*. EPRI, Palo Alto, CA: 2008. 1015010.

4.1-11. *Materials Reliability Program: NDE Technology for Detection of Thermal Fatigue Damage in Piping (MRP-23, Revision 2)*. EPRI, Palo Alto, CA: 2017. 3002010699.

4.1-12. *Mitigation of Thermal Fatigue in Unisolable Piping Connected to PWR Reactor Coolant Systems (MRP-29)*. EPRI, Palo Alto, CA: 2000. 1001017.

4.1-13. *Materials Reliability Program: Thermal Fatigue Monitoring Guidelines (MRP-32, Revision 1)*. EPRI, Palo Alto, CA: 2011. 1022563.

4.1-14. *Materials Reliability Program: Computer-Based NDE Training for Thermal Fatigue Cracking (MRP-36, Rev. 3)*. EPRI, Palo Alto, CA: 2017. 3002010700.

4.1-15. *Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146, Revision 2)*. EPRI, Palo Alto, CA: 2016. 3002007853.

4.1-16. *Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines – Supplemental Guidance (MRP-146S)*. EPRI, Palo Alto, CA: 2009. 1018330.

4.1-17. *Materials Reliability Program: Integrated Fatigue Management Guideline (MRP-148, Revision 1)*. EPRI, Palo Alto, CA: 2012. 1025159.

4.1-18. *Materials Reliability Program: Fatigue Licensing Basis Monitoring Guideline (MRP-149, Revision 1)*. EPRI, Palo Alto, CA: 2013. 3002000684.

4.1-19. *TFE 1.0 - MRP-170: EPRI Thermal Fatigue Evaluation per MRP-146, Version 1.0*. EPRI, Palo Alto, CA: 2006. 1013270.

4.1-20. *Materials Reliability Program: Assessment of Residual Heat Removal Mixing Tee Thermal Fatigue in PWR Plants (MRP-192, Revision 2)*. EPRI, Palo Alto, CA: 2012. 1024494.

4.1-21. *Materials Reliability Program: Proceedings of MRP-146, Revision 1 Training (MRP-339)*. EPRI, Palo Alto, CA: 2012. 1025017.

4.1-22. *BWRVIP-155: BWR Vessel and Internals Project, Evaluation of Thermal Fatigue Susceptibility in BWR Stagnant Branch Lines*. EPRI, Palo Alto, CA: 2006. 1013389.

4.1-23. *BWRVIP-196: BWR Vessel and Internals Project, Assessment of Mixing Tee Thermal Fatigue Susceptibility in BWR Plants*. EPRI, Palo Alto, CA: 2008. 1016570.

4.2 Neutron Irradiation Embrittlement

Neutron irradiation embrittlement is the second of the four specific degradation phenomena that is evaluated in the JNO Report [4.2-1]. Reactor pressure vessel (RPV) integrity, which includes assessment of neutron irradiation embrittlement of the ferritic RPV materials, is a critical element necessary to support continued, safe, NPP operation for the entire life of the plant. Figure 4-9 provides a comparison of a generic RPV material in the unirradiated and irradiated condition. As shown in the figure, neutron irradiation impacts both the reference nil-ductility transition temperature (RT_{NDT}), or shift, and the upper shelf energy (USE) by shifting the irradiated Charpy V-notch curve down and to the right. Management of these two effects forms the basis of maintaining RPV integrity.

In the U.S., there are numerous NRC regulations that must be addressed for license extension to 60 or more years of plant operation. Justification of safe plant operability to 60 years, or beyond, in terms of RPV integrity, can be used to demonstrate that recovery of long-term shutdown is technically feasible and acceptable within the existing license renewal framework. Thus, EPRI compared the JNO Report against the regulatory guides and NUREG reports that have been developed to assist U.S. licensees in meeting the regulations. U.S. NPP utility license holders utilize the SRP-LR, NUREG-1800, Revision 2 [4.2-2], and the GALL Report, NUREG-1801, Revision 2 [4.2-3], for generic guidance to when applying for license extension to 60 years. Similarly, for a potential SLR to 80 total years of plant operation, NUREG-2192 [4.2-4] provides the SRP information and NUREG-2191 [4.2-5] details the GALL for SLR.

EPRI performed a technical review of the JNO Report, which asserts that the RPV is the only vessel/piping component that experiences sufficient neutron irradiation to require aging management, utilizing relevant NRC standards as the basis for our review. This review compared the U.S. license extension methods, which could be cited during a long-term shutdown recapture evaluation, to the Japanese codes and standards to determine if any differences exist in the methods used by the JNOs and to identify areas where their approach could be non-conservative compared to applicable U.S. evaluations.

EPRI agrees with the fundamental premise of the JNO report regarding RPV integrity; the RPV is the most important pressure boundary component for light-water reactors (LWRs). Comments on low alloy steel neutron embrittlement, withdrawal and testing of RPV surveillance capsules, pressurized thermal shock (PTS), and USE reduction are included in the following subsections. EPRI also observed that a fourth aspect of RPV integrity, pressure-temperature (P-T) limit curves, is not included in the JNO Report. However, the JNO utilities have performed aging management using P-T limit curves and will continue to do so in accordance with Japan's regulatory requirements upon restarting operation. Additional comments and details on this aspect of RPV integrity are included in Subsection 4.2.5. Finally, EPRI recommends modifying the title of this subsection in the JNO Report by adding "*...of the Reactor Pressure Vessel*" since the entirety of this subsection is devoted to RPV integrity concerns and evaluations. This suggestion should minimize any confusion as to the intent of this portion of the JNO Report.

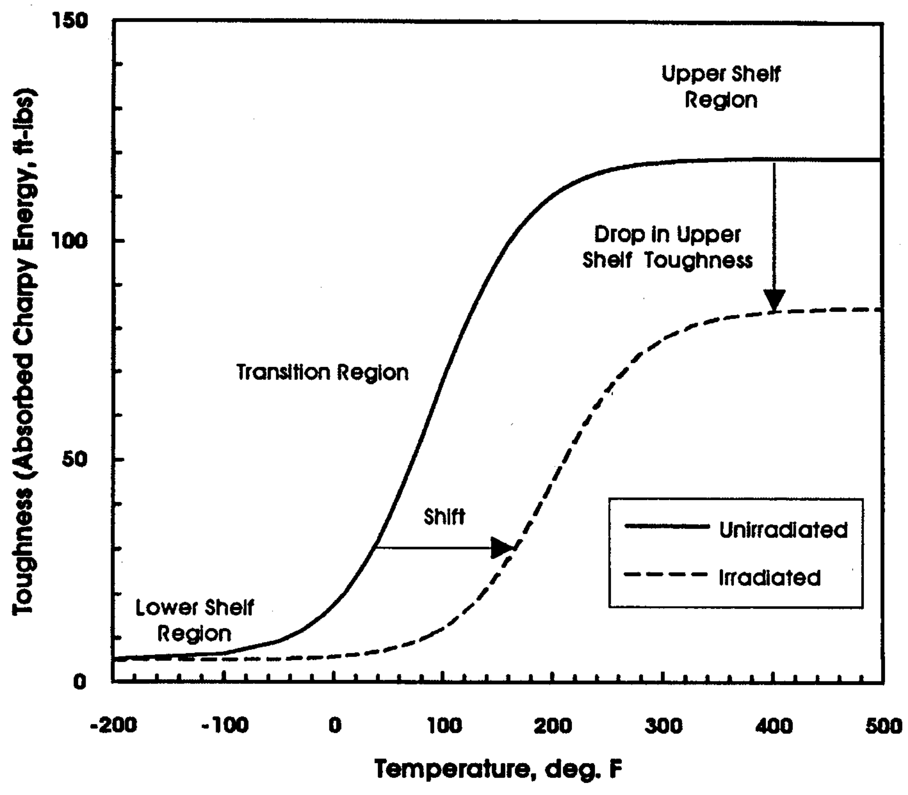

Figure 4-9
Illustration of RPV Material Toughness and Neutron Irradiation Effects

4.2.1 Neutron Embrittlement

The JNO Report discussion of the various mechanistic processes which lead to neutron embrittlement is consistent with the generally-accepted technical knowledge. The applicable fracture mechanics properties of concern for RPVs are documented in the report. EPRI also agrees with the JNO Report assertion that embrittlement increases with exposure to neutron fluence based on the chemical composition of the RPV ferritic material, particularly its copper (Cu) and nickel (Ni) contents. This is consistent with the U.S. embrittlement trend correlation (ETC) documented in Regulatory Guide 1.99, Revision 2 [4.2-6]. As shown in Figure 4-9, fracture toughness of RPV low alloy steels is highly temperature dependent. This means that, at low temperatures, a given steel exhibits brittle properties then proceeds through a brittle-to-ductile transition as temperature is increased, and finally achieves fully ductile properties at higher temperatures. Irradiation increases the brittle-to-ductile transition temperature, while reducing the ultimate ductile USE value. This is consistent with the JNO Report, as illustrated in Figures 3.2.2 (2)-3 and 3.2.2 (2)-4.

As noted, ETCs and the level of embrittlement they predict for a given material is a function of the Cu and Ni contents of the irradiated material and the amount of irradiation, or fluence, it receives. Increases to any of these three parameters will increase neutron irradiation embrittlement. The Cu and Ni contents of a given material are set upon completion of fabrication of the individual component and cannot change during plant operation or during a long-term shutdown event. Fluence is the only time-dependent variable in most ETCs (It is noted that some more modern ETCs also incorporate temperature into the calculations, as opposed to Regulatory Guide 1.99, Revision 2 [4.2-6], which merely requires the temperature to be within a certain range of typical plant T_{cold} or T_{inlet} values. However, this is a minor variable in most cases, and is not as impactful as neutron fluence). Increases in fluence will result in increasing embrittlement. Furthermore, it has been demonstrated for some ETCs [4.2-7] that a decrease in the rate of fluence accumulation, or flux, may also lead to increased embrittlement. These effects are shown in Figure 3.2.2 (2)-6 of the JNO Report. However, this flux effect phenomenon is not universally accepted, nor is it included in all currently available ETCs [4.2-8]. Therefore, its inclusion in the overall neutron embrittlement calculation for a plant's RPV can be considered conservative. However, it is noted that if a flux effect parameter is included in an ETC such as that used in Japan or in 10 CFR 50.61a [4.2-7] in the U.S., it would only have an impact on BWR predicted embrittlement since those plants have lower fluxes. PWR fluxes are much higher, as indicated in Figure 3.2.2 (2)-6. This is consistent with observations of the flux values in U.S. PWRs. Overall, PWR plants are predicted to have higher embrittlement than BWR plants, with consistent material properties and taking into consideration flux effect impacts, because the end of life (EOL) fluence of a typical PWR is an order of magnitude, or more, higher than a corresponding BWR plant.

