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MILLSTONE POWER STATION UNIT 2 – AUTHORIZATION AND SAFETY EVALUATION FOR ALTERNATIVE REQUEST NO. RR-05-06 (EPID L-2020-LLR-0097)

LICENSEE INFORMATION

Recipient's Name and Address: Mr. Daniel G. Stoddard
Senior Vice President and
Chief Nuclear Officer
Dominion Nuclear
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, VA 23060-6711

Licensee: Dominion Energy Nuclear Connecticut, Inc.

Plant Name and Unit: Millstone Power Station, Unit No. 2

Docket No.: 50-336

APPLICATION INFORMATION

Application Date: July 15, 2020

Application Agencywide Documents Access and Management System (ADAMS) Accession No.: ML20198M682

Supplement Date: March 19, 2021

Supplement ADAMS Accession No.: ML21081A136

Applicable Inservice Inspection (ISI) Program Interval and Interval Start/End Dates: Remainder of the fifth 10-year ISI interval and through the following sixth 10-year ISI interval for Millstone Power Station, Unit No. 2 (MPS2). The fifth 10-year ISI interval began April 1, 2020, and the sixth 10-year ISI interval is currently scheduled to end on March 31, 2040. The licensee recognizes that the existing 60-year license expires July 31, 2035. The U.S. Nuclear Regulatory Commission (NRC) staff noted that since the expiration of the 60-year operating license occurs before the end of the sixth 10-year ISI interval, the applicable end date of the proposed alternative is the end of the 60-year operating license.

Alternative Provision: The applicant requested an alternative under Title 10 of the *Code of Federal Regulations* (10 CFR), paragraph 50.55a(z)(1).

ISI Requirements: For American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 1 welds, the ISI requirements are those specified in Subarticle IWB-2500 of the ASME Code, Section XI, which requires the licensee to perform volumetric examinations of essentially 100 percent of the weld length as specified in ASME Code, Section XI, Table IWB-2500-1, for each Examination Category and Item No. listed below once every 10-year ISI interval:

- Examination Category B-B, Item No. B2.31, Steam Generator (SG) Primary Side Circumferential Head Welds
- Examination Category B-B, Item No. B2.40, SG Primary Side Tubesheet-to-Head Welds
- Examination Category B-D, Item No. B3.130, SG Primary Side Nozzle-to-Vessel Welds

For ASME Code Class 2 welds and nozzle inside radius sections, the ISI requirements are those specified in Subarticle IWC-2500 of the ASME Code, Section XI, which requires the licensee to perform volumetric and surface examinations as specified in ASME Code, Section XI, Table IWC-2500-1, for each Examination Category and Item No. listed below once every 10-year ISI interval. As noted in Table IWC-2500-1 for Examination Categories C-A and C-B, cases of multiple vessels of similar design, size, and service (such as SGs), the required examinations may be limited to one vessel or distributed among the vessels.

- Examination Category C-A, Item No. C1.10, Shell Circumferential Welds
- Examination Category C-A, Item No. C1.20, Head Circumferential Welds
- Examination Category C-A, Item No. C1.30, Tubesheet-to-Shell Welds
- Examination Category C-B, Item No. C2.21, Nozzle-to-Shell Welds
- Examination Category C-B, Item No. C2.22, Nozzle Inside Radius Sections

Applicable Code Edition and Addenda: 2013 Edition of the ASME Code, Section XI.

Brief Description of the Proposed Alternative: In Section 6.0 of Attachment 1 to its submittal dated July 15, 2020, the licensee stated that the proposed alternative is to increase the ISI interval from the current ASME Code, Section XI, requirement of 10 years to 30 years for the following MPS2 SG welds and nozzles, which are provided in Section 1.0 of Attachment 1 to the submittal.

Component ID, MPS2 SG1	Component ID, MPS2 SG2	Component Description	Item No.
SG-1-BHC-1-A	SG-2-BHC-1-A	Stay Cylinder Base to Hemisphere (Head) Weld	B2.31
SG-1-TSS-3-A	SG-2-TSS-3-A	Stay Cylinder to Tube Sheet Weld	B2.31
SG-1-BHC-2-A	SG-2-BHC-2-A	Hemisphere (Head) to Tube Sheet Weld	B2.40
SG-1-NH-2-A	SG-2-NH-2-A	Loop 1A/2A Cold Leg Nozzle to Hemisphere (Head) Weld	B3.130
SG-1-NH-4-A	SG-2-NH-4-A	Hot Leg Nozzle to Hemisphere (Head) Weld	B3.130
SG-1-NH-5-A	SG-2-NH-5-A	Loop 1B/2B Cold Leg Nozzle to Hemisphere (Head) Weld	B3.130
1-SC-2A	2-SC-2A	Lower Cone to Shell Weld	C1.10
1-SC-3	2-SC-3	Cone to Upper Shell Weld	C1.10
1-SC-4	2-SC-4	Hand Hole Ring to Shell Circumferential Weld	C1.10
1-SC-5	2-SC-5	Upper Shell to Lower Shell Circumferential Weld	C1.10
1-SC-6	2-SC-6	Lower Cone to Upper Cone Weld	C1.10
SG-1-THS-1	SG-2-THS-1	Secondary Head Circumferential Weld to Shell	C1.20
SG-1-THS-2	SG-2-THS-2	Head Circumferential Weld	C1.20
1-BHSC-2A	2-BHSC-2A	Hand Hole Ring to Tube Sheet Circumferential Weld	C1.30
SG-1-FW-1	SG-2-FW-1	Feed Water Nozzle to Shell Weld	C2.21

SG-1-MS-1	SG-2-MS-1	Main Steam Nozzle to Head Weld	C2.21
SG-1-FW-IR-1	SG-2-FW-IR-1	Feed Water Nozzle Inside Radius Section	C2.22
SG-1-MS-IR-1	SG-2-MS-IR-1	Main Steam Nozzle Inside Radius Section	C2.22

For additional details on the licensee's request, please refer to the documents located at the ADAMS Accession Nos. identified above.

STAFF EVALUATION

1.0 LICENSEE'S BASIS FOR PROPOSED ALTERTATIVE

The licensee referred to the results of the probabilistic fracture mechanics (PFM) analyses in the following Electric Power Research Institute (EPRI) reports for pressurized-water reactors (PWRs) as the primary basis for proposing to increase the ISI interval for the requested MPS2 SG welds and nozzles from 10 years to 30 years: non-proprietary EPRI report 3002015906, "Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head, and Tubesheet-to-Shell Welds," 2019 (Agencywide Documents Access Management System (ADAMS) Accession No. ML20225A141) and non-proprietary EPRI report 3002014590, "Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Nozzle Inside Radius Sections," April 2019 (ADAMS Accession No. ML19347B107). From this point forward, "EPRI report 15906" will refer to the former report and "EPRI report 14590" to the latter report.

The NRC staff's review focused on evaluating the PFM analyses in Section 8.3 of EPRI report 15906 and in Section 8.2 of EPRI report 14590 and verifying whether the deterministic fracture mechanics (DFM) analyses in both reports support the PFM results. The NRC staff reviewed the proposed alternative request for MPS2 as a plant-specific alternative. The NRC did not review the EPRI reports for generic use, and this alternative request does not extend beyond the MPS2 plant-specific authorization.

2.0 DEGRADATION MECHANISM

In Section 6.0 of Attachment 1 to the submittal, the licensee referred to the evaluation of potential degradation mechanisms in both EPRI reports and concluded that other than corrosion fatigue (also referred to as environmental assisted fatigue in both EPRI reports) and mechanical/thermal fatigue, there were no active degradation mechanisms identified that significantly affect the long-term structural integrity of the requested SG welds and nozzles of MPS2. The licensee stated that these fatigue-related mechanisms were considered in the PFM and DFM evaluations in the EPRI reports.

The NRC staff noted that the crack growth mechanism resulting from mechanical/thermal fatigue is fatigue crack growth (FCG), and that the effects of corrosion fatigue on FCG are included in the FCG rate selected for analyses (see Section 8.0 of this safety evaluation (SE)). The NRC staff finds the conclusion that corrosion fatigue and mechanical/thermal fatigue (both of which contribute to FCG) are the only active degradation mechanisms to be acceptable for the MPS2 plant-specific alternative request because: (1) FCG is known to be the dominant crack driving force in ferritic materials such as the SG welds and nozzles of MPS2 (see Section 5.1 of this SE); and (2) ferritic materials are known to be highly resistant to stress corrosion cracking under the operating conditions of the requested SG welds and nozzles of MPS2.

3.0 OVERALL PFM APPROACH

The PFM analyses in EPRI report 15906 were performed with the **PR**obabilistic **O**pti**M**ization of **I**n**S**p**E**ction (PROMISE), Version 2.0, software and with PROMISE, Version 1.0, for the PFM analyses in EPRI report 14590. See Section 3.1 of this SE for a discussion of the verification and validation (V&V) of both versions. Both versions of the software will be referred to as PROMISE from this point forward unless otherwise noted.

The overall PFM approach in both EPRI reports is based on a Monte Carlo sampling technique in which PROMISE samples parameters with statistical distributions, also called random parameters, many times to calculate a probability. Each sampling of parameters is known as a trial or a realization (see Section 10.4 of this SE for a discussion of the number of realizations used in the analysis). For each realization, PROMISE performs a DFM analysis based on linear elastic fracture mechanics (LEFM), to calculate a time to failure to develop a histogram of failure times, which is, briefly stated, a tally of failure times. Section 8.3.2.9 of EPRI report 15906 and Section 8.2.2.9 of EPRI report 14590 define failure as either rupture or leakage. Rupture is considered to occur when the applied stress intensity factor (SIF) exceeds plane strain crack initiation fracture toughness (K_{IC}). Leakage is considered to occur when the crack depth exceeds 80 percent of the wall thickness. From the histogram of failure times, PROMISE estimates the probability of failure (PoF) at a given time as the fraction of the total number of realizations that the computed failure time is less than the given time. The PoF is then determined on a per year basis and compared to an acceptance criterion of 1E-06 per year.

The NRC staff finds the overall PFM approach acceptable for the plant-specific MPS2 alternative request because the Monte Carlo technique is a widely used and accepted technique for calculating probabilities, and counting times to failure is counting the number of failures (i.e., the probability that the failure time is less than a given time is equivalent to the probability that a failure would occur within that given time).

The NRC staff noted that the acceptance criterion of 1E-06 failures per year is tied to that used by the NRC staff in the development of 10 CFR 50.61a, “Alternate fracture toughness requirements for protection against pressurized thermal shock events.” In that rule, the reactor vessel through-wall crack frequency (TWCF) of 1E-06 per year for a pressurized thermal shock event is an acceptable criterion because reactor vessel TWCF is conservatively assumed to be equivalent to an increase in core damage frequency, and as such meets the criteria in Regulatory Guide (RG) 1.174, “An Approach to for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.” This assumption is conservative because a through-wall crack in the reactor vessel does not necessarily increase core damage. The discussion of TWCF is explained in detail in the technical basis document for 10 CFR 50.61a, NUREG-1806 “Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61),” August 2007 (ADAMS Accession No. ML072830074).