Finally, EPRI concurs with the Venn diagram detailed on Figure 3.2.2 (2)-5 of the JNO Report that three contributing factors are needed to initiate a fracture in an RPV. These factors are:

1. Reduced fracture toughness due to neutron embrittlement,
2. Existence of a critically-sized defect or flaw, either present from component fabrication or service induced (i.e., fatigue cracking), and
3. Transient loadings due to an emergency event which combines high RPV pressure and low or reduced temperatures, such as a PTS event.

Regarding the first of these three points, neutron fluence is the only active variable during plant operation for increasing neutron embrittlement. In the U.S., it is accepted that accumulated fluence below 1 x 10^{17} n/cm^2 can be excluded or ignored in RPV integrity calculations. This fluence limit was originally based on the threshold value documented in 10 CFR 50, Appendix H

[4.2-12] which requires plants with predicted fluence levels above this minimum value to develop and implement an RPV surveillance program. The NRC confirmed this value in NRC RIS 2014-11 [4.2-9] and TLR-RES/DE/CIB-2013-01 [4.2-10]. Thus, if the level of accumulated fluence in an RPV material is below this threshold value, no significant embrittlement is predicted to occur and it can be excluded from the various RPV integrity evaluations which will be discussed in more detail in later sections of this report. During a long-term shutdown period, when the reactor core is not critical and the plant is not producing power, neutrons are not generated and the level of irradiation during this long-term shutdown period is essentially zero. As a result, there is no potential for neutron fluence to escape the reactor core, impact the RPV wall and cause further embrittlement. Therefore, irradiation embrittlement does not need to be considered as an aging mechanism for the JNO plants during the long-term shutdown period.

Regarding the second point, all RPVs, prior to installation and entering operation, are given a complete nondestructive examination (NDE) to confirm that either no flaws exist in the RPV or that any flaws that do exist are appreciably small and satisfy construction requirements. In Japan, the requirements of the JSME apply and, in the U.S., the requirements of the ASME Code [4.2-11] are used in performing inservice inspection. Any flaws discovered during these exams can be disposition or mitigated as part of the normal plant maintenance practices. Furthermore, since no loadings are applied to the RPV during the long-term shutdown period, any discovered flaws will not propagate. Thus, EPRI agrees that management of potential flaws will continue to be mitigated with respect to RPV integrity fracture initiation.

The third point above, regarding transient loadings or potential PTS events, will be discussed in further detail in Section 4.2.3.

4.2.2 Surveillance Capsule Withdrawal and Testing

RPV surveillance capsule programs are required for all U.S. plants by 10 CFR 50, Appendix H [4.2-12]. Appendix H incorporates, by reference, the requirements of American Society for Testing and Material (ASTM) E185-82 [4.2-13] for development of a surveillance program, establishing the capsule withdrawal schedule, and capsule testing and reporting requirements. In addition to these requirements, Revision 2 of the GALL Report provides further NRC guidance for U.S. plant surveillance programs for 60 years of plant operation. In general, a plant must satisfy the required number of capsules to be withdrawn and tested based on their predicted level of embrittlement per ASTM E185-82. In addition, the final capsule tested should have a neutron fluence greater than once, but less than twice, the predicted fluence at the end of the licensed operating life.

4.2.2.1 RPV Surveillance Programs General Review

Periodic removal of RPV surveillance capsules allows the mechanical properties of the RPV materials contained within the program to be tracked and assessed as the plant operates and accrues neutron fluence. In most plant capsule designs, the capsules are placed closer to the core than the RPV wall, such that they are irradiated faster than the RPV. This increase in irradiation is called a lead factor (LF), and the LF ensures that results from RPV surveillance programs will be generated before the plant reaches the neutron fluence accumulated by the capsule. Mechanical specimens including Charpy, tensile or fracture toughness specimens (usually Compact Tension [CTs]), temperature monitors and flux monitors are contained within a typical capsule, as shown in Figure 4-10. The mechanical specimens are manufactured from surplus pieces of RPV materials, and include both base metal and weld metal. The "key specimen" of most plant surveillance programs is the Charpy specimen. Testing of this specimen yields a Charpy curve, as documented in Figure 4-9 of this report and Figure 3.2.2 (2)-4 of the JNO Report. By withdrawing and testing surveillance materials periodically, a spectrum of Charpy curves can be generated at various fluence levels and used to show the progression of neutron embrittlement throughout plant operation.

In the U.S., a minimum of three capsules are required for operation to 40 calendar years. Similarly, in Japan at least three capsules are required to be withdrawn and tested up to 32 effective full power years (EFPY) of plant operation. Many plants must withdraw and test an additional capsule or two as a function of the level of predicted embrittlement of the RPV and the applicable RPV integrity standards. In addition, in terms of operation to 60 years or beyond, at least one additional capsule, over and above the original requirements, must be withdrawn and tested. In this way, the licensee can determine an accurate representation of the embrittlement trend of their RPV to the regulator.

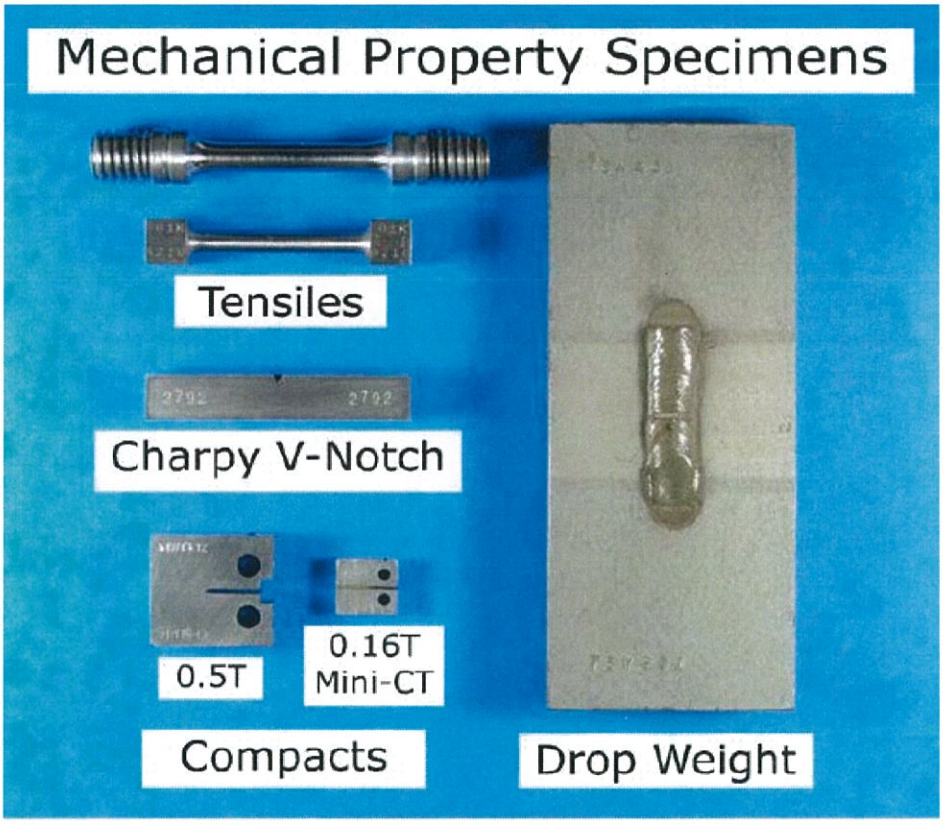

Figure 4-10
Photograph of Typical Test Specimens Used to Determine Mechanical Properties of RPV Materials [4.2-14]

4.2.2.2 Integrated Surveillance Programs

In the U.S., the Boiling Water Reactor Vessel and Internals Project (BWRVIP) developed an Integrated Surveillance Program (ISP) in the late 1990s. The BWR ISP was conceived as a program that combines surveillance materials from the existing U.S. BWR programs and the Supplemental Surveillance Program (SSP) to make sufficient materials available to improve compliance with 10 CFR 50 Appendix H. Instead of using the plant-specific surveillance data from a given plant, the data from all BWR surveillance programs were evaluated to select the "best" representative materials to monitor radiation embrittlement for that plant. Selection of the best representative materials for a particular plant considers heat number, similar chemistries, common fabricator, and the availability of unirradiated data. In matching the available surveillance plates and welds, some capsule materials are good representatives for the limiting materials of multiple plants. As a result, the ISP results in better representation of the limiting beltline materials for each plant, while reducing the number of capsules to be tested. The withdrawal of capsules not chosen as ISP capsules are deferred indefinitely. Therefore, the fleet using the ISP is allowed to operate the surveillance test program more efficiently, compared to the individual surveillance programs for the U.S BWR fleet.

4.2.2.3 Surveillance Program Conclusions

As confirmed in the JNO Report, plants will continue to monitor and assess their individual RPV integrity situations via surveillance capsule testing after restart of their respective plants. All plants that desire to operate 60 total calendar years, with or without the included service time for the long-term shutdown, must first test three capsules for 32 EFPY and then one additional surveillance capsule every 10 years after the 30th year of plant operation to satisfy Japanese Electric Association Code (JEAC) Document No. JEAC-4201 [4.2-14] and Nuclear Regulation Authority (NRA) requirements, respectively. Thus, if the return of the long-term shutdown service time is granted, the licensees may need to adjust their surveillance capsule withdrawal and testing schedules to incorporate this additional operational time. This is a standard evaluation that is typically done after each prior capsule withdrawal and testing as well as when the plants would have originally performed their 60-year evaluations. Thus, the surveillance capsule testing section of the JNO Report is logical and acceptable when compared to equivalent standards available in the U.S.

4.2.3 Pressurized Thermal Shock

EPRI reviewed the JNO approach for PTS evaluation documented in the JNO Report. Section 4.2.3.1 provides generic background on PTS and Section 4.2.3.2 details EPRI's assessment of the evaluations performed in the JNO Report.

4.2.3.1 PTS Background Information

PTS events are transients that produce a rapid decrease in the primary system coolant temperature to relatively low temperatures coincident with high primary system pressure. PTS events can result from a variety of causes including: system transients that may be initiated by instrumentation and control system malfunctions; accidents such as small break loss-of-coolant accidents (SBLOCA), main steam line breaks, and feedwater pipe breaks.

The operational parameters that control the severity of the PTS transient are the primary coolant system pressure, temperature and cooldown rate. A rapid cooldown of the primary coolant to relatively low temperature produces high transient tensile thermal stresses in the RPV wall near the RPV inner surface. These thermal stresses, in combination with the pressure stresses, create significant potential to extend a flaw that may exist, or is postulated to exist, at or near the RPV inner surface. The cooldown to relatively low temperature concurrently reduces the material toughness in the region near the RPV inner surface. The high thermal and pressure tensile stresses coupled with a low material toughness can significantly reduce the margin against RPV failure associated with flaw extension through the RPV wall.

In the U.S., the original PTS Rule, 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events" [4.2-15], was adopted in 1985 and applies only to PWRs. The rule establishes screening criteria below which the potential for RPV failure due to a PTS event is deemed to be acceptably low. The PTS screening criteria specifies the reference temperature for pressurized thermal shock, or RT_{PTS}. This is equivalent to an EOL RT_{NDT} calculated using a surface fluence value and cannot exceed 270°F (132.2°C) for plates, forgings, and axial weld materials, and 300°F (148.9°C) for circumferential weld materials. If these limits are exceeded, it is acceptable to utilize the Alternate PTS Rule, 10 CFR 50.61a [4.2-7], in lieu of flux reduction techniques or other operational restrictions on the plant to comply with 10 CFR 50.61. The Alternate PTS Rule is based on probabilistic fracture mechanics (PFM) and invokes additional requirements on a plant licensee. The main differences between the two PTS rules are a new ETC based on the work by Eason, Odette, Nanstad and Yamamoto (EONY), statistical data checks on surveillance data, and RPV inspection results analysis. The screening criteria have more subdivisions depending on material type and other parameters. The preceding general information regarding PTS was taken from and is described further in the RPV Integrity Primer, MRP-278, Revision 1 [4.2-16].

In the NRC's technical basis for the PTS Rule (SECY-82-465, dated November 23, 1982 [4.2-17]), large break loss-of-coolant accidents (LBLOCAs) had the lowest frequency of occurrence per reactor-year of any of the transients considered. The three transients discussed above (SBLOCA, main steam line breaks and feedwater pipe breaks) are considered the most significant PTS events in terms of their contribution to the overall risk of vessel failure. The PWR Westinghouse Owner's Group (WOG) considered LOCAs with pipe diameters greater than 2 inches (51 mm) to have negligible probability of causing vessel failure; thus, they were further excluded from analysis. The NRC [4.2-17] agreed with the WOG assessment. Note that in NUREG-1806, Vol. 1, Section 8.5.4.5.2 [4.2-18], the NRC's assessment performed to support development of the Alternate PTS Rule (10 CFR 50.61a [4.2-7]), the NRC confirmed that pipe breaks larger than 2.5 inches could be excluded. Therefore, EPRI considers the Japanese approach to treating LBLOCAs in their analysis to be conservative.

Finally, as noted in [4.2-16], PTS is not a concern for BWRs because there is a large volume of water in the RPV at saturated conditions. A sudden drop in temperature will condense steam and result in a decrease of pressure, so simultaneous conditions of low temperature and high pressure are improbable. Thus, PTS is not a concern for BWR plants. This is consistent with the discussion contained in the PTS section of the JNO Report.

In the U.S., a plant must calculate RT_{PTS} values for each RPV material at a fluence level corresponding to the expected fluence at the end of plant operation; 40 years for the original license, or at 60 years if the license extension is granted. If a plant satisfies the criteria of 10 CFR 50.61, no further assessments are performed. As noted in Section 4.3.1, since there is no neutron irradiation during the period of long-term shutdown, no increase in the RT_{PTS} vales would be expected. Thus, in the U.S., if a plant did experience a long-term shutdown, there would be no requirement to assess its impact on RT_{PTS} values since those values would not change.

However, the approach to PTS in Japan is different than that utilized in the U.S. In general, the Japanese Codes and Standards require a full assessment of various PTS transients to ensure against non-ductile, or brittle, fracture for postulated PTS events. Essentially, the Japanese standards require plant-specific PTS assessments rather than using screening criteria developed from generic analyses like those in place in the U.S. Thus, a review of the JNO approach to performing PTS assessments in accordance with Japanese standards is detailed in the next section.

4.2.3.2 Assessment of JNO Approach to PTS

In terms of the fracture toughness values of the RPV materials which are an input to the PTS assessments for each plant, EPRI evaluated the K_{Ic} approach that the JNO Report and all Japanese utilities utilize for establishing the material fracture toughness curves in their plant-specific PTS analyses. This approach shifts the K_{Ic} curve to a lower bound value of toughness for 60 years of plant operation.

The K_{Ic} approach is established by Japanese Standard JEAC 4206-2007 [4.2-19]. This standard shifts the K_{Ic} curve, uniquely developed based on the Japanese fracture toughness data, using the results of the most limiting, or most highly embrittled, surveillance capsule test results. This approach is equivalent to the use of Adjusted Reference Temperature (ART) values in the development of P-T limits curves. There, the embrittled ART value of the limiting material is used in the K_{Ic} curve to shift the P-T limit curve to accommodate increased embrittlement effects (see Section 4.2.5 for further details).

The methodology for determining the temperature shift, T_P, for the K_{Ic} approach is defined in JEAC 4206-2007. The method considers the most limiting surveillance capsule results and includes a margin term, similar to the Regulatory Guide 1.99, Revision 2 methods. Japanese utilities, as outlined in the JNO Report, perform this evaluation using fluence values for 60 years of plant operation. A 10-mm deep flaw was postulated per JEAC 4206-2007 and the fluence levels were attenuated to this depth in the vessel wall.

This K_{Ic} offsetting approach for PTS development is similar to what would be performed in the U.S. if a plant needed to perform a full PTS assessment. It is always required to incorporate embrittlement into any RPV integrity calculation, and the approach employed by the JNO plants is deemed adequate and conservative. As discussed in Section 4.2.1 and concluded in Section 4.2.3.1, since there is no further irradiation embrittlement during the long-term shutdown period, the fracture toughness values utilized for the original 60-year PTS evaluations are acceptable, and should be identical to those used had the long-term shutdown not occurred.

Next, in terms of the transient selection, EPRI previously confirmed that the following transients are the most severe and must be analyzed:

1. A SBLOCA;
2. A LBLOCA; and
3. A main steam line break.

The transients selected by the JNOs are consistent with PTS events analyzed in the U.S. for both the original and Alternate PTS Rules. All these transients involve a severe overcooling on the RPV inner diameter (ID) from design operating temperature to ambient, followed by rapid re-pressurization while at low temperature, which induces high tensile stresses on the postulated RPV defects that can potentially lead to vessel failure. As noted above, after the original shutdown process was completed, no further excursions to either high pressure or temperature have occurred during the long-term shutdown. Thus, there is no potential for a PTS transient event to occur during this time and therefore no challenge to the integrity of the RPV.

EPRI further notes that it is not credible for a high-pressure PTS event to occur during a long-term shutdown because the RPV is neither at high pressure nor high temperature. Thus, since no pressurized transients at high temperature are possible, it is concluded that any PTS events are impossible to occur during this period.

In conclusion, there is no potential for either a reduction in the fracture toughness caused by neutron fluence nor the ability to have a pressurized thermal transient during the long-term shutdown period. Thus, EPRI concludes that all plants included in the JNO evaluation are not at risk for a PTS event during the long-term shutdown period and can safely be restarted consistent with the applicable codes and standards of Japan, as well as those utilized by the NRC.

4.2.4 Upper-Shelf Energy Reduction

10 CFR 50, Appendix G [4.2-12] provides the U.S. requirements for prevention against ductile failure in a NPP RPV, and specifies minimum USE requirements as follows:

> *"Reactor vessel beltline materials must have Charpy upper-shelf energy in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation or Director, Office of New Reactors, as appropriate, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code."*

Thus, in the U.S., all plant RPV ferritic materials with neutron fluence values greater than $1x10^{17}$ n/cm^2 (E > 1.0 MeV) must maintain Charpy USE greater than 50 ft-lb (67.8 J) during plant operation. If the USE limit cannot be met, a supplemental evaluation known as an Equivalent Margins Analysis (EMA) must be performed to justify continued safe operation [4.2-20 and 4.2-21]. Furthermore, the determination of USE must take into consideration the effects of embrittlement, utilizing the correlations documented in Regulatory Guide 1.99, Revision 2 [4.2-6] and the results of all available and relevant surveillance tests for a given material. A USE value determined using generic correlations is based on Regulatory Guide 1.99, Revision 2 Position 1.2, and a USE value calculated using surveillance data is based on Regulatory Guide 1.99, Revision 2 Position 2.2.

It was previously established that no further neutron embrittlement affecting either the ductile-to-brittle transition temperature or causing a reduction of USE is possible during the long-term shutdown period. EPRI agrees with the conclusions of the JNO Report in regard to USE and the need to perform an EMA, as necessary, if the plant has already shown a USE value below 50 ft-lb (67.8 J). Furthermore, USE evaluations are conducted to ensure against *ductile* failure of the RPV. In general, as shown on Figure 4-9, ductile failures on the upper-shelf of a material's Charpy energy impact curve will only occur at operating temperatures. Therefore, during the long-term shutdown period when the RPVs of the Japanese fleet are at ambient temperature and no pressurized transients can occur which have the potential to challenge the RPV wall integrity, it is not possible to have a ductile failure of the RPV. Thus, EPRI agrees with the conclusions of the JNO Report in regard to USE.

4.2.5 Pressure-Temperature Limit Curves

P-T limit curves are periodically developed on a plant-specific basis to define plant operation limits during plant heatup and cooldown conditions to protect the RPV from brittle fracture. This is accomplished by controlling the rates of temperature and pressure changes. As the RPV operates and experiences greater neutron irradiation and embrittlement, the P-T limits become increasingly restrictive to maintain the margins against brittle failure. Depending on the materials in the RPV, at end of life conditions, the limits may be so restrictive as to cause operational hardship. For that reason, the P-T limits used earlier in a plant's operating life are typically defined for the level of embrittlement expected in a shorter, near-term interval in the RPV's life – e.g., a specific number of EFPY or fluence period – instead of EOL fluence conditions. The P-T limit curves must therefore be updated periodically as the RPV fluence approaches the end of the assumed fluence interval. The P-T limits may also need updating whenever an RPV material surveillance capsule is pulled and tested and the data indicate an increase in the RT_{NDT}. Finally, in the U.S., the requirements for development of P-T limits are documented in 10 CFR 50, Appendix G [4.2-12] and ASME Code, Section XI, Appendix G [4.2-11].

EPRI recommends that the JNO Report describe that P-T limit curves provide further protection for RPV integrity because they are required to be validated or updated for use prior to each plant's return to operation. Consistent with prior conclusions made in this section, since no neutron embrittlement is predicted to occur during the long-term shutdown period, the P-T limit curves in use by each plant before the long-term shutdown should remain applicable for each plant in the Japanese fleet. In accordance with NRA requirements, the JNO utilities should review their plant-specific P-T limit curves, as necessary, by verifying that the actual EFPY of a given plant does not exceed the EFPY used for preparing their plant-specific P-T limit curves. Therefore, the JNOs consider that P-T limit curves are appropriately applied. As discussed in MRP-429 [4.2-22] for Takahama Units 1 and 2, EPRI concluded that P-T limit curves had been appropriately maintained. However, these conclusions for the JNO fleet should be confirmed in the JNO Report for completeness.