The NRC staff also noted that the TWCF criterion of 1E-06 per year was generated using a very conservative model for reactor vessel cracking. The NRC staff finds that the licensee’s use of 1E-06 failures per year based on the reactor vessel TWCF criterion is acceptable for the requested SG welds and nozzles of MPS2 because the impact of a SG vessel failure is less than the impact of a reactor vessel failure on overall risk. The NRC staff further noted that comparing the probability of leakage to the same criterion is conservative because leakage is less severe than rupture.

Lastly, the NRC staff noted that acceptance criterion of 1E-06 failures per year is lower and, thus more conservative, than the criterion the NRC staff accepted in proprietary report BWRVIP-05 "BWR [boiling-water reactor] Vessel and Internals Project: BWR Reactor Pressure Vessel Weld Inspection Recommendation, September 1995"; non-proprietary report BWRVIP-108NP-A, "BWR Vessel and Internals Project: Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to -Vessel Shell Welds and Nozzle Blend Radii, October 2018" (ADAMS Accession No. ML19297F806); and non-proprietary report BWRVIP [Boiling Water Reactor Vessel and Internals Project]-241NP-A, "BWR Vessel and Internals Project: Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," October 2018" (ADAMS Accession No. ML19297G738). These EPRI reports were developed prior to or around the time the rules for PTS were reevaluated, and as such the acceptance criterion for failure frequency in the reports is based on the guidelines for PTS analysis in RG 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors" that were available at the time. RG 1.154 was later withdrawn in 2011.

Based on the above discussion, the NRC staff finds the use of the acceptance criterion of 1E-06 failures per year for PoF acceptable for the plant-specific MPS2 alternative request.

3.1 Software Verification and Validation (V&V)

The NRC staff conducted an audit of PROMISE, Version 1.0, in 2020 (at the time only this version of the software was available) to verify that it properly implemented PFM principles and has undergone adequate V&V. The NRC staff issued the audit summary report by letter dated December 10, 2020 (ADAMS Accession No. ML20258A002). The NRC staff also noted the benchmarking of PROMISE, Version 1.0, with another PFM software, VIPERNOZ, discussed in Section 8.2.3.2 of EPRI report 14590; this benchmarking was part of the V&V of the software. Even though the NRC has not formally accepted VIPERNOZ, it is the PFM software used in the BWRVIP-108 report for which the NRC staff has issued an SE dated December 19, 2007 (ADAMS Accession No. ML073600374). As documented in the audit summary report, the NRC staff requested additional benchmarking runs with VIPERNOZ contained in Structural Integrity Associates (SIA) report 1900064.407.R2 (Enclosure 3 in ADAMS Accession No. ML20253A311). While this report was submitted as part of another plant-specific submittal by Southern Nuclear Operating Company (SNC)[1] that referenced EPRI report 14590 as the technical basis, the benchmarking runs were performed with generic stresses instead of plant-specific stresses. The NRC staff reviewed the benchmark runs in 1900,064.407.R2, and determined that the results showed adequate agreement between PROMISE, Version 1.0, and VIPERNOZ for both probability of leakage values and probability of rupture values for different ISI scenarios. The NRC staff noted that with this benchmark of PROMISE, Version 1.0, benchmarking of PROMISE, Version 2.0, that is additional to the one performed in Section 8.3.3.2 of EPRI report 15906 is not necessary because of the adequate V&V performed for the difference between the two versions, as discussed next.

In the supplement dated March 19, 2021, the licensee identified the difference between PROMISE, Version 1.0, and PROMISE, Version 2.0, and described the V&V performed for this

[1] The NRC staff issued its SE of the SNC submittal by letter dated January 11, 2021 (ADAMS Accession No. ML20352A155).

difference. The licensee stated that the main difference between the two versions is that PROMISE, Version 1.0, applies a single, user-specified examination coverage value to all inspections assumed over the component evaluation time period, whereas PROMISE, Version 2.0, applies unique, user-specified examination coverage value, to each inspection assumed over the component evaluation period. The licensee stated that both versions assume 100 percent coverage for the pre-service inspection (PSI) examination. The NRC staff determined that the V&V of PROMISE, Version 2.0, is adequate because the licensee demonstrated that the code change was properly implemented for only those cases where examination coverages for each inspection were specified.

Based on the discussion above, the NRC staff finds that PROMISE, Version 1.0, and Version 2.0, received adequate V&V and, therefore, are acceptable for use in the MPS2 plant-specific alternative request for the SG welds and nozzles of MPS2.

4.0 PARAMETERS MOST SIGNIFICANT TO PFM RESULTS

In a PFM analysis, examples of the various input parameters that contribute to the final PoF value include crack dimensions, K_{IC}, stress, crack growth rate, and ISI schedule, all of which may be further defined by sub-parameters (such as the exponent term in the crack growth rate). Analysts typically use two sensitivity tools to understand the effects of the input parameters. Sensitivity analyses (SA) help identify the major contributors to the final PoF value, and sensitivity studies (SS) help in determining the impact of each in parameter on the final PoF value.

In Section 8.3.4.2 of EPRI report 15906 and Section 8.2.4.2 of EPRI report 14590, EPRI performed SA to determine the dominant parameters that contribute to the probability of leakage and rupture. The results of these SA are in Tables 8-11 and 8-12 of both reports. For probability of leakage, EPRI determined that the most dominant contributor is the FCG rate coefficient, and for probability of rupture, EPRI determined that the most dominant contributor is K_{IC}.

The NRC staff reviewed the overall results of the SA with respect to the MPS2 plant-specific alternative request. Onset of leakage is driven by growth of the postulated crack by the FCG rate, which is a measure of how fast the postulated crack would grow to 80 percent of the wall thickness; and FCG rate is proportional to the FCG rate coefficient. Thus, the NRC staff finds that the FCG rate coefficient being the dominant contributor to probability of leakage to be reasonable. Rupture is driven by applied SIF (which is driven by stress) or K_{IC} since applied SIF and K_{IC} (represented by the K_{IC} are the two parameters in the governing expression in LEFM: applied SIF < K_{IC}. Thus, the NRC staff finds that K_{IC} being the dominant contributor to probability of rupture to be reasonable. The NRC staff noted that even though the applied SIF did not come out as the dominant contributor in the SA, it is one of the significant parameters reflected in the parameter of stress in the SS, as discussed in the next paragraph.

In Section 8.3.4.3 of EPRI report 15906 and Section 8.2.4.3 of EPRI report 14590, EPRI performed SS on the following parameters: stress, K_{IC}, initial crack depth, number of flaws, flaw density (applicable only for the nozzle inside radii [NIR]), crack size distribution, FCG rate, probability of detection (POD), ISI schedule, and number of realizations. EPRI concluded that the most significant parameters are stress, K_{IC}, and flaw density in the NIR. As with the SA results for probability of rupture, the NRC staff finds this overall result of the SS reasonable for the MPS2 plant-specific alternative request since these are the parameters that directly affect the governing expression in LEFM (flaw density in the NIR affects the applied SIF side of the expression).

During the audit of PROMISE (see Section 3.1 of this SE), the NRC staff observed that ISI schedule and examination coverage have a significant impact on the PoF. The NRC staff requested two SIA letter reports that cover these topics, 1900064.406.R0 and 1900064.407.R2, which were included as Enclosures 2 and 3, respectively, in the SNC submittal (ADAMS Accession No. ML20253A311). Even though these two SIA letter reports were part of SNC's plant-specific alternative request, the impact of ISI schedule and examination coverage is a generic observation of the NRC staff on the PFM methodology.

The SA, SS, and the NRC staff's observations during the audit of PROMISE on the effects of ISI schedule and examination coverage was identified, the following significant parameters or aspects of the PFM analyses that warrant a close evaluation: stress analysis, K_{IC}, flaw density, FCG rate coefficient (or simply FCG rate), and effect of ISI schedule and examination coverage. The NRC staff discussed and closely evaluated each of these parameters in the next five sections of this SE. The NRC staff also evaluated other parameters or aspects of the analyses in Section 10.0 of this SE.

5.0 STRESS ANALYSIS

5.1 Selection of Components and Materials

In Attachment 2 to the submittal, the licensee evaluated the plant-specific applicability of the components and materials selected and analyzed in both EPRI reports to the SG welds and nozzles of MPS2. The licensee showed that MPS2 met the component configuration and material criteria. The acceptability of meeting the criteria, however, depends on the acceptability of the component and material selection described in both EPRI reports, which the NRC staff evaluated below.

In Section 4.5 of EPRI report 15906, EPRI discussed the selection of the following components for analysis: two primary side SG nozzle configurations to represent the primary side nozzles and one SG configuration to represent the primary side lower head of the SG and the secondary side shell of the SG. In Section 4.5 of EPRI report 14590, EPRI discussed the selection of representative secondary side SG nozzle configurations for analysis: two main steam (MS) nozzles and one feedwater (FW) nozzle. EPRI used these selected components for finite element front end analyses (FEA, see Section 5.4 of this SE) to determine stresses in the SG welds and nozzles, which the licensee referenced for the corresponding SG welds and nozzles requested for MPS2. The NRC staff noted that EPRI considered Babcock & Wilcox (B&W) nozzle designs, but since the MPS2 SGs are designed by Combustion Engineering (CE), the staff evaluated only the SG configurations in both reports applicable to a CE design. In selecting the components, EPRI considered geometry, operating characteristics, materials, field experience with respect to service-induced cracking, and the availability and quality of component-specific information.

EPRI concluded that variations in the configuration of SG lower heads, shells, and nozzles among the various designs are not significant, and that the most important parameter, ratio of radius-to-thickness (R/t) of the primary side nozzles, primary side lower head, secondary side shell, and secondary side nozzles, can be addressed through SS in the PFM evaluation.
Table 4-4 of EPRI report 15906 and Figure 4-4 of EPRI report 14590 show the various R/t ratios considered.

The NRC staff reviewed the discussion of R/t ratios in Sections 4.3 through 4.6 of EPRI report 15906 and EPRI report 14590, and finds the SG component configurations selected for stress analysis acceptable representatives for the corresponding SG welds and nozzles requested for the MPS2 plant-specific alternative request because differences in R/t ratios are small and, therefore, differences in stresses would be reasonably addressed through the SS on stress. To verify the dominance of the R/t ratio, the NRC staff reviewed the through-wall stress distributions in Sections 7.1 and 7.2 of EPRI report 15906 and in Sections 7.1 and 7.3 of EPRI report 14590 to confirm that the pressure stress is dominant, which would confirm the dominance of the R/t ratio. For the B3.130 weld analyzed in EPRI report 15906, and C2.21 welds and C2.22 NIR analyzed in EPRI report 14590, the NRC staff finds that EPRI's conclusion in both EPRI reports about the R/t ratio being the dominant parameter in evaluating the various configurations to be acceptable for the MPS2 plant-specific alternative request since the pressure stress is the dominant stress as evidenced in Figures 7-17 and 7-18 of EPRI report 15906 and in Figures 7-9, 7-10, and 7-32 through 7-35 of EPRI report 14590. However, for the B2.31, B2.40, C1.10, C1.20, and C1.30, welds analyzed in EPRI report 15906, EPRI did not show the stress distributions due to pressure separately.

In the supplement dated March 19, 2021, the licensee compared through-wall stress distributions due to combined thermal and pressure with those due to pressure only. In two of the comparisons, the NRC staff observed that although the pressure stress was notable, it was not necessarily dominant since the thermal stress was also high. The controlling geometric parameter for thermal stress is thickness. Even though the thermal stress is also high, the NRC staff determined that the SG configuration analyzed for the B2.31, B2.40, C1.10, C1.20, and C1.30, welds in EPRI report 15906 would still be adequate for the corresponding welds of MPS2 because the thickness of the MPS2 SG at these weld locations is close to the thickness of the SG model analyzed in EPRI report 15906. The NRC staff noted that the MPS2 B2.31 welds are covered by the MPS2-specifc SG model, as discussed next.

The NRC staff needed clarification and explanation for the following welds identified as Item No. B2.31 in Section 1.0 of Attachment 1 to the submittal (also listed in the beginning of this SE): "Stay Cylinder Base to Hemisphere (Head)" welds (component IDs SG-1-BHC-1-A and SG-2-BHC-1-A) and "Stay Cylinder to Tube Sheet" welds (component IDs SG-1-TSS-3-A and SG-2-TSS-3-A). It was not clear to the NRC staff whether these welds were represented in Figure 7-24 of EPRI report 15906. In the supplement dated March 19, 2021, the licensee provided the additional clarification and explanation. The licensee clarified that "Stay Tube to Head Weld" in Figure A2 of Attachment 2 to the submittal is the same as "Stay Cylinder Base to Hemisphere (Head)" weld. Similarly, the licensee clarified that "Stay Tube to Tubesheet Weld" in Figure A2 is the same as "Stay Cylinder to Tube Sheet" weld. Furthermore, the licensee explained in the supplement that it generated a SG model using the MPS2-specific configuration to address these welds. The NRC determined that the stresses generated from this MPS2-specific SG configuration are appropriate for the requested MPS2 SG B2.31 welds (SG-1-BHC-1-A, SG-2-BHC-1-A, SG-1-TSS-3-A, and SG-2-TSS-3-A) because the SG geometry used to create the SG model is specific to MPS2.

In Section 5.1 of both EPRI reports, EPRI discussed the material properties for the ferritic materials, SA-533 Grade B Class 1, SA-508 Class 2, SA-533 Grade A Class 2, and SA-516 Grade 70, that were selected for the stress analysis. In Section 5.1 of EPRI report 15906, EPRI also discussed the material properties of SA-240 Type 304 stainless steel assumed for all cladding material. The NRC staff finds these materials acceptable for the MPS2 plant-specific alternative request because they are common materials used in SG vessel base metal and cladding.

In Attachment 2 to the submittal, as stated in the first paragraph of this section, the licensee showed that MPS2 met the component configuration and material criteria in both EPRI reports. Therefore, together with the MPS2-specific SG model for the requested B2.31 welds, the NRC staff finds that the component configuration and materials of the requested SG welds and nozzles of MPS2 are acceptable.

5.2 Selection of Transients

In Attachment 2 to the submittal, the licensee evaluated the plant-specific applicability of the transients selected in both EPRI reports to the SG welds and nozzles of MPS2. For the MPS2 primary side SG nozzles, lower head, and secondary side shell, the licensee stated that the MPS2 transients and number of cycles projected to occur over a 60-year life are bounded by those in Tables 5-7 and 5-9 of EPRI report 15906. For the MPS2 SG MS and FW nozzles, the licensee stated that the MPS2 transients and number of cycles projected to occur over a 60-year life are bounded by those in Table 5-5 of EPRI report 14590. The acceptability of meeting the transient criteria, however, depends on the acceptability of the transient selection described in both EPRI reports, which the NRC staff evaluated below.

In Section 5.2 of both EPRI reports, EPRI discussed the thermal and pressure transients under normal and upset conditions considered relevant to SG welds and nozzles. EPRI showed a comparison of design cycles with projected cycles based on transient monitoring systems and explained that even though design cycles for loading and unloading transients are in the range of 20,000, the projected cycles for these transients are less than 1,000 for 60 years of operation because many plants do not load-follow.

EPRI developed a list of transients for analysis, shown in Tables 5-7 and 5-9 of EPRI report 15906 and Table 5-5 of EPRI report 14590, based on transients that have the largest temperature and pressure variations. EPRI stated that additional cycles of the Loss-of-Load transient addressed the transients not explicitly selected for analysis in EPRI report 14590. In the supplement dated March 19, 2021, the licensee explained that the number of cycles of the Reactor Trip transient in EPRI report 15906 bounds the combined number of cycles expected for the Reactor Trip transient and other transients not explicitly selected, such as the Large Step Load Decrease transient. The licensee also stated that the remaining transients in Table 5-6 of EPRI report 15906 were excluded because they are not typical or expected transients in PWRs.

The NRC staff reviewed the discussion of transients in Section 5.2 of both EPRI reports and in the supplement to the submittal, and determined that the transients defined in Tables 5-7 and 5-9 of EPRI report 15906 and Table 5-5 of EPRI report 14590 selected for analysis are reasonable for the MPS2 plant-specific alternative request because the transient selection was based on large temperature and pressure variations that are conducive to FCG and on transients that are expected to occur in PWRs.

EPRI did not consider test conditions beyond a system leakage test. EPRI stated that since any pressure tests will be performed at operating pressure, no separate test conditions need to be included in the evaluation. The NRC staff reviewed the MPS2 Updated Final Safety Analysis Report and noted that Sections 4.2.1 and 4.3.2 (ADAMS Accession No. ML20209A354) specify the following design-basis number of cycles and pressures for hydrostatic and leak testing: 10 cycles of primary side hydrostatic testing at 3,110 pounds per square inch gauge (psig) per and 10 cycles of secondary side hydrostatic testing at 1,235 psig; and 200 cycles of primary side leak testing at 2485 psig and 200 cycles of secondary side leak testing at 985 psig. The NRC staff determined that the SS on stress in both EPRI reports address the pressure values during the hydrostatic and leak testing since they are bounded by the stress multiplier of 1.9 in Tables 8-19 and 8-20 of EPRI report 15906 and the stress multipliers of 1.5 and 2.2 in Tables 8-15 and 8-16 of EPRI report 14590.

In terms of cycles, the NRC staff noted that in Tables A1, A2, and A3, in Attachment 2 of the submittal, the licensee stated that there are 121 projected 60-year cycles of Heatup and Cooldown for MPS2. This leaves 179 cycles of Heatup and Cooldown that can account for the MPS2 pressure tests. However, the total design number of pressure testing (hydrostatic and leak) is 210 cycles, which would leave 31 design cycles of pressure testing unaccounted for. In the supplement dated March 19, 2021, the licensee explained that the 210 cycles for pressure testing are design-basis cycles and do not reflect the MPS2 operating practice and the important pressurization cycles captured by the MPS2 fatigue management program. The licensee also stated that leakage tests are performed at MPS2 instead of hydrostatic tests, and that the leakage tests are integral to the heatup process. The NRC staff determined that since the leakage tests are integral to the heatup process, the number of cycles of leakage test need not be separate to the 121 projected 60-year cycles of the Heatup/Cooldown transient for MPS2, and that the 121 cycles are bounded by the 300 cycles of the Heatup/Cooldown transient assumed in both EPRI reports.

The NRC staff needed confirmation that at the maximum primary and secondary side pressures during the hydrostatic and leak tests of MPS2, the temperature of the primary side affecting the primary side welds requested for MPS2 in the submittal and the temperature of the secondary side affecting the secondary side welds requested for MPS2 in the submittal are high enough such that the upper shelf K_{IC} value of 200 ksi√in (Knowledge and Skills Inventory) assumed in the EPRI reports for K_{IC} is appropriate. As discussed in the previous paragraph, the licensee performs leakage tests instead of hydrostatic tests, and the MPS2 leakage tests are integral to the heatup process (i.e., included in the Heatup/Cooldown transient). As discussed in Section 6.0 of this SE, the licensee's comparisons of applied SIF history with K_{IC} resolved the NRC staff's note of low K_{IC} that can occur during the beginning and ending portions of the Heatup/Cooldown transient.

Based on the discussion above, the NRC staff finds that the test conditions at MPS2 are adequately captured in the Heatup/Cooldown transient analyzed in both EPRI reports.

EPRI did not evaluate faulted or emergency conditions separately. In Section 5.2 of EPRI report 15906, EPRI stated that for the primary side of the SG, transients due to faulted conditions do not have significant stress variation because any faulted transient in the reactor coolant system (RCS) would lead to depressurization and a decrease in stress in the RCS. In Section 5.2 of both EPRI reports, EPRI assumed that for the secondary side of the SG during faulted condition transients, the temperature and pressure in the SG secondary side would decrease and that the decrease in temperature is expected to be small. EPRI, therefore, concluded that the transients selected for analysis (i.e., those in Tables 5-7 and 5-9 of EPRI report 15906 and Table 5-5 of EPRI report 14590) bound the faulted condition transients.

The NRC staff determined this to be a reasonable conclusion because the depressurization would lower stresses during these faulted events, and as such, the stresses generated during the faulted condition would be bounded by those generated during the transients selected for analysis. The NRC staff noted that plants typically include either faulted or emergency condition, not both, in the design basis. Also, considering that a faulted (or emergency) condition transient is rare, typically only one event during the life of a plant, the NRC staff determined that the faulted condition transient would have little impact on FCG. Accordingly, based on this discussion, the NRC staff determined that not evaluating faulted or emergency conditions separately is acceptable.

The NRC staff noted that even though the cycles in Tables 5-7 and 5-9 of EPRI report 15906 and Table 5-5 of EPRI report 14590 are for a 60-year period, the crack growth analyses performed in the PFM analysis due to the listed transients are based on the cycles in the tables that are converted into cycles per year (e.g., 5,000 cycles in 60 years is equivalent to 84 cycles per year) and the crack growth computed for up to 80 years.

In Attachment 2 to the submittal, as discussed in the first paragraph of this section, the licensee showed that MPS2 met the transient criteria in both EPRI reports. Therefore, the NRC staff finds that the transient loads for the requested SG welds and nozzles of MPS2 are acceptable.