4.2.6 Conclusions

EPRI agrees that the JNO Report conclusions for the neutron embrittlement assessment are a reasonable and logical extension of the fact that no embrittlement effects from neutron fluence are expected during long-term shutdown because of the absence of a fission reaction in the reactor core. Furthermore, during the long-term shutdown, no pressurized thermal transients, at either high pressure, low temperature or both, will occur. Thus, no PTS events that challenge the integrity of the RPV at under brittle conditions, nor any events that could induce a ductile failure via an RPV rupture, are predicted to occur. Therefore, there are no consequential aging effects or damage that occurs to the RPV during the long-term shutdown. As such, there are no technical or safety impediments to the recovery of the long-term shutdown period by the JNOs regarding RPV integrity.

4.2.7 References

Note: References highlighted in yellow are available for purchase from EPRI; they are available at no charge to EPRI members.

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4.2-3. NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," December 2010.

4.2-4. NUREG-2192, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants," July 2017.

4.2-5. NUREG-2191, Volumes 1 and 2, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," July 2017.

4.2-6. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

4.2-7. 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

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4.2-11. ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure," 2017 Edition.

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4.2-16. *Materials Reliability Program: Reactor Pressure Vessel Integrity Primer (MRP-278, Revision 1): A Primer on Theory and Application.* EPRI, Palo Alto, CA: 2017. 3003007951.

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4.2-20. ASME B&PV Code, Section XI, Appendix K, "Assessment of Reactor Vessels with Low Upper Shelf Charpy Impact Energy Levels," 2017 Edition.

4.2-21. Regulatory Guide 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less than 50 Ft-Lb," June 1995.

4.2-22. *Materials Reliability Program: EPRI Review of the Kansai Takahama Units 1 and 2 Aging Evaluations for Extending Operational Periods (MRP-429).* EPRI, Palo Alto, CA: 2018. 3002012037.

4.3 Irradiation Assisted Stress Corrosion Cracking

Irradiation Assisted Stress Corrosion Cracking (IASCC) is the third of the four specific degradation phenomena that is evaluated in the JNO Report. Section 3.2.2.1(3) [4.3-1] reviews the effect of neutron irradiation on microstructure and mechanical properties of stainless steel core internals. Irradiation leads to formation of dislocation loop and solute clusters in the matrix and micro-segregation or depletion of solutes at the grain boundaries. These changes in microstructure result in increases in yield and tensile strengths and a decrease in ductility and fracture toughness of stainless steels. In addition, the material becomes susceptible to IASCC [4.3-1]. Therefore, IASCC must be addressed in the aging management technical evaluation. The IASCC susceptibility increases with fluence, and materials that were not susceptible in the non-irradiated state may develop both intergranular and transgranular cracks caused by IASCC.

The JNO Report also reviews actions to deal with factors contributing to IASCC in PWRs, particularly IASCC in baffle former bolts (BFBs). The results of IASCC research by the Japan Nuclear Energy Safety Organization (JNES) were incorporated in Japan Nuclear Safety Institute

(JANSI), "Guideline for Inspection and Evaluation of Core Internals and Atomic Energy Society of Japan, Code on Implementation and Review of Nuclear Power Plant Aging Management Program."

The JNO Report also describes actions to address IASCC in BWRs, including the core shroud, upper grid plate (top guide) and core support plate. The inspection and evaluation methodology for these components is described in a series of JANSI Guidelines for Inspection and Evaluation of Core Internals and Figure 3.2.2 (3)-12 of the report. The crack growth evaluation and integrity evaluation is performed in accordance with the specific procedure specified by the JSME Code on Fitness-for-Service for Nuclear Power Plants and the regulatory system of integrity evaluation (Reference 3-17 in the JNO Report). Inservice inspection is also performed according to the JSME Rules on Fitness-for-Service, which assures the integrity of components in combination with the above inspection and evaluation criteria.

From EPRI's perspective, the integrity evaluation approaches to address IASCC of core internals described in the JNO Report are technically adequate and similar to those used in U.S. PWRs and BWRs.

Section 3.2.2.1(3) of the JNO Report [4.3-1] asserts that long-term plant shutdown has no significant effects on the initiation and growth of SCC for the reactor coolant pressure boundaries and core internals when temperature decreases during plant shutdown below 100°C and when the reactor is not critical such that a neutron irradiation flux is not present. The JNO Report concludes that the possibility of a new SCC initiation in a cooling water environment below 100°C during a shutdown period and the growth rate of existing SCC cracks is sufficiently lower than that during typical operation (288°C or 550°F) and therefore such SCC can be accommodated by normal routine inspections after the resumption of plant operation. The JNO Report also notes that there is some risk of SCC due to unusual chloride containing atmospheres such that chloride-induced SCC could occur.

4.3.1 Discussion

For non-irradiated stainless steels, EPRI agrees with the JNO Report, based both on EPRI-sponsored research results documented in MRP-236 and MRP-352 [4.3-2 and 4.3-3] and other independent research reports by P. Andresen [4.3-4 and 4.3-5]. EPRI notes that irradiation due to neutron bombardment while a reactor is critical can cause the SCC cracking to be accelerated or aggravated (which is termed IASCC), compared to the same material and stress combination in an unirradiated environment. When a reactor is defueled and not operating (not critical), the neutron flux is not present; thus, the IASCC mechanism reverts to classic SCC. Therefore, EPRI concurs with the JNO Report conclusion that long-term plant shutdown has no effect on initiation/growth of IASCC in irradiated BWR and PWR internals.

Research by P. Andresen [4.3-4 and 4.3-5] concludes that a peak in crack growth rate in sensitized stainless steel occurs at ≈ 200°C in certain chloride-containing water chemistries, which typically is a factor of 30 to 100 times higher than at 288°C or at 25°C. Additionally, NRC Information Notice (IN)-2011-04 [4.3-6] concluded that austenitic stainless-steel piping is susceptible to transgranular (TG) SCC when tensile stresses are applied in a chloride environment where local temperatures exceed approximately 60°C (140°F).

4.3.2 Conclusion

EPRI concludes that neither SCC nor IASCC is relevant for reactor components exposed to low temperatures below 60°C (140°F) for long-term shutdowns. Also, during shutdowns while the reactor is not critical, there would be no neutron irradiation acceleration of SCC (i.e., IASCC) due to the absence of neutron fluence.

4.3.3 References

Note: References highlighted in yellow are available for purchase from EPRI; they are available at no charge to EPRI members.

4.3-1. Hokkaido Electric Power Co., Inc., et al., "Technical Report Regarding the Effects of Nuclear Power Plant Operating Period on Aging Degradation of Major Components/Structures," English Version, July 2018.

4.3-2. *Materials Reliability Program: Stress Corrosion Cracking of Stainless Steel Components in Primary Water Circuit Environments of Pressurized Water Reactors* (MRP-236, Revision 1). EPRI, Palo Alto, CA: 2017. 3002009967.

4.3-3. *Materials Reliability Program: Assessment of the Current Status and Completeness of Work on Inner and Outer Diameter Stress Corrosion Cracking of Austenitic Stainless Steels in PWR Plants* (MRP-352). EPRI, Palo Alto, CA: 2013. 3002000135.

4.3-4. "Effects of Temperature on Crack Growth Rate in Sensitized Type 304 Stainless Steel and Alloy 600," Peter L. Andresen, CORROSION, September 1993, Vol. 49, No. 9, pp. 714-725

4.3-5. "Effects of Temperature and Corrosion Potential on SCC," Peter L. Andresen and Russell A. Seeman, 15th International Conference on Environmental Degradation, Edited by: Jeremy T. Busby, Gabriel Ilevbare, and Peter L. Andresen, TMS (The Minerals, Metals & Materials Society), 2011.

4.3-6. NRC Information Notice (IN)-2011-04, February 23, 2011, "Contaminants and Stagnant Conditions Affecting Stress Corrosion Cracking In Stainless Steel Piping in Pressurized Water Reactors," NRC ADAMS Accession No. ML103410363.

4.4 Thermal Aging of Cast Austenitic Stainless Steel

Thermal aging of cast austenitic stainless steel (CASS) is the fourth of the four specific degradation phenomena that is evaluated in the JNO Report [4.4-1]. Section 3.2.2.1(4) of the JNO Report asserts that long-term plant shutdown has no significant effects associated with thermal aging of components made with CASS materials. The JNO Report concludes that thermal aging of CASS is expected to occur in components whose service temperature is 250°C or higher, and notes that the reactor coolant temperature during plant shutdown is below 100°C. Therefore, no effects of plant long-term operation on the thermal aging of CASS are expected.

4.4.1 Discussion

In general, it is known that microstructural changes may occur in CASS materials during exposures to high temperatures. These changes can alter mechanical properties and have been shown to lead to reductions in fracture toughness and ductility. The potential significance and management of the effects of thermal aging embrittlement for CASS components was one of the open technical issues for a number of the License Renewal Industry Reports (IRs), and significant materials research into this effect has been completed over many years [4.4-2 and 4.4-3].

In the mid-1990s, EPRI identified screening criteria for evaluating the potential significance of thermal aging embrittlement effects for CASS components for the license renewal term [4.4-4]. The research evaluated the extensive set of fracture toughness data available in the literature for thermally-aged CASS materials, and the categorization of those data as a function of delta ferrite and molybdenum contents, casting method, and duration of aging at various temperatures. This research provided the basis for resolution of an important license renewal technical issue, namely, the adequacy of current programs for inspecting and evaluating the effects of thermal aging embrittlement for LWR reactor coolant system primary pressure boundary and reactor internals CASS components.

The EPRI research determined that the magnitude of the reduction in fracture toughness properties depends upon the casting method used to make the component, the material chemical composition, and the duration of exposure at operating temperatures conducive to the embrittlement process. Static castings are more susceptible than centrifugal castings, high-molybdenum-content castings are more susceptible than low-molybdenum-content castings, and high delta-ferrite castings are more susceptible than low delta-ferrite castings. The rate of embrittlement is temperature dependent with operating temperatures on the order of 320°C (608°F) increasing the embrittlement rate relative to the rate occurring at operating temperatures on the order of 285°C (545°F).

In general, thermal aging effects below these temperatures were not considered significant regardless of the duration of exposure. Specifically, the research conclusions include:

- Low-molybdenum (e.g., SA 351 Grade CF-3 and CF-8) material that is centrifugally cast is not subject to significant thermal aging embrittlement at service temperatures up to 320°C (608°F) and for service times up to 525,000 hours (60 years).
- Low-molybdenum material that has been cast statically and has delta ferrite content equal to or less than 20% (as estimated by either calculation or measurement) is not subject to significant thermal aging embrittlement at service temperatures up to 320°C (608°F) and for service times up to 525,000 hours (60 years).
- High-molybdenum (e.g., SA 351 Grade CF-3M and CF-8M) material that is centrifugally cast and has a delta ferrite content equal to or less than 20% (as estimated by either calculation or measurement) is not subject to significant thermal embrittlement at service temperatures up to 320°C (608°F) and for service times up to 525,000 hours (60 years).