5.3 Other Operating Loads

In Section 5.2 of both EPRI reports, EPRI stated that loads from the piping attached to the nozzles were not considered in the analysis. The NRC staff finds this assumption acceptable for the MPS2 plant-specific alternative request since bending stresses due to the attached piping have marginal impact on the orientation of the flaws postulated in the nozzle-to-shell welds and NIR of the SG nozzles analyzed in the EPRI reports and referenced for MPS2.

In Section 8.2.2.3.2 of EPRI report 15906 and Section 8.2.2.4.2 of EPRI report 14590, EPRI discussed the cosine distribution assumed for the through-wall residual stress due to welding. EPRI treated this distribution as a non-random parameter and stated that since the nozzle-to-shell weld is close to the NIR in some configurations, this cosine distribution is applicable to both locations. EPRI stated that this residual stress distribution has been used in past projects, particularly, BWRVIP-108 for which that the NRC staff has issued an SE dated December 19, 2007 (ADAMS Accession No. ML073600374).

BWRVIP-108 is the PFM-based technical basis for the reduction of the number of nozzles inspected in BWR pressure vessels on which some of the inputs for both EPRI reports were based. The NRC staff noted that the residual stress distribution in BWRVIP-108 is for welding in thick-walled vessels. The NRC staff finds the cosine distribution EPRI assumed for the through-wall residual stress due to welding acceptable for the MPS2 plant-specific alternative request since the nozzle-to-shell welds and NIR sections of the requested SG nozzles of MPS2 are in the thick-walled locations of the SG vessel or nozzle. The NRC staff noted from Figure A5 in Attachment 2 to the submittal that the NIR location of the MPS2 SG FW nozzle is relatively far away from the nozzle-to-shell weld and, thus, the assumption of residual stress in the FW NIR of MPS2 is conservative. Finally, the NRC staff noted that treatment of the through-wall residual stress as a constant (non-random) parameter in both EPRI reports is reasonable because it only has a mean load effect on FCG and does not affect the range of applied load that is the main driver of FCG.

The NRC noted that the welds in the primary side of the MPS2 SG are cladded and, therefore, the effect of clad residual stress needed to be included. In the supplement dated March 19, 2021, the licensee included the effect of clad residual stress using Equation 8-1 of non-proprietary EPRI report 3002015905, “Technical Bases for Inspection Requirements for PWR Pressurizer Head, Shell-to-Head, and Nozzle-to-Vessel Welds,” December 2019 (ADAMS Accession No. ML21021A271). The licensee determined the probability of leakage per year with the clad residual stresses calculated from Equation 8-1 of EPRI report 15905 included with all other stresses, for an ISI scenario of PSI + 10 + 20 + 50, which bounds the licensee’s proposed alternative ISI of PSI + 10 + 20 + 30 + 40 + 70 (see Section 9.0 of this SE). The licensee showed that for the limiting location, SGPTH-P4A in EPRI report 15906, adding the clad residual stress had no impact on both probability of leakage and rupture values.

The NRC staff determined that using Equation 8-1 of EPRI report 15905 would adequately account for clad residual stress because, as can be seen in the equation, the clad residual stress decreases with increasing temperature, which is the expected behavior of clad residual stress with respect to temperature. However, the NRC staff observed that adding the clad residual stress determined from Equation 8-1 of EPRI report 15905 with the FEA stresses described in Section 7 of EPRI report 15906 for the cladded welds could result in a lower net stress within the cladding because the effect of differential thermal expansion between the cladding and base metal (an effect that results in a compressive stress within the cladding) is included twice: first in the thermal stress from the FEA (see Section 5.4 of this SE) and second in Equation 8-1 of EPRI report 15905, which includes the differential thermal expansion effect in addition to residual stress due to the welding of the cladding.

To resolve the doubling of the thermal expansion differential effect, the NRC staff further looked into how the clad residual stress affects the postulated flaws in the cladded welds analyzed in EPRI report 15906. The depth of the postulated flaws is described by the distribution derived from flaw data from Pressure Vessel Research User's Facility (PVRUF) project (see Section 10.1 of this SE). This distribution is shown in Equation 8-1 of EPRI report 15906. Based on this postulated flaw distribution, the depths of flaws that are evaluated in the PFM analysis 90 percent of the time are 0.0787 inch or less. The thickness of the cladding in the modeled SG in EPRI report 15906 is 0.27 inch. Since 90 percent of time the postulated flaw depth is 0.0787 inch or less, which is well within the cladding thickness, the postulated flaw would be within the stainless steel cladding, where the K_{IC} is much higher than that of the ferritic steel base meta, and, therefore, the total applied stress will not lead to failure. Accordingly, the NRC staff determined that even if the effect of differential thermal expansion between the cladding and base metal was accounted for only once, there would be little or no impact to the final probability of leakage and rupture values.

Based on the discussion above, the NRC staff finds the treatment of other loads described in this section of the SE acceptable for the requested SG welds and nozzles of MPS2.

5.4 Finite Element Analyses (FEA)

In Section 7 of both EPRI reports, EPRI discussed the FEA to determine stresses due to internal pressure and thermal transients for the selected geometries discussed in Section 5.1 of this SE. In Attachment 2 of the supplement to the submittal, the licensee discussed the FEA for the MPS2-specific SG model. The NRC staff reviewed the modeling details (elements used, boundary conditions, symmetry assumptions, etc.) and finds that they are consistent with standard FEA practice.

The NRC staff also reviewed the stress contour plots and the through-thickness stress distributions from the FEA and finds them acceptable for the plant-specific alternative request. For instance, the NRC staff noted the average hoop stress due to a 1,000 pounds per square inch (psi) internal pressure at the SG FW nozzle shell in Figure 7-34 of EPRI report 14590 is approximately 28,000 psi. The NRC staff verified this value by calculating a hoop stress of 1,000(R/t) = 1,000(25) = 25,000 psi (R/t = 25 for the SG FW nozzle shell from page 4-18 of EPRI report 14590).

The NRC staff noted that the through-wall stress distribution plots for the thermal transients in

Figures 7-17 and 7-18 of EPRI report 15906 for the B3.130 welds show compressive stresses at the inner surface (except the Heatup/Cooldown transient). Tensile stresses at the inside surface are typically expected for transients that have temperature drops, such as Reactor Trip, which has a temperature drop of 75 degrees Fahrenheit (°F) in 10 seconds as described in Section 5.2.2 of EPRI report 15906. In the supplement dated March 19, 2021, the licensee explained that for the transients with temperature drops, the SG starts hot and that the differential thermal expansion between the stainless steel cladding and the low alloy steel base metal causes significant compression on the inside surface at the start of these transients. The licensee included figures that showed that the inside surface stress became less compressive during the transient and explained the temperature drop was not large enough for the stresses to become tensile except for the cooldown portion of the Heatup/Cooldown transient. The NRC staff verified independently that the differential thermal expansion between the stainless cladding and low alloy steel base metal can generate a compressive stress on the inside surface at hot conditions and that the compressive stress can remain during the transient if the temperature drop is not large enough. Accordingly, the NRC staff determined that having compressive stresses on the inside surface for the transients in question is reasonable, even though the licensee's stress value on the inside surface is more compressive than the NRC staff's value. The NRC staff noted that the compressive stress in the FEA caused by the differential thermal expansion between the stainless cladding and low alloy steel base metal is due to the effect of two different adjacent materials and does not account for the residual stresses within the clad generated from the welding process of the clad. An iterative procedure in the FEA would need to be performed in order to determine a stress-free temperature that would adequately simulate the effect of clad residual stress in the FEA. The NRC staff discussed the topic of clad residual stress in Section 5.3 of this SE.

The NRC staff noted that EPRI modeled the effect of the thermal sleeve in the SG FW nozzle by applying an effective heat transfer coefficient to the nozzle surface protected by the thermal sleeve (see Sections 7.3.1 and 7.3.2.2 of EPRI report 14590). The NRC staff also noted that EPRI included in the plant-specific applicability criteria in Section 9.0 of the report that the FW nozzle design should have an integrally attached thermal sleeve. The licensee stated in Attachment 2 to the submittal that the MPS2 FW nozzles have an integrally attached thermal sleeve.

Based on the discussion above, the NRC staff determined that the pressure and thermal stresses calculated through FEA in both EPRI reports are acceptable for referencing for the requested SG welds and nozzles of MPS2, and that the FEA in Attachment 2 of the supplement are appropriate for the requested SG B2.31 welds of MPS2.

6.0 FRACTURE TOUGHNESS (K_{IC})

In Sections 8.2.2.5 and 8.3.2.7 of EPRI report 15906 and Section 8.2.2.7 of EPRI report 14590, EPRI assumed for K_{IC} of ferritic materials an upper shelf K_{IC} value of 200 ksi√in based on the upper shelf K_{IC} value in the ASME Code, Section XI, A-4200. EPRI treated K_{IC} as a random parameter normal distribution with a mean value of 200 ksi√in and a standard deviation of 5 ksi√in, stating that these assumptions are consistent with the BWRVIP-108 project. As discussed in Section 5.1 of this SE, MPS2 meets the material criteria in the EPRI report and, thus, the NRC staff determined that the K_{IC} parameters above are applicable to MPS2.

The NRC staff had accepted the treatment of upper shelf K_{IC} as a random parameter in BWRVIP-108 through sensitivity studies on K_{IC} that the NRC staff requested, which changed the standard deviation to less than 5 ksi√in and greater than 5 ksi√in (see December 19, 2007, SE of BWRVIP-108; both EPRI reports also include SS on K_{IC} as discussed below). Apart from this, EPRI stated in both EPRI reports that the reference nil-ductility temperature (RT_{NDT}) assumed for the SG welds and nozzles in the reports is 60 °F. In the supplement dated March 19, 2021, the licensee determined an upper bound, MPS2-specific RT_{NDT} value of 0 °F for the MPS2 SGs, based on a review of all plant-specific SG materials specifications. Since the MPS2-specific RT_{NDT} value of 0 °F is lower than the RT_{NDT} value of 60 °F assumed in the EPRI analyses, the NRC staff noted that this would have an overall conservative effect in the assumption of the K_{IC} value for the MPS2 SGs. For these reasons, the NRC staff finds that the mean and standard deviation values of upper shelf K_{IC} used in both EPRI reports are reasonable for the MPS2 plant-specific alternative request, even though statistical distributions that would more accurately account for the uncertainty in upper shelf K_{IC} should have been used.

EPRI further explained that an upper shelf K_{IC} value of 200 ksi√in can be used for K_{IC} since the minimum temperature from all applicable transients is sufficiently high (i.e., the temperature in the SG welds and nozzles analyzed in both EPRI reports stays in the range in which the upper shelf value of 200 ksi√in is applicable).

The NRC staff noted that the temperature of the requested SG welds and nozzles of MPS2 can be as low as 70 °F for the Heatup/Cooldown transient, as shown in Tables A2 through A4 in Attachment 2 to the submittal (Table A4 refers to Table 5-5 of EPRI report 14590). The NRC staff further noted that a K_{IC} value lower than the upper shelf K_{IC} of 200 ksi√in assumed in the analysis may exist during the beginning and ending portions of the Heatup/Cooldown transient. In the audit summary report for PROMISE (see Section 3.1 of this SE), the NRC staff observed from plots of total applied SIF history for the Heatup/Cooldown transient at the SG FW nozzle analyzed in EPRI report 14590 that the total applied SIF can exceed K_{IC} at a value as low as 130 ksi√in; this exceedance occurs during the beginning and ending of the Heatup/Cooldown transient. The NRC staff determined that the SS on K_{IC} in Tables 8-13 and 8-14 of EPRI report 14590 adequately addresses this low K_{IC} value because a K_{IC} value of 80 ksi√in resulted in a probability of rupture of 3.75×10^{-8} per year for the limiting location analyzed in the report (SG FW nozzle-to-shell weld, Item No. C2.21), which is approximately two orders of magnitude below the criterion of 1E-06 per year. Also, the applied SIF history was for deep and long flaws, and the occurrence of deep and long flaws with the flaw distribution used in the analysis (see Section 10.1 of this SE) is rare during the Monte Carlo sampling. In the supplement dated March 19, 2021, the licensee showed similar applied SIF history plots for the Heatup/Cooldown transient for two locations that include the limiting location, SGPTH-4A, in EPRI report 15906. The plots showed that the applied SIF did not exceed the lowest value of K_{IC} of about 115 ksi√in, which the licensee determined from the MPS2-specific RT_{NDT} value 0 °F for the MPS2 SG materials.

The comparisons of applied SIF history with K_{IC} discussed above resolve the NRC staff's note of low K_{IC} during the beginning and ending portions of the Heatup/Cooldown transient. However, the NRC staff observed that modeling K_{IC} as a function of temperature would have been a better approach, as there may be cases at the very beginning and very ending of the Heatup/Cooldown transient when K_{IC} could drop below 80 ksi√in analyzed in one of the SS studies on fracture toughness in Tables 8-14 and 8-16 of EPRI report 15906 and Tables 8-13 and 8-14 of EPRI report 14590. This could lead to slightly higher PoF values, but the NRC staff determined for this MPS2 plant-specific alternative request that the impact on PoF values would be minimal since pressure would be low at the very beginning and ending of the Heatup/Cooldown transient.

In Attachment 2 to the submittal, the licensee stated that MPS2 has not experienced a loss-of-power transient that could result in a thermal shock due to unheated auxiliary FW being introduced into a hot SG vessel. The NRC staff noted that this pertains to the secondary side of the MPS2 SG. In the supplement dated March 19, 2021, the licensee stated that MPS2 has not experienced a thermal shock event in the primary side of the RCS that can affect the K_{IC} value assumed in the analysis in EPRI report 15906. Therefore, the NRC staff determined that the impact of a thermal shock on the assumed K_{IC} value for K_{IC} need not be considered.

Based on the above discussion and the discussion in Sections 5.1 and 5.2 of this SE that the materials and transient loads are acceptable for the requested SG welds and nozzles of MPS2, the NRC staff finds the K_{IC} model in both EPRI reports acceptable for the requested SG welds and nozzles of MPS2.

7.0 FLAW DENSITY

In Table 8-8 of EPRI report 15906 and Table 8-7 of EPRI report 14590, EPRI indicated that 1.0 flaw per weld is used in all welds except at the MS and FW NIR, in which 0.001 flaw per NIR is used. EPRI stated that these values are consistent with those the NRC staff approved in BWRVIP-108. As discussed in the SE of BWRVIP-108 dated December 19, 2007, the NRC staff had requested that the number of flaws per nozzle at the NIR be revised to 0.1 flaw per nozzle in the PFM analyses in BWRVIP-108, and therefore, the acceptable number of flaws at the NIR is 0.1 flaw per nozzle (discussed further two paragraphs down).

The NRC staff determined in the December 19, 2007, SE of BWRVIP-108 that based on a surface-breaking flaw density of 0.01 flaw per cubic foot (flaw/ft^3), 1.0 flaw per weld is conservative for the nozzle weld configurations analyzed in BWRVIP-108. Similarly, using a surface-breaking flaw density of 0.01 flaw/ft(feet)^3 per weld and the volumes of the subject welds of MPS2 estimated from the dimensions in Attachment 2 of the submittal, the NRC staff determined that 1.0 flaw per weld is conservative for the requested primary and secondary SG welds of MPS2.

In Section 8.2.4.3.4 of EPRI report 14590, EPRI changed the number of flaws in the NIR from 0.001 to 0.1, and as a result, the probabilities of leakage and rupture increased by two orders of magnitude but were still significantly below the acceptance criterion of 1E-06 per year. This result, however, is only the effect of changing to 0.1 flaw per nozzle at the MS and FW NIR. Other significant parameters such as stress and K_{IC} should be considered. The NRC staff discussed the combined effects of stress, K_{IC}, and flaw density at the MS and FW NIR in Section 11.2 of this SE. The NRC staff observed in the audit summary report for PROMISE (see Section 3.1 of this SE) that the flaw density value in the MS and FW NIR is a multiplier on the probability values reported in the software output. The NRC staff determined that for the MPS2 plant-specific alternative request that multiplying by the fractional flaw density is a reasonable approach to account for the lower number of flaws expected in the nozzle forging.

Based on the discussion above, the NRC staff finds that the assumption of 1.0 flaw per weld is acceptable for the requested SG primary and secondary welds of MPS2. On the assumed number of flaws per nozzle at the MS and FW NIR of MPS2, the NRC staff finds that the acceptable number of flaws is 0.1 per nozzle. Even though EPRI assumed 0.001 flaw per nozzle at the MS and FW NIR for the base case PFM analyses, EPRI performed a SS on the impact of 0.1 flaw per nozzle. Considering this SS and the NRC staff's discussion of the results in Tables 8-15 and 8-16 of EPRI report 14590 in Section 11.2 of this SE, the NRC staff determined that EPRI report 14590 has adequate information for the MPS2 plant-specific alternative request that addresses the effect of 0.1 flaw per nozzle, compared to 0.001 flaw per nozzle, for the requested SG MS and FW NIR of MPS2.

8.0 FCG RATE

In Section 8.3.2.6 of EPRI report 15906 and Section 8.2.2.6 of EPRI report 14590, EPRI stated that the FCG rate for ferritic steels, as defined in the 2017 Edition of the ASME Code, Section XI, Appendix A, paragraph A-4300, is used in the evaluation. The NRC staff verified that the 2017 Edition of the ASME Code, Section XI, is the latest edition incorporated by reference in 10 CFR 50.55a. The NRC staff also confirmed that the FCG rate in A-4300 in the 2017 Edition of ASME Code, Section XI, is the same as that in the 2013 Edition of ASME Code, Section XI, because the 2013 Edition of ASME Code, Section XI, is MPS2's code of record. The NRC staff noted that the FCG rate in ASME Code, Section XI, A-4300, is applicable to both BWR and PWR.

Specifically, the FCG rate is defined with a log-normal distribution with the median value defined as the FCG rate in ASME Code, Section XI, A-4300, and with a value of 0.467 for the uncertainty parameter. The NRC staff finds the uncertainty parameter of 0.467 acceptable for the MPS2 plant-specific alternative request because it is based on over 1,000 FCG rate data for low alloy steels.

In Section 8.3.4.1 of EPRI report 15906 and Section 8.2.4.1 of EPRI report 14590, EPRI stated that assuming the A-4300 curve as median is conservative since the actual data from which the A-4300 curve is based on represent the 95 percent confidence limit of the data. The NRC staff clarifies that 95 percent confidence limit here means that the A-4300 curve bounds the median of the data 95 percent of the time; it does not mean that the A-4300 curve is the 95th percentile of the data. Because of the amount of available data for ferritic FCG rate, however, the difference between 50 percent confidence limit on the median and 95 percent confidence limit on the median would likely be small. Thus, the NRC staff determined for the MPS2 plant-specific alternative request that assuming the A-4300 as the median curve would only be slightly conservative.

EPRI stated that the associated threshold on the FCG rate is also log-normally distributed and that the log-normal distributions on the rate and threshold are consistent with the approach used in the xLPR, a PFM software sponsored by the NRC and EPRI. The NRC staff confirmed that the FCG rate in xLPR received adequate V&V.

In Section 8.3.4.3.7 of EPRI report 15906 and Section 8.2.4.3.7 of EPRI report 14590, EPRI performed a SS on the effect of FCG rate on probability of leakage by comparing the A-4300 FCG rate with the FCG rate used in BWRVIP-108. The result of the SS showed that the A-4300 FCG rate led to a lower probability of leakage for the limiting case (SGPTH-P4A) in EPRI report 15906 and a higher probability of leakage for the limiting case (FEW-P3A) in EPRI report 14590. The NRC staff finds the SS result for the limiting case in EPRI report 14590 acceptable since the A-4300 FCG rate resulted in a higher probability of leakage. For the limiting case in EPRI report 15906, EPRI stated that the A-4300 FCG rate resulted in a lower probability of leakage because of the higher crack growth rate from the BWRVIP-108 model that considered a conservative R-ratio of 0.7. EPRI also stated the limiting case had high negative value of the minimum applied SIF (K_{min}). The NRC staff determined that the high negative value of K_{min} would result in a negative R-ratio, assuming a positive value of maximum applied SIF (K_{max}), and this effect is captured in the A-4300 FCG rate since it considers all possible R-ratios. Since a negative R-ratio would result in a lower FCG rate than a positive R-ratio, e.g., the

R-ratio of 0.7 used in the BWRVIP-108 model, the A-4300 FCG rate would result in a lower crack growth that would lead to a lower probability of leakage.

Based on the discussion above, the NRC staff finds that the A-4300 FCG rate used in the analyses is acceptable for the requested SG welds and nozzles of MPS2.

9.0 ISI SCHEDULE AND EXAMINATION COVERAGE

In Section 6 of Attachment 1 to the submittal, the licensee stated that for MPS2, PSI examination followed by four 10-year ISI examinations (PSI + 10 + 20 + 30 + 40) have been performed, and in Attachment 3 to the submittal included the inspection history and examination coverage results for the requested SG welds and nozzles of MPS2 for the third and fourth 10-year ISI intervals. The licensee is proposing an alternative that would extend the ISI interval to 30 years after the first four 10-year ISI examinations (i.e., an alternative ISI schedule of PSI + 10 + 20 + 30 + 40 + 70).

In Section 8.3.4.1.1 of EPRI report 15906 and Section 8.2.4.1.2 of EPRI report 14590, EPRI discussed the effect of various ISI schedules on the PoF results; and in Section 8.3.5 of EPRI report 15906 and Section 8.2.5 of EPRI report 14590, EPRI discussed inspection (i.e., examination) coverage.