In parallel with these efforts, Argonne National Laboratory (ANL) developed approaches for conservatively predicting the fracture toughness behavior of thermally aged CASS based on material chemistry information and/or service history [4.4-5 and 4.4-6]. These correlations were developed from 80 different compositions of CASS which were aged up to 58,000 hours at 350°C (662°F).

The NRC evaluated this research as a technical basis for aging management programs for utility license renewal applications [4.4-7]. The NRC technical assessment of this approach accepted the proposed screening criteria in the EPRI report. The NRC concluded that these are applicable to all primary pressure boundary and reactor vessel internal (RVI) components constructed from SA-351 Grades CF3, CF3A, CF8, CF8A, CF3M, CF3MA, or CF8M. Additionally, the NRC conservatively applied the criteria to all service conditions above 250°C (482°F). Finally, the BWRVIP also evaluated thermal and irradiation embrittlement of CASS materials to support screening for license renewal. A publicly available report documenting the results of this research is included in BWRVIP-234NP-A [4.4-8].

4.4.2 Conclusion

Based on the existing body of research, for plant components where temperature exposure is below 250°C (482°F), no reduction in fracture toughness is expected. The aging effect of thermal embrittlement at lower temperatures is not considered relevant and long-term aging effects below 250°C (482°F) do not need to be considered for CASS components. For reactor internals, there is no neutron fluence accumulation during shutdown periods and thus no impact on aging. Therefore, there is no aging effect associated with thermal aging of CASS materials during a long-term shutdown.

4.4.3 References

4.4-1. Hokkaido Electric Power Co., Inc., et al., "Technical Report Regarding the Effects of Nuclear Power Plant Operating Period on Aging Degradation of Major Components/Structures," English Version, July 2018.

4.4-2. "PWR Reactor Coolant System License Renewal Industry Report, Revision 1," Report No. 90-07 Nuclear Management and Resources Council (NUMARC), Washington, DC, May 1992; also, EPRI Report No. TR-103844s, July 1994.

4.4-3. "BWR Primary Coolant Pressure Boundary License Renewal Industry Report, Revision 1," Report No. 90-09, Nuclear Management and Resources Council (NUMARC), Washington, DC, April 1992; also, EPRI Report No. TR-103843s, July 1994.

4.4-4. *Evaluation of Thermal Aging Embrittlement for Cast Austenitic Stainless Steel Components in LWR Reactor Coolant Systems.* EPRI, Palo Alto, CA: 1997. TR-106092.

4.4-5. O.K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG/CR-4513, Revision 1, August 1994.

4.4-6. O.K. Chopra and W.J. Shack, "Assessment of Thermal Embrittlement of Cast Stainless Steels," NUREG/CR-6177, May 1994.

4.4-7. NRC License Renewal Issue No. 98-0030, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components," 5/29/2000, NRC ADAMS Accession No. ML003717179.

4.4-8. *BWRVIP-234NP-A: BWR Vessel and Internals Project, Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steels for BWR Internals.* EPRI, Palo Alto, CA, November 2017, 3002010550NP, NRC ADAMS Accession No. ML18040A505.

4.5 Overall Conclusions

Based on EPRI's review of the JNO Report approach, logic, and associated conclusions for these four specific degradation phenomena, EPRI generally agrees that these phenomena may be addressed by actions associated with aging management, and that these phenomena will not progress significantly during long-term shutdown.

5
REVIEW OF DEGRADATION OF ELECTRICAL INSTRUMENTATION EQUIPMENT

This section documents EPRI's review of the logic employed in the JNO Report, Section 3.2.2.2 [5-1], in which they evaluated the aging effects on cables during plant operation and qualification of cables to assure they can perform their safety function at any time during their qualified life. Additionally, EPRI reviewed Section 3.3.1 of the JNO Report which further evaluated how conditions during the long-term shutdown affected cable insulation degradation and qualified life.

5.1 Cable Aging Logic Review and Evaluation

As discussed in Section 3.2.2.2 of the JNO Report [5-1], cable electrical insulation integrity must be maintained to prevent cable failure due to loss of insulation resistance (dielectric strength) and to contain the applied voltage stress that occurs during normal operation. In addition to the voltage stress, cables are designed to withstand the thermal stress that results from the current they must supply to the end device (for example, a motor, transmitter, relay, etc.). Thermal stress also occurs from the ambient temperature to which the cable is exposed. Power plant ambient temperatures can vary from temperature and humidity controlled locations (like the main control room), to high temperature and humidity caused by operating equipment (steam turbines, large motors, etc.), steam lines, etc. Near the reactor vessel, radiation exposure, even though at low levels, can also contribute to the degradation of cable insulation over the long operating life of the plant.

EPRI Report 3002010641 [5-2] contains low voltage cable aging management guidance and identifies the primary cause of cable insulation degradation for low voltage and instrument and control cables is oxidation resulting from thermal stress. As stated above, thermal stress can result from the current the cable conductor is designed to carry (the cable ampacity rating) and the ambient environment where the cable is located. Typically, cables in nuclear power plants are not operated to the full extent that their design allows which varies by insulation types from 75°C (167°F) to 90°C (194°F) for most of cables used in NPPs. Except in rare cases where cables operate at their designed ampacity rating, ambient temperature becomes the main source of cable aging. EPRI's interactions with its membership, such as reviewing of laboratory evaluations of cables removed from NPPs, has shown ambient operating temperature to be the main cause of insulation degradation.

Radiation has been shown by research studies to have a damaging effect on cable insulation. However, EPRI Report 3002010404 [5-3] collected data to evaluate temperature and radiation levels in the areas where cables are routed near the reactor vessel that shows low levels of radiation compared to qualification levels.

The JNO Report discusses long-term integrity testing (environmental qualification testing, or qualification testing) that has been performed by its members to assure that thermal and radiation aging do not cause the loss of function of cable insulation even if exposed to the steam, temperature, radiation and pressure of a severe accident. The qualified life was evaluated using two types of test methodologies; one is recommended by IEE Japan for sequential aging and the other is called the ACA methodology specified by then-JNES (currently NRA) for simultaneous aging. The first test performed sequential, accelerated thermal and radiation aging to bound the aging that would occur during the qualified life period. This can be seen in Table 5-1 for Takahama Unit 1 as an example of flame retardant PH cables for a PWR and Table 5-2 for the same cable type used at Tokai 2 as an example of a BWR. In addition to life time dose, the cables are also irradiated to a dose representative of a severe accident and then subjected to simulated accident steam, pressure, and thermal levels to bound the design basis accident levels. To judge if the cable is qualified, it must pass an alternating current voltage withstand test which is performed following the completion of each design basis accident (DBA) test. The cable must not fail when energized to 3,200 volts/mm of insulation thickness for five minutes to determine if the cable is qualified. All cables needed to safely operate the plant in a safe condition after postulated accidents must be qualified in a manner like these examples. Note that for Tokai 2, because of the higher plant thermal and radiation conditions during normal operation, the qualified life that resulted is a much smaller qualified life when compared to a similar cable type for Takahama Unit 1.

Table 5-1
Sequential Aging Long-Term Integrity Test Requirements for Flame Retardant PH Cables at Takahama Unit 1

Test	Test Condition	IEE Japan Requirement
Accelerated thermal ageing	140°C -9 days	The test condition envelops the condition, "124°C -9 days", which is converted from the actual environmental condition assuming a general environment at 65°C *1 for 60 years for cables of Takahama Unit 1.
Irradiation (normal operation + DBA/severe accident)	500 kGy+1,500 kGy	The test condition envelops the total cumulative dose of 760 kGy at Takahama Unit 1 adding cumulative doses in the event of DBA at 607 kGy to doses equivalent to the period of normal operation at 153 kGy *2. The test condition also envelops the total cumulative dose of 653 kGy adding cumulative doses in the event of DBA at 500 kGy to doses equivalent to the period of normal operation at 153 kGy *2.
Exposure to DBA/severe accident	Maximum temperature : 190°C Maximum pressure : 0.41 MPa [gage]	The test condition envelops the maximum temperature of approx. 122°C and maximum pressure of approx. 0.26 MPa [gage] in the event of a design basis accident as well as the maximum temperature of approx. 138°C and maximum pressure of approx. 0.305 MPa [gage] in the event of a severe accident.

*1 This temperature has been set by adding an increase due to energization and some margins to the ambient temperature (approx. 48°C) surrounding cables inside the containment vessel.

*2 0.29 [Gy/h] × (24×365.25) [h/y] ×60[y]=153 kGy

Table 5-2
Sequential Aging Long-Term Integrity Test Requirements for Flame Retardant PH Cables at Tokai 2

Test	Conditions	Description
Accelerated thermal ageing	121°C ×126 hours (control cables) 121°C ×251 hours (other cables)	The test condition envelops a normal operating period of 15 years for flame retardant PN control cables and 30 years*1 for other flame retardant PN cables at the maximum ambient temperature of 65.6°C inside the containment vessel.
Irradiation (normal operation + DBA/severe accident)	Radiation dose : 1,175 kGy	The test condition envelops the assumed dose at Tokai 2 of approx. 326 kGy (adding the maximum cumulative doses in the event of DBA at 2.6×10^{2} kGy to doses equivalent to the period of 30-years normal operation at approx. 66 kGy). The test condition also envelops the assumed dose at Tokai 2 of approx. 706 kGy (adding the maximum cumulative doses in the event of DBA at 640 kGy to doses equivalent to the period of 30-years normal operation at approx. 66 kGy).
Exposure to DBA/severe accident	Maximum temperature : 235°C Maximum pressure : 0.62 MPa [gage]	The test condition envelops the maximum temperature of 171°C and maximum pressure of 0.31 MPa in case of a DBA as well as maximum temperature of 235°C and maximum pressure of 0.62 MPa [gage] in the event of a severe accident.

*1 The operating period has been set considering the service period after cable replacement.

The second qualification test discussed in the JNO Report replaces the sequential, accelerated aging for the lifetime thermal and radiation with simultaneous thermal and radiation aging. Research has shown that for some material simultaneous aging can cause greater insulation degradation than sequential aging. Additionally, this aging is done at lower dose rates and temperature levels to better approximate how cables naturally age in operating plants. Examples were provided (see Table 5-3 and Table 5-4 below) for the flame retardant PH cables used at Takahama Unit 1 and Tokai 2 as representative of the respective PWR and BWR reactor designs.