The NRC staff noted the impact of ISI schedule on the PoF values, as shown in Table 8-10 of EPRI report 15906 and Figure 8-9 of EPRI report 14590. In the audit summary report for PROMISE (see Section 3.1 of this SE), the NRC staff documented how ISI is implemented in the software, as described in the following: the number and frequency of ISI are input into the software; at the specified times of ISI, flaws are either detected or not detected with the chance of detection/non-detection given by the POD curve (see Section 10.2 of this SE). If detected, a flaw is assumed to be repaired or properly dispositioned, and thus, cannot cause failure; if not detected, the flaw continues to grow, and thus, can lead to failure. The NRC staff determined this to be a better approach than applying an adjustment factor to the failure probabilities since the effect of the POD curve would be propagated into the failure probabilities each time ISI is implemented. As discussed in Section 3.1 of this SE, the NRC staff requested additional benchmarking runs with VIPERNOZ contained in SIA report 1900064.407.R2 (Enclosure 3 in ADAMS Accession No. ML20253A311). These benchmarking runs were performed with various ISI schedules and generic stresses. The comparison plots in Figures 1 through 4 of SIA report No. 1900064.407.R2 showed adequate agreement between PROMISE, Version 1.0, and VIPERNOZ. Implementation of ISI did not change in PROMISE, Version 2.0, (see Section 3.1 of this SE). Thus, because of the adequate implementation and benchmarking of ISI, the NRC finds that the PoF values in EPRI reports 15906 and 14590 adequately included the effect of ISI schedule. Since the licensee referenced the PFM results in both EPRI reports for the requested SG welds and nozzles of MPS2, the NRC staff finds that the licensee adequately included the effect of ISI schedule on the PoF values for those SG welds and nozzles of MPS2.

In Section 8.3.5 of EPRI report 15906 and Section 8.2.5 of EPRI report 14590, EPRI stated that it assumed 100-percent inspection of the required volume (i.e., 100-percent examination coverage) of the SG welds and nozzles analyzed in both reports during each of the ISI scenarios evaluated in the reports, which assumes 100-percent examination coverage during PSI. EPRI further explained that based on its statements on the PSI-only examinations in Section 8.3.4.1 of EPRI report 15906 and Section 8.2.4.1.1 of EPRI report 14590, by performing complete 100-percent examination coverage during PSI, no other examinations are needed for safe plant operation for 80 years, and that based on this, any additional ISI examinations after

PSI would reduce the already low PoF values, and that, therefore, the PFM evaluations with 100-percent examination coverage assumed for all ISI also apply to partial, i.e., less than 100-percent, examination coverage. In Section 11.0 of this SE, the NRC staff explained its non-acceptance of the licensee's and EPRI's conclusion on PSI-only examinations. Thus, the NRC staff determined that partial examination coverage plays a vital role in the final PoF values.

In the audit summary report for PROMISE (see Section 3.1 of this SE), the NRC staff documented how the software implements examination coverage for a case with 50 percent coverage: in a given PFM evaluation, the POD curve is not applied for approximately 50 percent of the number of realizations at the specified times of ISI and, thus, for 50 percent of the realizations, a postulated flaw would continue to grow. The NRC staff determined that this an acceptable approach for implementing examination coverage since its effect would be propagated into the failure probabilities each time ISI is implemented.

In Attachment 3 of the submittal, the licensee provided the examination history during the third and fourth 10-year ISI intervals for the requested SG welds and nozzles of MPS2. The examination history shows that no flaws exceeded the ASME Code, Section XI, acceptance standards and that the examination coverage can be as low as about 50 percent. In the supplement dated March 19, 2021, the licensee confirmed that the examination coverage during the first and second 10-year ISI interval would have been no less than the lowest coverage reported during the third 10-year ISI interval (56.3 percent) for the MPS2 SG welds and NIR that were retained after the replacement of the bottom portion of both MPS2 SGs. With respect to examination coverage, the licensee also referred to Section 8.4 of EPRI reports 15906 and 14590 and to the conclusion in Attachment 1 of the submittal, which stated that the PFM and DFM evaluations demonstrated that after PSI, no other inspections are required until 80 years to meet the acceptance criterion of 1E-06 failures per year. In Section 11.0 of this SE, the NRC staff explained its non-acceptance of the PSI-only examinations.

In Section 8.3.5 of EPRI report 15906, EPRI included PFM results that show the effect of 50 percent examination coverage on PoF values, which is further discussed in Section 11.0 of this SE. Even though EPRI report 14590 did not include PFM results that show the effect of less than 100 percent examination coverage, SIA report 1900064.406.R0 (Enclosure 2 of ADAMS Accession No. ML20253A311), included results that show this effect for a 50 percent examination coverage for the limiting case in Table 8-9 of EPRI report 14590, FEW-P3A. Even though SIA report 1900064.406.R0 was part of SNC's plant-specific alternative request, the impact of examination coverage was for case FEW-P3A in Table 8-9 of EPRI report 14590. The PFM results showed that the probability of rupture at 80 years between 100 percent examination coverage and 50 percent examination coverage stayed the same at 1.25E-09 per year. The results also showed that for an ISI schedule of PSI + 10 + 20 + 30 + 60, which bounds the licensee's proposed alternative ISI schedule of PSI + 10 + 20 + 30 + 40 + 70 for MPS2, the probability of leakage at 80 years increased from 2.50E-09 per year to 5.95E-06 per year, which is greater than the criterion of 1E-06 per year. The NRC staff noted that even though there was an increase in probability of leakage going from 100 percent to 50 percent examination coverage, leakage is not component rupture and would be managed by the plant leakage detection system. In addition, the acceptance criterion of 1E-06 failures per year is intended for probability of rupture (see Section 3.0 of this SE) not probability of leakage, and that therefore, applying the criterion to probability of leakage is conservative.

Therefore, because examination coverage was adequately implemented and PFM results for less than 100 percent examination coverage were included as discussed above, the NRC finds that the licensee adequately addressed the effect of examination coverage on the PoF values for the requested SG welds and nozzles of MPS2.

10.0 OTHER CONSIDERATIONS

10.1 Initial Flaw Depth and Length Distribution

In Section 8.3.2.2 of EPRI report 15906 and Section 8.2.2.2 of EPRI report 14590, EPRI stated that the statistical distribution of initial flaw depth used in the PFM analyses is based on the data from the PWR vessel used in the PVRUF project. The NRC staff noted that the PVRUF depth distribution consisted of mostly small fabrication flaws that break the inner surface of the component. Thus, the initial flaws in the PFM analyses consist of small surface-breaking flaws.

The flaw data on which the PVRUF depth distribution are based have been used extensively in PFM analyses of PWR components that have been accepted by the NRC, notably in the development of the technical basis for 10 CFR 50.61a (NUREG-1806 “Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61),” August 2007 [ADAMS Accession No. ML072830074]). Considering these previous analyses, and the discussion in Section 7.0 of this SE on flaw density and the thicknesses of the requested SG welds and nozzles of MPS2 the NRC staff estimated from the figures in Attachment 2 of the submittal, the NRC staff determined for the MPS2 plant-specific alternative request that applying the PVRUF depth distribution to the requested SG welds and nozzles of MPS2 is reasonable.

In Section 8.3.2.2 of EPRI report 15906 and Section 8.2.2.2 of EPRI report 14590, EPRI also described the length distribution used in the PFM analyses. EPRI cited NUREG/CR-6817, “A Generalized Procedure for Generating Flaw-Related Inputs for the FAVOR Code,” dated April 2004 (ADAMS Accession No. ML040830499), and a proprietary SIA document, 1700313.301, which specifically describes the development of the log-normal distribution for the length. As the NRC staff observed in the audit summary report for PROMISE (see Section 3.1 of this SE), the flaw data for the length distribution was derived from the most conservative of three sets of flaw data, and as such, the NRC staff finds for the MPS2 plant-specific alternative request that the length distribution is acceptable for the analysis results referenced for the requested SG welds and nozzles of MPS2.

10.2 Probability of Detection

In Section 8.3.2.3 of EPRI report 15906 and Section 8.2.2.3 of EPRI report 14590, EPRI stated that the POD curve used in the analyses was the same POD curve used in the BWRVIP-108 analyses. The NRC staff confirmed that the POD curve in Figure 8-6 of EPRI report 15906 and Figure 8-2 of EPRI report 14590 is the same as the POD curve in BWRVIP-108. The NRC staff noted that the nozzle-to-shell welds and NIR analyzed in BWRVIP-108 were associated with the reactor pressure vessel and that the POD curve was, therefore, developed based on the ultrasonic testing (UT) requirements in the ASME Code, Section XI, Appendix VIII (this is also reflected in the discussion of POD in the December 19, 2007, SE of BWRVIP-108).

The welds and NIR in the referenced EPRI reports are associated with the SG vessel for which the UT requirements of ASME Code, Section V, apply. The NRC staff noted that, in practice, the POD curve based on the UT requirements of the ASME Code, Section V, could be lower than the POD curve based on the UT requirements of the ASME Code, Section XI, Appendix VIII. To evaluate the acceptability of the Appendix VIII-based POD curve on the SG vessel components for which the UT examination requirements of ASME Code, Section V apply, the

NRC staff assessed the PVRUF cumulative probability distribution shown in Equation 8-1 of both EPRI reports against the Appendix VIII-based POD curve shown in Figure 8-6 of EPRI report 15906 and Figure 8-2 of EPRI report 14590. The PVRUF distribution represented by Equation 8-1 of both EPRI reports, in effect, says that there is about a 90 percent probability that the initial flaw depth used in the PFM analyses is equal to or less than 0.0787 inches. This flaw depth is on the lower portion (left side) of the Appendix VIII-based POD curve. While the NRC staff expects that, in practice, a POD curve based on the ASME Code, Section V, could be lower than the Appendix VIII-based POD curve, it would not be much lower for flaw depths equal to or less than 0.0787 inches, which are flaw depths that are analyzed 90 percent of the time and for which the POD is already very low at about 18 percent.

Based on the discussion above and that POD is not one of the parameters that significantly affects the PFM results (see Section 4.0 of this SE), the NRC staff determined for the MPS2 plant-specific alternative request that a Section V-based POD curve would have minimal impact on the PFM results compared to an Appendix VIII-based POD curve, and, therefore, the Appendix VIII-based POD curve is adequate for use in the PFM analyses referenced for the requested SG weld and nozzles of MPS2.

10.3 Models

The NRC staff evaluated the flaw distribution and POD models in Sections 10.1 and 10.2, respectively, of this SE and the K_{IC} and the FCG rate models in Sections 6 and 8, respectively.

In Section 8.2.2.4 of EPRI report 15906 and Section 8.2.2.5 of EPRI report 14590, EPRI described the fracture mechanics models used in the analyses. For both semi-elliptical circumferential and axial surface cracks in a cylindrical configuration, EPRI employed SIF models from API-579/ASME-FFS-1. Similar crack models were used to analyze postulated flaws in nozzle-to-shell welds in BWRVIP-108 and BWRVIP-241, which the NRC staff approved in SEs dated December 19, 2007 (ADAMS Accession No. ML073600374), and April 19, 2013 (ADAMS Accession No. ML13071A240), respectively.

Since the SG nozzle-to-shell welds (Item Nos. B3.130 and C2.21) of MPS2 are similar in configuration to the components analyzed in BWRVIP-108 and BWRVIP-241, the NRC staff finds for this MPS2 plant-specific alternative request that the cylindrical SIF models in both EPRI reports are appropriate for the postulated flaws in the requested SG nozzle-to-shell welds of MPS2.

The NRC staff finds for this MPS2 plant-specific alternative request that the cylindrical SIF models in both EPRI reports appropriate for the postulated flaws in the requested SG shell welds (Item No. C1.10) of MPS2 because these welds are located in the cylindrical portions of the MPS2 SG shell.

The NRC staff finds for the MPS2 plant-specific alternative request that the cylindrical SIF models in both EPRI reports appropriate for the postulated flaws in the requested SG tubesheet-to-head welds and tubesheet-to-shell welds (Item Nos. B2.40 and C1.30) of MPS2 because there is less geometric constraint in the cylindrical SIF model compared to the geometric constraint in these MPS2 welds; this means that for the same stress, the cylindrical SIF model would generate a larger, and therefore conservative, applied SIF value compared to the applied SIF value for the same postulated flaw in these MPS2 welds. The bulk of the tubesheet imposes added restraint to the postulated flaws in these welds, which would lower the applied SIF compared to the cylindrical model.

For the postulated flaws in the requested SG head welds (Item Nos. B2.31 and C1.20) of MPS2, the NRC staff compared the applied SIF equation for a postulated axial flaw in a cylinder given in API-579/ASME-FFS-1 to the applied SIF equation for the same flaw in a sphere. The applied SIF equation in a sphere applies to postulated flaws in the SG head welds. The NRC staff determined that the applied SIF for the cylindrical model is slightly higher than the applied SIF for the spherical model. Therefore, the NRC staff finds for the MPS2 plant-specific alternative request that the cylindrical SIF models in the EPRI reports are also appropriate for the postulated flaws in the requested SG head welds of MPS2.

In Section 8.2.2.5 of EPRI report 14590, EPRI described the NIR crack model. EPRI employed a weight function-based SIF solution for the model. In the audit summary report for PROMISE (see Section 3.1 of this SE), the NRC staff described the V&V effort for the NIR crack model. The NRC staff finds the NIR crack model acceptable because it provided similar SIF values compared to SIF values from finite element analysis. Thus, the NRC staff finds for the MPS2 plant-specific alternative request that the NIR crack model in EPRI report 14590 is appropriate for the postulated flaws in the requested SG MS and FW NIR (Item No. C2.22) of MPS2.

10.4 <u>Uncertainty</u>

In Section 8.3.1.2 of EPRI report 15906 and Section 8.2.1.2 of EPRI report, EPRI considered both aleatory uncertainty (random or inherent uncertainty) and epistemic uncertainty (uncertainty due to state of knowledge) and stated that these uncertainties entailed two sampling loops: an aleatory and epistemic loop. In Section 8.3.4.1 of EPRI report 15906 and Section 8.2.4.1 of EPRI report 14590, EPRI stated that it considered all random parameters aleatory because they are conservative or based on large sets of data (for example, the FCG distribution was developed from over 1,000 fatigue datapoints developed in PWR water environments). EPRI performed 10 million aleatory realizations and 1 epistemic realization.

The NRC staff noted that representing all variables as aleatory will result in probabilities that represent the mean of the distribution. From the distribution of results presented, the NRC staff noted that the 50th percentile (median) was very close to the results when the licensee assumed all aleatory realizations (mean). This is expected for a distribution that is normally distributed but may be different for a skewed distribution.

In the audit summary report for PROMISE (see Section 3.1 of this SE), the NRC staff documented observations on percent error and implementation of aleatory and epistemic realizations. With regard to the observation on percent error, the NRC staff notes that large percent errors that result from probabilistic analyses where only one failure happens in 10 million realizations can be impactful if the results approach the acceptance criteria. Assuring sufficient realizations and proper sampling of the input space will reduce the error with these calculations. In addition, the overuse of conservative inputs in a probabilistic analysis can mask the importance of other random variables and should be avoided. However, since the limiting location for the base cases in both EPRI reports had probabilities of leakage and rupture more than two orders of magnitude below the acceptance criterion of 1E-06 per year, the NRC staff finds for the MPS2 plant-specific alternative request that the large uncertainty in the low probability results reflected by the large percent error is reasonable.

In Table 8-8 of EPRI report 15906 and Table 8-7 of EPRI report 14590, EPRI indicated that there were no uncertainties in the transient stresses. The NRC staff finds for the MPS2 plant-specific alternative request that treating transient stresses as constant rather than random is acceptable since the transients were selected based on large temperature and pressure variations, as discussed in Section 5.2 of this SE.

Uncertainties on other random parameters not covered in this section are discussed in other sections of this SE (e.g., Section 8 for FCG rate).

Based on the discussion above, the NRC staff finds for the MPS2 plant-specific alternative request that the licensee's handling of uncertainty is acceptable for the analysis results referenced for the requested SG welds and nozzles of MPS2.

10.5 Convergence

In Section 8.3.4.3.10 of EPRI report 15906 and Section 8.2.4.3.10 of EPRI report 14590, EPRI conducted a SS to determine if the number of realizations resulted in a converged solution in the PoF values. The results shown in Table 8-31 of EPRI report 15906 and Table 8-27 of EPRI report 14590, and described in the NRC staff's observation in the audit summary report for PROMISE (see Section 3.1 of this SE) on convergence indicate little difference in PoF values between 10^7 and 10^8 realizations. Based on these results, the NRC staff finds for the MPS2 plant-specific alternative request that the number of realizations used in the analyses, 10^7 realizations, is acceptable for the results referenced for the requested SG welds and nozzles of MPS2. This number of realizations is acceptable even though the uncertainty is high for those cases where only one failure occurs within an analysis, as described in Section 10.4 of this SE.

10.6 DFM Analysis

In Section 8.2 of EPRI report 15906 and Section 8.3 of EPRI report 14590, EPRI performed DFM analyses with an initial flaw depth based on the maximum depth specified in the ASME Code, Section XI, acceptance standards and average values of all other parameters considered random in the PFM analyses. All analyzed locations resulted in many years to reach leakage (80 percent of the component thickness), the least being 147 years. No locations reached an applied SIF of greater than 200 ksi√in, but the FW NIR resulted in an applied SIF of 101 ksi√in at 80 years, which, considering a factor safety of 2, is greater than the allowable SIF value of 200/2 = 100 ksi√in. The NRC staff finds this acceptable for MPS2 since the MPS2 FW NIR received 100 percent coverage and indications detected were below the ASME Code, Section XI, acceptance standards, and the results of the DFM were for a postulated flaw with a depth equivalent to the specified depth in the ASME Code, Section XI, acceptance standards. The NRC staff also noted that the results for the FW NIR in EPRI report 14590 have some conservatism because the applied loads include the effect of welding residual stress (see

Section 5.3 of this SE). Thus, the NRC staff determined for the MPS2 plant-specific alternative request that overall the DFM analyses support the PFM analyses.

11.0 PFM RESULTS RELEVANT TO ALTERNATIVE RELIEF REQUEST (RR)-05-06

In Section 6 of Attachment 1 to the submittal, the licensee stated that based on the PFM results, after PSI, no other inspections are required for up to 80 years of plant operation to meet the acceptance criterion of 1E-06 failures per year. A similar observation is in Section 8.3.4.1 of EPRI report 15906 and Section 8.2.4.1.1 of EPRI report 14590, which states that performing only PSI examination without any other post PSI examinations is acceptable for 80 years of plant operation while maintaining plant safety. The NRC staff does not find either general conclusion acceptable since it does not account for the effect of the combination of the most significant parameters or the added uncertainty of low probability events.

With respect to EPRI report 15906, the NRC staff determined that since the PFM analyses in the report were based on representative SG vessels and representative primary side nozzles, the uncertainties on the different parameters (which are different from the sampling uncertainty discussed in Section 10.4 of this SE) should be taken into account, especially those from the significant parameters of stress and K_{IC} before a general conclusion can be made on PSI-only examinations. Also, the effects of other significant parameters such as examination coverage should be considered. As an example, even for a case more favorable than PSI-only examination, such as PSI + 10 + 20 + 40 + 60, the probability of rupture at 80 years for the limiting location changed from 5.00E-09 per year to 6.46E-06 per year (Table 8-33 of EPRI report 15906), which exceeds the criterion of 1E-06 per year. While the NRC staff acknowledged that this study assumed conservative values for stress and K_{IC} simultaneously (thereby, accounting for uncertainties in these two parameters) and an examination coverage commonly encountered in the field, the NRC staff also noted that had the same study been performed for the PSI-only case, the probability of rupture values would have been much higher, and that only one or two changed parameters out of the three could easily lead to probability of ruptures greater than 1E-06 per year.

With respect to EPRI report 14590, the NRC staff determined that since the PFM analyses in the report were based on representative SG MS and FW nozzles, the uncertainties on the different parameters (which are different from the sampling uncertainty discussed in Section 10.4 of this SE) should be taken into account, especially the significant parameters of stress, fracture toughness, and flaw density, before a general conclusion can be made on PSI-only examinations. As an example, even for a case more favorable than PSI-only examination, such as PSI + 20 + 40 + 60, the probability of rupture at 80 years for the FW NIR changes from 1.25E-12 per year to 5.3E-06 per year (Table 8-28 of EPRI report 14590), which exceeds the criterion of 1E-06 per year. This was a result of a SS that changed at the same time stress, fracture toughness, and flaw density at the NIR to more conservative values, which accounted for uncertainties in these parameters. While the NRC staff acknowledged that this study assumed conservative values for all three parameters simultaneously, the NRC staff also noted that had the same study been performed for the PSI-only case, the probability of rupture values would have been much higher, and that only one or two parameters with conservative values could easily lead to probability of ruptures greater than 1E-06 per year.

Given the discussion on uncertainty above and in Section 10.4 of this SE, the NRC staff determined that uncertainty in the PFM results need to be addressed through sufficient realizations and proper sampling before general conclusions can be considered for the PSI-only cases. Lastly, the NRC staff observed that PSI-only examinations, as compared to the proposed alternative of PSI + 10 + 20 + 30 + 40 + 70, would have a much more adverse effect on risk-informed principles, particularly since PSI-only examinations would eliminate any future performance monitoring needed for risk-informed decision making.

As discussed in Section 9.0 of this SE, the licensee is seeking the alternative ISI schedule of PSI + 10 + 20 + 30 + 40 + 70. Therefore, the NRC staff determined that the PFM results for PSI + 10 + 20 + 30 + 40 + 70 are the results relevant to the licensee's proposed alternative, as discussed next.