Table 5-3
Simultaneous Aging Long-Term Integrity Test for Flame Retardant PH Cables for Takahama Unit 1

Test	Condition	Description
Accelerated simultaneous thermal and radiation ageing	100°C -94.8 Gy/h—4,003 hours	The test condition envelops 60-years normal operation given the environmental conditions inside the containment vessel, the time of cable replacement and long-term maintenance management policies for Takahama Unit 1 according to the superposition of time-dependent data technique based on the test results obtained from "ACA project".
Irradiation (DBA)	1,500 kGy	The test condition envelops the maximum cumulative dose of 607 kGy assumed at Takahama Unit 1 in case of a DBA.
Exposure to DBA/severe accident	Maximum temperature : 190°C Maximum pressure : 0.41 MPa [gage]	The test condition envelops the maximum temperature of approx. 122°C and maximum pressure of approx. 0.26 MPa [gage] assumed at Takahama Unit 1 in case of a DBA.

Table 5-4
Simultaneous Aging Long-Term Integrity Test Conditions for Flame Retardant PH Cables for Tokai 2

Test	Condition	Description
Accelerated simultaneous thermal and radiation ageing	100°C -94.7 Gy/h—6,990 hours	The test condition envelops 28-years normal operation *1, which has been obtained by extrapolating the test results obtained from "ACA project" to the environmental condition inside CV at Tokai 2 based on the technique of the superposition of time dependent data.
Irradiation (DBA)	Radiation dose : 500 kGy	The test condition envelops the maximum cumulative dose of 2.6×10^2 kGy assumed at Tokai 2 in case of a DBA.
Exposure to DBA/severe accident	Maximum temperature : 171°C Maximum pressure : 0.427 MPa [gage]	The test condition envelops the maximum temperature of 171°C and maximum pressure of 0.31 MPa [gage] assumed at Tokai 2 in case of a DBA.

*1 The operating period has been set considering the service period after cable replacement.

5.1.1 Evaluation of the JNO Methodology

The Japan requirements for environmental qualification are contained in the IEE Japan and ACA guide. This guidance was compared to that contained in IEEE 323-1974 [5-4] and IEEE 383-1974 [5-5] which are the standards endorsed by the NRC for cable environment qualification in Regulatory Guide 1.89 [5-6]. The IEE Japan requirements are in alignment with the IEEE standard requirements. The one difference between the IEE Japan requirements and the IEEE standards [5-4, 5-5] is that only sequential, accelerated thermal and radiation aging have been used historically to environmentally qualify cables. The use of simultaneous, accelerated aging required by the ACA guide is one instance where the JNO cables were subjected to additional testing. Theoretically, simultaneous accelerated aging is a more severe test than is required by the IEEE standards [5-4, 5-5] or required by the NRC for cables in U.S. NPPs. Also, the performance of this additional cable qualification test was typically not required by other international EPRI members outside of Japan.

In reviewing the actual thermal and radiation aging in the sequential aging tests for Takahama Unit 1 and Tokai 2, the test values used are more severe than design temperatures and radiation levels required by the IEE Japan standards. All the values used for Japan qualification testing include conservatism in each value of thermal aging (lifetime and accident), radiation aging (lifetime and accident) and pressure (accident). The DBA accident temperature is 8°C greater than required, and the radiation dose and the accident pressure used are greater than the 10%

margin for pressure, radiation, and voltage required by the IEEE 323 standard [5-4]. Qualification of U.S. cables typically did not exceed the 10% margin. This is another case where the test conditions used in the Japan qualification tests have added conservatism. The cumulative conservatism provides additional assurance the cables that pass the Japan qualification tests will be able to perform their design function under the most severe accident conditions. As such, the qualified life obtained is a conservative value of the time that the cable could perform its design function under DBA conditions even at the end of its qualified life.

Part 2 of Section 3.2.2.2 of the JNO Report discusses additional maintenance activities that are performed by some, but not all, JNO members on the cable insulation to monitor degradation. Tests performed by some JNO members to evaluate cable degradation may include dielectric insulation testing (tan delta testing) for the applicable 6.6 kV outdoor, medium voltage cables that are subject to being submerged, insulation resistance testing of low voltage power cables (< 600 volts), and periodic functional checks of instrument and control cables (< 400 volts) during periodic inspections. These are typical tasks performed by other EPRI members in their cable aging management programs to detect cable insulation degradation. The EPRI cable program aging management guides [5-2, 5-7] recommends periodic visual inspections, insulation resistance testing (for low voltage and unshielded medium voltage cables exposed to wet environments) and very low frequency, tangent delta (dissipation factor) testing. Finally, it was noted that JNO members replace cables prior to the end of their actual qualified life to further ensure that the cables will perform when required.

5.1.2 Summary of Cable Aging Logic Review

Based on the review of Section 3.2.2.2 of the JNO Report, the JNO members perform qualification testing using test parameters above both Japan and U.S. standard requirements. This provides conservatism that the qualified cables will perform their design safety function until the end of their qualified life, or determine if they need replacement prior to the end of that qualified life. The maintenance practices performed by some JNO members are in alignment with EPRI cable aging management guidance and the maintenance practices of EPRI members who have cable aging management programs developed based on the EPRI guidance documents for low and medium voltage cables [5-2, 5-7].

5.2 Review of Aging Effects for Long-Term Shutdown Logic

5.2.1 Thermal and Radiation Aging During the Longer-Term Shutdown

Section 3.3.1 of the JNO Report provides the logic that JNO plants developed regarding plant conditions during the long-term shutdown compared to plant conditions during power operation to explain how milder plant conditions affect cable qualified life. Table 5-5 (Table 3.3.1-1 from the JNO Report) shows the comparison between temperature and dose rates during normal plant operation and during plant shutdown conditions.

Table 5-5
Environmental Conditions in Operation and During Shutdown

Plant	Area	Environmental condition at actual unit			
		Temperature [°C]		Dose rate [Gy/h]	
		Normal operation	Shutdown *1	Normal operation	Shutdown
Takahama Unit 1	Inside CV (loop equipment area)	50	24	0.0130	0.001 or less *3
	Outside CV (MS area *2)	40	24	0.0009	-*4
Takahama Unit 2	Inside CV (loop equipment area)	48	24	0.0202	0.001 or less*3
	Outside CV (MS area *2)	40	24	0.0009	-*4
Mihama Unit 3	Inside CV (loop equipment area)	31	24	0.3882	0.001 or less*3
	Outside CV (MS area *2)	40	24	0.0013	-*4
Tokai 2	Inside CV	65.6	25	0.250	0.001 or less*3
	Inside reactor building	40	25	0.00015	-*4

*1 A value adding some margins to the average of CV temperature indications during shutdown

*2 Main steam piping/main feedwater piping intermediate building area and main steam piping diesel generator building area

*3 0.001Gy/h is a conservative value considering environmental dose equivalent rates in the adjacent areas during refueling outage inspection.

*4 It has been determined that there are no effects of irradiation since dose rates are expected to be very low as this area is located outside CV.

Table 5-5 shows the significant difference in both temperature levels between operating and shutdown values. The delta between the test condition values for thermal and radiation aging and the normal operating plant values again highlight the range of conservatism that was included in the qualification test. The JNO Report considers the effect on cable insulation degradation for a postulated 10-year shutdown period where the cables have been in a thermal environment of 24-25°C versus the operating temperatures. The Tokai 2 operating ambient temperature and the Tokai 2 and Mihama Unit 3 operating radiation levels are the only outliers, but those operating, temperature and radiation level differences were taken into account in the cable qualifications that were performed. An example for this is the Tokai 2 cables which were qualified to a 15 or 30-year qualified life instead of the 60 years used in other cable qualifications.

While some natural aging occurs on the cable insulation at shutdown thermal conditions, that is offset by the conservatism added to the qualification tests or reduced qualified life.

In the case of radiation aging, the dose levels during shutdown are so insignificant that little if any radiolysis aging of the insulation could occur. Combining this with the conservatism of the qualification test based on the dose rates used and total lifetime dose absorbed by the cable insulation, it is assured that no appreciable cable qualified life, or aging, occurs while the plant is shutdown.

5.2.2 Conclusion of Review Aging Effects for Long-Term Shutdown Period

The JNO evaluation logic that thermal and radiation conditions during long-term shutdown results in minimal aging of cable insulation materials is correct. This conclusion is based not only on the operating temperatures and radiation levels the cables are exposed to over the expected 10-year plant shutdown, but also the methods used to obtain qualified life are conservative and underestimate qualified life because of that conservatism. The items below summarize the conservativeness of the qualified life:

- Plant ambient temperature operating conditions in all but Tokai 2 (which is 65.6°C) containments typically do not exceed 50°C. Tokai 2 adjusted for the higher ambient temperatures by limiting the qualified life obtained in their qualification test to account for the actual plant operating temperature conditions.
- The temperature and radiation values used in the PWR plants' (Takahama Units 1 and 2 and Mihama Unit 3) accelerated aging were above the IEE Japan recommended values. Those IEE Japan values are conservative, when compared to typical operating temperatures seen in EPRI research [5-3] that the cables are exposed to inservice, so using even higher values increases the conservatism of the tests performed.
- Cables at Tokai 2 operate in higher containment temperatures and radiation levels during operation but they were qualified for a shorter life that more than offsets the more severe operating conditions.
- IEE Japan qualification testing requires both sequential thermal and radiation aging. A second qualification was performed using the ACA guide. The ACA requires simultaneous thermal and radiation aging. Simultaneous aging is considered to be a more severe test and is an additional test beyond environmental qualifications that are typically done outside of Japan.
- Proper maintenance testing, inspections and replacement practices ensure the cables will perform their safety function for the operating period for which they are installed in the plant.

5.3 References

Note: References highlighted in yellow are available for purchase from EPRI; they are available at no charge to EPRI members.

5-1. Hokkaido Electric Power Co., Inc., et al., "Technical Report Regarding the Effects of Nuclear Power Plant Operating Period on Aging Degradation of Major Components/Structures," English Version, July 2018.

5-2. *Low-Voltage and Instrumentation and Control Cable Aging Management Guide, Revision 1.* EPRI, Palo Alto, CA: 2017. 3002010641.

5-3. *Results of Radiation and Temperature Monitors: Research for Installed Cables at US Nuclear Power Plants in Support of Long Term Operation.* EPRI, Palo Alto, CA: 2013. 3002010404.

5-4. Institute of Electrical and Electronics Engineers, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," IEEE Std. 323-1974.

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6 REVIEW OF DEGRADATION OF CONCRETE

This section documents the EPRI's review of the JNO Report [6-1] logic that the major factors associated with the potential reduction in the strength of concrete during the long-term shutdown period are carbonation (neutralization) and chloride penetration.

The review focuses on the approach used to conclude that the evaluation results have sufficient margins, that plant operating life is not affected, and that these evaluations are technically valid in demonstrating the effects of a prolonged shutdown period do not challenge the plant operating life.

Specifically, EPRI reviewed the approach used for management of major aging phenomena assumed in concrete structures and the potential effect of long-term plant shutdown on aging degradation. The EPRI review is summarized below based on each degradation mechanism of concern.

6.1 Carbonation Induced Corrosion

Carbonation in concrete occurs when calcium hydroxide within the cement paste reacts with the carbon dioxide from the environment. When the reaction occurs, the alkalinity of the concrete decreases to a level, typically below pH 12, disrupting the passive layer of the steel reinforcement and promoting corrosion of the steel reinforcement. The corrosion process due to carbonation occurs at a much slower rate when compared to the corrosion due to chlorides. It is generally accepted that deterioration due to carbonation will begin when the carbonation front reaches the depth of the reinforcement. At this point the presence of both water and oxygen is required for corrosion to occur. However, the rate at which subsequent corrosion occurs is uncertain. Therefore, EPRI recommends a structure-specific evaluation be performed when carbonation has reached the depth of reinforcement to assess the condition of the structure. Note that carbonation only affects concrete strength after it reaches reinforcement depth, the reinforcement corrodes, expands and the concrete cracks. The two main factors contributing to carbonation induced corrosion are exposure to atmospheric carbon dioxide and the concrete cover depth of reinforcement.

The JNO Report indicates that visual inspections are performed on a regular basis to identify symptoms of corrosion. Additionally, core samples are extracted to measure the depth of carbonation from the exterior surface of the concrete and modeling of the carbonation front is performed. A recent study performed by EPRI using data obtained from three structures in the Takahama Unit 1 nuclear power plant showed that carbonation induced corrosion was not a concern for the structures evaluated [6-2]. The models used for that study assume an un-cracked condition, design cover depth, and concrete carbonation data from three different structures.

The effects of carbonation can be monitored by measuring the depth of carbonation periodically and by performing visual inspections for signs of corrosion. Visual inspections will identify symptoms once reinforcement corrosion has initiated.

EPRI performed an assessment of the carbonation penetration in three different structures at Takahama Unit 1 for 60 years of operation [6-2]. Table 3.3.2-3 of the JNO Report summarizes carbonation studies at Takahama Unit 1, Takahama Unit 2, Mihama Unit 3 and Tokai 2 showing significant margins. The same assessment method was used in the JNO Report for all four units. An analysis was not performed by EPRI for Mihama Unit 3 and Tokai 2, but for the structures in Takahama Units 1 and 2, the EPRI analysis confirms the JNO assessment that carbonation is not a concern for 60 years of operation. Based on these results, EPRI agrees with the JNO report that in the case of Takahama Units 1 and 2, there is a sufficient margin for 70 years of operation regarding carbonation.

During a long-term shutdown, reinforced concrete structures will continue to be exposed to carbon dioxide from the environment. The exposure to carbon dioxide is completely independent of the operation of the plant and the carbonation front will continue to advance during shutdown. Routine visual inspections can identify symptoms of reinforcement corrosion. While the degradation mechanism continues during plant shutdown, its effects can be managed through continued maintenance and aging management efforts. Therefore, based on the evaluations performed and assuming the maintenance and aging management efforts are followed, EPRI agrees that the effects of carbonation over the long-term shutdown can be adequately managed.

6.2 Chloride-Induced Corrosion

Corrosion of reinforced concrete due to chloride penetration occurs when a structure is exposed to a chloride environment and the chloride concentration at the level of the reinforcement reaches a threshold value that allows the initiation and propagation of corrosion of the steel reinforcement. Chloride-induced corrosion is also accelerated by the presence of oxygen and water. The corrosion process will initiate as pitting corrosion generating corrosion products and loss of cross section of the steel reinforcement. If there is sufficient cross section loss of the steel reinforcement, the overall structural performance of the structure can be compromised. In general, once chloride induced corrosion is identified through delamination, cracking, or corrosion staining, the propagation of deterioration is fast and will typically require the removal and replacement of affected concrete.

The JNO Report indicates that visual inspections are performed on a regular basis to identify symptoms of corrosion. Additionally, core sample extraction, determination of chloride levels, and modeling for chloride ingress are part of the aging management technical evaluation performed by the plants. The report presents data on core samples extracted from four Japanese nuclear plants after 37 to 40 years of operation to measure the chloride concentration profile. The chloride concentrations at the level of the reinforcement range between 0.04% to 0.05% per unit weight of concrete.

During a long-term shutdown, reinforced concrete structures will continue to be exposed to chlorides. The exposure to chlorides is completely independent of the operation of the plant and the chloride concentration in the concrete will continue to increase with time. Routine visual inspection can identify symptoms of reinforcement corrosion. While the degradation mechanism continues during plant shutdown, its effects can be managed through continued maintenance and aging management efforts. Therefore, based on the evaluations performed and assuming the maintenance and aging management efforts are followed, EPRI agrees that the effects of chloride-induced corrosion over the long-term shutdown can be adequately managed.

6.3 Alkali-Silica Reaction

Alkali-silica reaction (ASR) in concrete structures occurs when alkalis present in the cement paste react with amorphous silica present in the aggregates and moisture present in the paste. This reaction results in the formation of a gel that swells as it absorbs water. Swelling of the gel exerts internal pressure in the concrete that causes cracks due to its low tensile strength. The effect of alkali-silica reaction in concrete is a decrease in mechanical properties, especially elastic modulus, and an increase in deformation which can impact the functionality of equipment attached to concrete. Several EPRI reports [6-3 through 6-6] address the degradation, inspection, repair and management of ASR affected structures. Because ASR can be detected after many years of operation, EPRI report 3002005389 [6-7] provides guidance on available tools for early detection of ASR.

ASR can manifest even after many years of operation. Because ASR aging effects can include deformation of concrete structures, the first symptoms of ASR in concrete may be equipment misalignments, closing of seismic gaps, etc. These effects are normally not evaluated by civil engineering personnel. If the potential for ASR exists, monitoring for symptoms of ASR is recommended. A method suggested by the Architectural Institute of Japan (AIJ) [6-8] is to monitor cracking during visual inspection for ASR degradation, which is consistent with the revised ACI349.3R guideline [6-9].

During a long-term shutdown reinforced concrete structures could be susceptible to ASR as long as they have moderately reactive aggregates and accessibility to water. ASR in concrete is completely independent of the operation of the plant. Routine visual inspection can identify symptoms of ASR. Note that, although ASR degradation, if present, can continue during plant shutdown, its effects can be managed through continued maintenance and aging management efforts.

6.4 Freeze-thaw

Freeze-thaw in concrete structures occurs when water in the capillary pores freezes due to outside low temperatures and expands generating cracking due to the low tensile strength of concrete. Cracking usually accelerates the ingress of deleterious substances that can corrode the reinforcement. Freeze-thaw occurs in areas exposed to cyclic freezing temperatures. The AIJ guidelines recommend inspecting for freeze-thaw only in susceptible locations.

During a long-term shutdown, reinforced concrete structures will continue to be susceptible to freeze-thaw in plants located in cold climates. While the degradation mechanism, if present, may continue during plant shutdown, its effects can be managed through continued maintenance and aging management efforts.

6.5 Reduced Strength Due to Irradiation

Neutron and gamma radiation are absorbed and attenuated by the concrete biological shield, which is a massive concrete structure surrounding the reactor pressure vessel in nuclear power plants. Energetic neutrons, which originate in the reactor core, pass through the core internals and reactor pressure vessel and interact with the concrete in the biological shield. Over time, this can cause a reduction of the strength of the concrete and can cause swelling of the affected concrete near the inner diameter of the biological shield due to the agglomeration of point defects (vacancies) in the aggregates. Such swelling is more pronounced in silica-based aggregates than calcium-based aggregates. The reference limit used in Japan for neutron fluence in concrete biological shields, beyond which there may be a change in concrete mechanical properties and local swelling, is 10^{20} neutrons/cm^2 [6-10]. Gamma irradiation originating from the core and in the concrete due to neutron capture can cause radiolysis of water and internal heating, resulting in dehydration of the concrete which can affect the mechanical properties. The reference limit used in Japan for gamma dose in concrete biological shields, beyond which there can be a change of mechanical properties, is currently 2×10^8 Gy [6-10].

In Japan, it was determined that at Takahama Unit 1, a PWR, the maximum neutron fluence in the concrete biological shield at 60 years of operation will be 4.43×10^{19} n/cm^2 (E > 0.1 MeV), which is well below the threshold used in Japan for neutrons. The 60-year gamma dose in the Takahama Unit 1 concrete was evaluated to be 2.31×10^8 Gy, which is above the threshold used in Japan for gamma irradiation in the concrete near the inner diameter of the concrete biological shield. EPRI concurs with the assessment that the volume of concrete potentially affected by gamma rays in the Takahama Unit 1 concrete biological shield is small relative to the overall volume of the structure. Furthermore, recent work by Maruyama sponsored by the NRA shows that there is no effect of gamma radiation on concrete properties up to 2.3×10^8 Gy [6-10]. Tokai 2, which is a BWR, is expected to experience a maximum neutron fluence in the concrete biological shield of 4.1×10^{15} n/cm^2 (E > 0.1 MeV) and maximum gamma dose in the concrete of 7.8×10^4 Gy. Both values are well below the threshold limits. In 2015, destructive testing of concrete cores extracted from the regions of the biological shields at Takahama Unit 1 and Tokai 2 that experience the highest radiation loads showed that the compressive strength values in the region are above the design values of both plants. In addition, in 2016 a core sample was taken from the concrete reactor vessel support pedestal at Tokai 2 which showed that the compressive strength exceeds the design value. Note that core sampling and mechanical testing of the concrete biological shield is not done as a part of aging management in the U.S. and there are no U.S. regulations that require such testing as a part of license extension applications.

During a long-term shutdown, concrete irradiation damage is not expected to progress as a function of time as the fuel is removed from the reactor pressure vessel and there is no fission.

6.6 Reduced Strength Due to Elevated Temperature

Elevated temperature exposure of the concrete biological shield has the potential to cause degradation and change the mechanical properties of the concrete. The primary mechanism for this at intermediate temperatures is dehydration of the concrete by evaporation of non-chemically bound water. Heating of the biological shield concrete occurs by two heat sources – the reactor vessel and gamma radiation. The Architectural Institute of Japan has established limits of 65°C for structures and 90°C for local areas in concrete structures [6-11]. According to the Japan Society of Mechanical Engineers, dissipation of hydrated water contained in the concrete starts near 190°C, causing a degradation in mechanical properties [6-12]. The maximum temperatures estimated in the concrete biological shield at Takahama Unit 1 and Tokai 2 were below the threshold limits. For further reference, a generic analysis of heating of a PWR concrete biological shield due to gamma radiation was published by EPRI [6-13].