11.1 SG vessel primary side and secondary side welds (Item Nos. B2.31, B2.40, B3.130, C1.10, C1.20, and C1.30)

The NRC staff noted that even though EPRI 15906 report does not have PoF results for PSI + 10 + 20 + 30 + 40 + 70, it has results for PSI + 20 + 40 + 60 or PSI + 10 + 20 + 40 + 60, either of which would bound the former since ISI is implemented more times in the former. Therefore, the NRC staff evaluated the PFM results in the SS in Section 8.3.4.3 of EPRI report 15906 relevant to the proposed alternative ISI schedule of PSI + 10 + 20 + 30 + 40 + 70 by assessing the results for PSI + 20 + 40 + 60 or PSI + 10 + 20 + 40 + 60.

Table 8-33 of EPRI report 15906 shows the probability of rupture results for the SS on the combined effect of fracture toughness, stress, and 50 percent examination coverage. These probability of rupture results are for an ISI schedule of PSI + 10 + 20 + 40 + 60, which bounds the licensee's proposed alternative of PSI + 10 + 20 + 30 + 40 + 70 for the reasons the NRC staff previously stated. As shown in Table 8-33, the limiting probability of rupture is 6.46E-06 per year, which exceeds the criterion of 1E-06 per year. The NRC staff discussed the acceptability of the K_{IC} model of 200 ksi√in with standard deviation of 5 ksi√in in Section 6.0 of this SE on K_{IC}.

Therefore, the NRC staff evaluated the effect of the other two parameters, stress and examination coverage, assuming a mean K_{IC} and its standard deviation (200 ksi√in and 5 ksi√in, respectively) were the same as those in the base case. Tables 8-19 and 8-20 of EPRI report 15906 contain the SS on stress for an ISI schedule of PSI + 20 + 40 + 60, but they don't show the effect of examination coverage. The inspection history of the SG welds and nozzles of MPS2 in Attachment 3 of the submittal shows that a B2.31 weld had an examination coverage as low as about 50 percent. Table 8-19 of EPRI report 15906 shows that the probability of rupture for the limiting location, SGPTH-P4A (referenced for the Item Nos. B2.31 and B2.40 welds of MPS2), is 7.33E-07 per year, which is close to but less than the criterion of 1E-06 per year. This probability of rupture value of 7.33E-07 per year, however, is with a stress multiplier of 1.90. In the supplement dated March 19, 2021, the licensee included additional PFM results for the Item No. B2.31 welds of MPS2 using the MPS2-specific SG model (see Section 5.1 of this SE). The probability of rupture value from the SS on stress from the licensee's additional PFM evaluations is bounded by 7.33E-07 per year. The NRC staff noted that without other tables that show the effect of examination coverage specifically on the limiting location (SGPTH-P4A) in the SS on stress, the NRC staff cannot determine the impact of a 50 percent examination coverage on the limiting value of 7.33E-07 per year from the SS on stress. The licensee also included in the supplement the thickness of 7.0 inches of the MPS2 SG lower head and the R/t ratio of 11.7 of the MPS2 SG lower head. The NRC staff noted that for the base case in Table 8-19 of EPRI report 15906, the R/t ratio of the modeled SG lower head is 11.90 per the dimensions given in Table 4-2 of EPRI report 15906. Since the R/t ratio of the MPS2 SG lower head is lower than the base case value and a lower R/t ratio reflects the stress level in the SG lower head, the NRC staff determined that for the limiting location, SGPTH-P4A, the probability of rupture value of 1.25E-09 per year for the base case in Table 8-19 of EPRI report 15906 (with a stress multiplier of 1 instead of 1.90) can be applied to MPS2.

The NRC staff noted that the increase due to a 50 percent examination coverage on the probability of rupture value is about 350 times on the limiting location, SGPTH-P4A, from Tables 8-32 and 8-33 of EPRI report 15906. Given that the probability of rupture for MPS2 is 1.25E-09 per year as discussed above, the NRC staff calculated a conservative estimate of the impact of a 50 percent examination coverage on this value, which is 350 x (1.25E-09 per year) = 4.38E-07 per year, which is less than the criterion of 1E-06 per year. The NRC staff noted that applying the multiplier of 350 to the probability of rupture value of 1.25E-09 per year from Table 8-19 of EPRI report 15906 is conservative because the results in Tables 8-32 and 8-33 of EPRI report 15906 include the combined effects of the standard deviation of 30 ksi√in on K_{IC} and stress multiplier of 1.33, both of which tend to increase the probability of rupture.

Finally, the NRC staff noted that since the licensee's proposed alternative is through 60 years of operation, the probability values should be based on 60 years of operation. Tables 8-19 and 8-20 of EPRI report 15906 are for 80 years of operation and at 60 years of operation, the results could be up to 80/60 = 1.3 times larger since the number of failures would be divided by 60 years instead of 80 years (assuming the number of failures have been reached by 60 years). As discussed in Section 3.0 of this SE, PoF at a given time is estimated as the fraction of the total number of realizations that the computed failure time is less than the given time. In short, this means that PoF is the number of failure times within a given time divided by the total number of realizations. For instance, if the given time is 60 years, PoF is the number of failure times that are less than 60 years divided by the total number of realizations. Since the number of failure times could be reached before 60 years, the PoF value could be the same at 60 years and at 80 years. And since the licensee's proposed alternative is through 60 years of operation, this PoF value should be divided by 60 years instead of 80 years to obtain the PoF per year value. The NRC staff determined that the factor of 1.3 has no impact on the NRC staff's conclusion on the probabilities shown in Tables 8-19 and 8-20 of EPRI report 15906. Thus, the NRC staff determined that the PFM analyses in EPRI report 15906 adequately address uncertainties in the PoF values relevant to the licensee's proposed alternative of PSI + 10 + 20 + 30 + 40 + 70 for the requested SG primary side and secondary side welds of MPS2.

11.2 <u>SG MS and FW nozzle-to-shell welds and NIR (Item Nos. C2.21 and C2.22)</u>

The NRC staff noted that even though EPRI 14590 report does not have PoF results for PSI + 10 + 20 + 30 + 40 + 70, it has results for PSI + 20 + 40 + 60, which would bound the former since ISI is implemented more times in the former. Therefore, the NRC staff evaluated the PFM results in the SS in Section 8.2.4.3 of EPRI report 14590 relevant to the proposed alternative ISI schedule of PSI + 10 + 20 + 30 + 40 + 70 by assessing the results for PSI + 20 + 40 + 60.

Table 8-28 of EPRI report 14590 shows the probability of rupture results for the SS on the combined effect of fracture toughness, stress, and flaw density. These probability of rupture results are for an ISI schedule of PSI + 20 + 40 + 60, which bounds the licensee's proposed alternative of PSI + 10 + 20 + 30 + 40 + 70 for the reasons the NRC staff previously stated. The probability of rupture value is 5.30E-06 per year for the FW NIR, which is the limiting case. The value of 5.30E-06 per year is greater than 1E-06 per year, and the parameters changed were the following: stress (multiplier of 1.0 to 1.5), standard deviation on upper shelf K_{IC} (5 ksi√in to 30 ksi√in), and flaw density (0.001 to 0.1 flaw per nozzle). The NRC staff discussed the acceptability of the K_{IC} model of 200 ksi√in with standard deviation of 5 ksi√in in Section 6.0 of this SE on fracture toughness.

Therefore, the NRC staff evaluated the effect of the other two parameters – stress and flaw density, assuming a mean K_{IC} and standard deviation (200 ksi√in and 5 ksi√in, respectively) were the same as those in the base case. Tables 8-15 and 8-16 of EPRI report 14590 contain the SS on stress. Table 8-15 of EPRI report 14590 shows that the probabilities of rupture are well below the criterion of 1E-06 per year, even when considering the 0.1 flaw per nozzle at the MS and FW NIR, which would change the probability of rupture results from 1.25E-12 to 1.25E-10 per year (see discussion in Section 7.0 of this SE on NIR flaw density as a multiplier). Table 8-16 of EPRI report 14590 shows that the limiting nozzle-to-shell weld probability of leakage is 1.04E-06 per year and the limiting NIR probability of leakage is 1.06E-06 per year (adjusted for 0.1 flaw per nozzle). Even though these values are greater than the criterion of 1E-06 per year, the NRC staff finds them acceptable because they are leakage probability values as opposed to rupture probability values. The NRC staff also noted that the results for the MS and FW NIR have some conservatism because the applied loads include the effect of welding residual stress (see Section 5.3 of this SE).

Finally, the NRC staff noted since the licensee's proposed alternative is through 60 years of operation, the probability values should be based on 60 years of operation. Tables 8-15 and 8-16 of EPRI report 14590 are for 80 years of operation and at 60 years of operation, the results could be up to 80/60 = 1.3 times larger since the number of failures would be divided by 60 years instead of 80 years (assuming the number of failures have been reached by 60 years). The NRC staff determined that this factor of 1.3 has no impact on the NRC staff's conclusion on the probabilities shown in Tables 8-15 and 8-16 of EPRI report 14590. Thus, the NRC staff determined that the PFM analyses in EPRI report 14590 adequately address uncertainties in the PoF values relevant to the licensee's proposed alternative of PSI + 10 + 20 + 30 + 40 + 70 for the requested SG MS and FW nozzle-to-shell welds and NIR of MPS2.

11.3 Conclusion on PFM Results Relevant to Alternative RR-05-06

Based on the discussion above, the NRC staff finds that the proposed alternative of PSI + 10 + 20 + 30 + 40 + 70 for the requested SG welds and nozzles of MPS2 would result in a PoF per year that is reasonably below the acceptance criterion of 1E-06 per year.

CONCLUSION

The NRC staff has determined that the proposed alternative in the licensee's request referenced above would provide an acceptable level of quality and safety.

The NRC staff concludes that the licensee has adequately addressed the regulatory requirements set forth in 10 CFR 50.55a(z)(1).

The NRC staff authorizes the use of proposed alternative RR-05-06 at MPS2 for the remainder of the fifth 10-year ISI interval and into the following sixth 10-year ISI interval up to the end of the 60-year operating license, which expires July 31, 2035. The NRC did not review the referenced EPRI reports for generic use, and this approval does not extend beyond the MPS2 plant-specific authorization.

All other ASME Code, Section XI, requirements for which an alternative was not specifically requested and authorized remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributors: David Dijamco
David Rudland

Date: July 16, 2021

James G. Danna

Digitally signed by James G. Danna
Date: 2021.07.16 11:46:38 -04'00'

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

cc: Listserv

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 2 - AUTHORIZATION AND SAFETY EVALUATION FOR ALTERNATIVE REQUEST NO. RR-05-06 (EPID L-2020-LLR-0097) DATED JULY 16, 2021

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NAME	RGuzman	KEntz	SRohrer /f/Burkhart
DATE	6/21/2021	6/21/2021	6/21/2021
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NAME	DWidrevitz	JDanna	
DATE	4/26/2021	7/15/2021	

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