Core samples drilled from the concrete biological shields at a PWR (Takahama Unit 1) and BWR (Tokai 2) that are subject to elevated temperatures were mechanically tested. The results show that both cores exceeded their design strength at 40 and 38 years for Takahama Unit 1 and Tokai 2, respectively. As such, it is apparent that reduced concrete strength due to elevated temperatures is not an active degradation mechanism in the two units. Note that core sampling and mechanical testing of the concrete biological shield is not done as a part of aging management in the U.S. and there are no U.S. regulations that require such testing as a part of license extension applications.

During a long-term shutdown, degradation of biological shielding concrete due to elevated temperatures does not occur as the heat sources (reactor vessel and gamma irradiation) are not active.

6.7 Reduced Radiation Shielding Performance Due to Elevated Temperature

The principal function of the concrete biological shield is to block radiation emanating from the reactor pressure vessel during power operations. A loss of moisture due to elevated temperature (radiation heating and heat transfer from the reactor pressure vessel) and decrease of density of aggregates due to neutron induced vacancy agglomeration (radiation induced volumetric expansion) can affect the local shielding properties in the affected volume of concrete. It is unlikely that concrete below the temperature limits of 65°C for structures and 90°C for local areas will cause a significant reduction in shielding capacity due to elevated temperatures. For reference, the concrete temperature limits are well below the specifications for gamma and neutron radiation shielding temperature limits defined in [6-14]. Radiation induced volumetric expansion may cause a slight decrease in shielding capacity but is inherently a near surface phenomenon, thus it is not expected to have a significant impact on overall shielding performance.

During a long-term shutdown, concrete thermal and irradiation damage are not expected to progress as a function of time as the fuel is removed from the reactor pressure vessel and there is no fission.

6.8 Reduced Strength due to Mechanical Vibration

Concrete can deteriorate as a result of mechanical fatigue when subjected to vibrations. Cyclic movements of equipment inside a nuclear plant can introduce vibrations into a concrete structure. Over time, these vibrations can cause fatigue cracks and reduce the strength of concrete structures. Given the time dependent nature of fatigue, an assessment of how fatigue due to mechanical vibrations can affect concrete strength should be considered for aging management.

The JNO Report indicates that visual inspections are performed on a regular basis to identify cracking in concrete foundations and support structures subjected to mechanical vibrations. The mechanical vibrations are also routinely monitored for any abnormalities that could indicate a change in the support structure. These inspections and monitoring are adequate to identify concrete cracking caused by mechanical vibrations.

During a long-term shutdown, the mechanical vibrations in the plant are reduced or eliminated.

6.9 Conclusions

Some concrete aging mechanisms, such as radiation damage and elevated temperature exposure, are only active when the plant is in operation and thus do not progress during long-term shutdown.

Aging mechanisms such as carbonation and chloride ingress may be active during long-term shutdown. However, while the degradation mechanisms continue during plant shutdown, EPRI agrees their effects can be managed through continued maintenance and aging management efforts.

Aging mechanisms such as ASR and freeze-thaw, if present, may continue progressing during long-term shutdown. However, while these degradation mechanisms can continue during plant shutdown, EPRI agrees their effects can be managed through continued maintenance and aging management efforts.

Mechanical vibrations are reduced or eliminated during long-term shutdown.

6.10 References

Note: References highlighted in yellow are available for purchase from EPRI; they are available at no charge to EPRI members.

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6-4. *Long-Term Operations: Aging Management of Concrete Structures Affected by Alkali-Silica Reaction.* EPRI, Palo Alto, CA: 2017. 3002010093.

6-5. *Mitigation and Repair of Concrete Structures Affected by Alkali-Silica Reaction (ASR).* EPRI, Palo Alto, CA: 2017. 3002010300.

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7
SUMMARY

NPP operating periods are limited to 60 years in Japan. Most of the Japanese NPPs experienced, or are still experiencing, prolonged plant shutdown periods after the Great Japan Earthquake of 2011. These prolonged shutdown periods have significantly reduced the remaining licensed operating times for the NPPs. As a result, the eleven JNOs have completed technical assessments of aging-related degradation for NPP infrastructure originating from the long-term shutdown period. These assessments are documented in the JNO Report that justifies NPP operating life can be determined based on total actual NPP operating years. The JNO Report asserts that a long-term shutdown period (i.e., approximately 10 calendar years) would not challenge the plant operating life from a technological viewpoint. The JNO Report further asserts that the effect of the long-term shutdown on NPP aging degradation is negligible.

In the U.S., NPPs utilize the SRP-LR, NUREG-1800, Revision 2 [7-1], and the GALL Report, NUREG-1801, Revision 2 [7-2], for generic regulatory guidance when applying for license extension to 60 years. Similarly, for a potential SLR to 80 years of plant operation, NUREG-2191 [7-3] details the GALL for SLR and NUREG-2192 [7-4] provides the SRP information.

These documents, along with EPRI's technical reports, were the primary bases for EPRI's technical assessments in this report. Specifically, EPRI made comparisons of the operating period extension methods, which are directly analogous to the recovery of a prolonged shutdown period, used by the JNOs to the methodology of license extension in the U.S. to ascertain if any gaps exist and to identify areas where the JNO approach may be non-conservative or in need of improvement when compared to U.S. evaluations.

This assessment generated the following conclusions based on the review of the JNO Report performed by EPRI:

Section 2: Review of Overall Approach to Technical Evaluation of Aging Degradation

EPRI agrees with the overall approach taken by the JNOs. The relevant aging and degradation mechanisms have been evaluated and the proper management processes put in place. The identified changes to plant maintenance plans, which account for the long-term shutdown period to ensure all relevant aging is detected or monitored, seem proper. As such, it is concluded that there are adequate aging management practices and programs in place to effectively manage aging issues and maintain plant safety for 60 years of actual operation, in addition to the period of long-term shutdown.

Section 3: Review of Aging Phenomena under Routine Maintenance Control

EPRI agrees with the JNO Report conclusions that corrosion and wall thinning can be adequately managed during long-term plant shutdown by routine maintenance, including component and system replacements and inspection activities. Furthermore, during the long-term shutdown, systems or portions of systems with no fluid flow will not experience wall thinning. As such, these issues do not represent a technical impediment to the recovery of the long-term shutdown period by the JNOs.

Section 4: Review of Four Specific Degradation Phenomena

1. Low Cycle Fatigue of Class 1 Components

   EPRI agrees with the content of the JNO Report for the LCF topics addressed in the JNO Report and the overall approach described in the JNO Report to assess the impact of long-term shutdown on LCF. EPRI further concurs with the three important conclusions documented in the JNO Report for LCF. In addition, EPRI notes that the LCF aging management program at all plants requires that every thermal transient be captured, tracked, and evaluated on a continuous basis as identified in the JNO Report. Therefore, even in the unlikely event that a transient of significance was to occur during any plant shutdown, existing aging management programs would capture and evaluate the impact of the event. Based on the reviews performed by EPRI in this report, EPRI agrees with the main JNO Report conclusion regarding LCF that, since no transient events are expected to occur during long-term shutdown, long-term plant shutdowns will not cause significant effects on LCF.

2. Neutron Irradiation Embrittlement

   EPRI agrees that the JNO Report conclusions for the neutron embrittlement assessment are a reasonable and logical extension of the fact that no embrittlement effects from neutron fluence are expected because of the absence of a fission reaction in the reactor core during long-term shutdown. Furthermore, during the long-term shutdown, no pressurized thermal transients, at either high pressure, low temperature or both, are possible to occur. Thus, no PTS events that challenge the integrity of the RPV under brittle conditions, nor any events that could induce a ductile failure via an RPV rupture, are predicted to occur. It is therefore concluded that there are no consequential aging effects or damage that occurs to the RPV during the long-term shutdown. As such, there are no technical or safety impediments to the recovery of the long-term shutdown period by the JNOs regarding RPV integrity.

3. Irradiation Assisted Stress Corrosion Cracking

   EPRI concludes that neither SCC nor IASCC are relevant for reactor components exposed to low temperatures below 60°C (140°F) during a long-term shutdown. Also, during shutdowns while the reactor is not critical, there is no neutron irradiation acceleration of SCC (i.e., IASCC) due to the absence of neutron fluence.

4. Thermal Aging of Cast Austenitic Stainless Steels

   Based on the existing body of research, for CASS plant components where temperature exposure is below 250°C (482°F), no reduction in fracture toughness is expected. The aging effect of thermal embrittlement at lower temperatures is not considered relevant and long-term aging effects below 250°C (482°F) do not need to be considered for CASS components. Therefore, there is no aging effect associated with thermal aging of CASS materials during a long-term shutdown.

5. Overall Conclusion

   Based on EPRI's review of the JNO Report approach, logic, and associated conclusions for these four specific degradation phenomena, EPRI generally agrees that these phenomena may be addressed by actions associated with aging management, and that these phenomena will not progress significantly during long-term shutdown.

Section 5: Review of Degradation of Electrical Instrumentation Equipment

The JNO members perform qualification testing using test parameters above both Japan and U.S. standard requirements. This introduces conservatism that the qualified cables have assurance they will perform their design safety function until the end of their qualified life, or that they will be replaced prior to the end of that qualified life. The maintenance practices performed by some JNO members are in alignment with EPRI cable aging management guidance and the maintenance practices of EPRI members who have cable aging management programs developed based on the EPRI guidance documents for low and medium voltage cables. Furthermore, the JNO evaluation logic that thermal and radiation conditions during long-term shutdowns result in minimal aging of cable insulation materials is correct. This conclusion is based not only on the operating temperatures and radiation levels the cables are exposed to over the expected 10-year plant shutdown, but also because the methods used to obtain qualified life are conservative and underestimate qualified life because of that conservatism.

Thus, EPRI agrees with the technical assessments and conclusions of the JNO Report and foresees no technical barriers to the recovery of the long-term shutdown period with respect to degradation of electrical instrumentation equipment.

Section 6: Review of Degradation of Concrete

Some concrete aging mechanisms, such as radiation damage and elevated temperature exposure, are only active when the plant is in operation and thus do not progress during long-term shutdown. Other concrete aging mechanisms, such as carbonation, chloride ingress, ASR and freeze-thaw, may be active during long-term shutdown. However, while the degradation mechanisms continue during plant shutdown, their effects can be managed through continued maintenance and aging management efforts.

Thus, EPRI agrees with the technical assessments and conclusions of the JNO Report and foresees no technical barriers to the recovery of the long-term shutdown period with respect to degradation of concrete structures.

Finally, where appropriate for each individual technical area, EPRI provided recommendations for potential improvements to the JNO Report, which the JNOs can include in their updated technical assessment for the recovery of the prolonged shutdown period, as necessary. EPRI did not identify any significant gaps or challenges that would preclude the JNO fleet from recovering the prolonged shutdown period and subsequently operating for 60 total years from a technical and safety standpoint. This conclusion is based on the JNO Report assessments, as well as additional qualitative assessments performed by EPRI consistent with U.S. license renewal methodologies and standards.

7.1 References

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