UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 2, 2021

Mr. David P. Rhoades
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LIMERICK GENERATING STATION, UNIT 1 AND 2 – ISSUANCE OF RELIEF REQUEST I4R-24 ASSOCIATED WITH RESIDUAL HEAT REMOVAL HEAT EXCHANGER CATEGORY C-A AND C-B EXAMINATIONS FOR THE REMAINDER OF THE FOURTH 10-YEAR ISI INTERVAL AND UP TO THE END OF THE 60-YEAR OPERATING LICENSES (EPID L-2020-LLR-0122)

Dear Mr. Rhoades:

By letter dated September 4, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20252A135), as supplemented by letter dated April 1, 2021 (ADAMS Accession No. ML21091A041), Exelon Generation Company, LLC (the licensee), submitted a request for relief from the requirements of the American Society of Mechanical Engineers Boiler & Pressure Vessel Code (ASME Code), ASME Section XI IWC-2500(a), Table IWC-2500-1, Examination Categories C-A and C-B for Limerick Generating Station, Units 1 and 2 (Limerick).

Specifically, pursuant to the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), "Acceptable level of quality and safety," the licensee requested increasing the inspection interval for the residual heat removal (RHR) heat exchanger welds and nozzles examinations to the end of the current operating licenses for Unit 1 an Unit 2 in lieu of the current ASME Code, Section XI, Division 1 10-year inspection frequency.

The U.S. Nuclear Regulatory Commission (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 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 I4R-24 at Limerick, for the remainder of the fourth 10-year ISI interval and up to the end of the 60-year operating license of each unit (October 26, 2044, for Unit 1) and (June 22, 2049, for Unit 2).

The NRC's authorization of the proposed alternative does not infer or imply the approval of the Electric Power Research Institute (EPRI) Report 18473 for generic use.

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.

If you have any questions, please contact the Limerick Project Manager, V. Sreenivas, at 301-415-2597 or V.Sreenivas@nrc.gov.

Sincerely,

/RA/

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

Docket Nos.: 50-352
50-353

Enclosure:
Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PROPOSED ALTERNATIVE REQUEST I4R-24 REGARDING RESIDUAL HEAT REMOVAL

HEAT EXCHANGER CATEGORY C-A AND C-B EXAMINATIONS

EXELON GENERATION COMPANY, LLC.

LIMERICK GENERATING STATION, UNIT 1 AND UNIT 2

DOCKET NOS. 50-352 AND 50-352

1.0 INTRODUCTION

By letter dated September 4, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20252A135), as supplemented by letter dated April 1, 2021 (ADAMS Accession No. ML21091A014, Exelon Generation Company, LLC (the licensee), submitted a request for relief from the requirements of the American Society of Mechanical Engineers Boiler & Pressure Vessel Code (ASME Code), ASME Section XI IWC-2500(a), Table IWC-2500-1, Examination Categories C-A and C-B for Limerick Generating Station (Limerick or LGS), Units 1 and 2. Specifically, pursuant to the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), "Acceptable level of quality and safety," the licensee requested increasing the inspection interval for the residual heat removal (RHR) heat exchanger welds and nozzles examinations to the end of the current operating licenses for Unit 1 and Unit 2 in lieu of the current ASME Code, Section XI, Division 1 10-year inspection frequency.

The U.S. Nuclear Regulatory Commission (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 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 I4R-24 at Limerick for the remainder of the fourth 10-year inservice inspection (ISI) interval and up to the end of the 60-year operating license of each unit (October 26, 2044, for Unit 1) and (June 22, 2049, for Unit 2).

2.0 REGULATORY EVALUATION

The NRC regulations in 10 CFR 50.55a(z) state, in part, that alternatives to the requirements of paragraph (f) of 10 CFR 50.55a may be used, when authorized by the NRC, if the licensee demonstrates (1) the proposed alternatives would provide an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Proposed Alternative

The licensee requested alternative I4R-24 for the residual heat removal (RHR) heat exchanger welds and nozzles of Limerick Units 1 and 2 shown in the table below. The licensee stated that the proposed alternative is to increase the ISI interval for these components from the current ASME Code, Section XI, Division 1 10-year requirement to the end of the current operating licenses, which are October 26, 2044 and June 22, 2049 for Limerick Units 1 and 2, respectively. The licensee stated that the increase in ISI interval equates to an extension of 27 years, 8 months, 25 days for Unit 1 and 32 years, 4 months, 21 days for Unit 2 from the end of the third ISI Interval (January 31, 2017) at which time all ASME Code, Section XI, Division 1 requirements were satisfied.

Limerick Unit1	ASME Code Category	ASME Code Item No.	Component ID	Description
1	C-A	C1.10	RHR-HXAR-4	Shell Ring 1 to Flange Weld
1	C-A	C1.10	RHR-HXBR-4	Shell Ring 1 to Flange Weld
2	C-A	C1.10	2AE-205 SG-1	Shell (Ring #1) to Flange Weld
2	C-A	C1.10	2BE-205R W20	Shell Ring 1 to Flange Weld
1	C-A	C1.20	RHR-HXAR-1	Head to Shell Ring 3 Weld
1	C-A	C1.20	RHR-HXBR-1	Head to Shell Ring 3 Weld
2	C-A	C1.20	2AE-205 SG-5	Shell Head to Shell Weld (Ring #4)
2	C-A	C1.20	2BE-205R W26	Shell Ring 3 to Shell Head Weld
1	C-B	C2.21	RHR-HXAR-N3	Nozzle to Head Weld
1	C-B	C2.21	RHR-HXAR-N4	Nozzle to Shell 1 Weld
1	C-B	C2.21	RHR-HXBR-N3	Nozzle to Head Weld
1	C-B	C2.21	RHR-HXBR-N4	Nozzle to Shell 1 Weld
2	C-B	C2.21	2AE-205 N-3-1	Inlet Nozzle (N-3) to Shell Head Weld
2	C-B	C2.21	2BE-205R N-4-1	Outlet Nozzle (N-4) to Shell Weld
2	C-B	C2.21	2AE-205 N-4-1	Outlet Nozzle (N-4) to Shell Weld
1	C-B	C2.22	RHR-HXAR-N3IR	Nozzle N3 Inner Radius
1	C-B	C2.22	RHR-HXAR-N4IR	Nozzle N4 Inner Radius
1	C-B	C2.22	RHR-HXBR-N3IR	Nozzle N3 Inner Radius
1	C-B	C2.22	RHR-HXBR-N4IR	Nozzle N4 Inner Radius
2	C-B	C2.22	2AE-205 N-3-1 Inner Radius	Inlet Nozzle (N-3) Inner Radius
2	C-B	C2.22	2BE-205R N-4-1 IR	Outlet Nozzle (N-4) Inner Radius
2	C-B	C2.22	2AE-205 N-4-1 Inner Radius	Outlet Nozzle (N-4) Inner Radius

3.2 NRC Staff Evaluation

3.2.1 Licensee's Basis for Proposed Alternative

The licensee referred to the results of the probabilistic fracture mechanics (PFM) analyses in the following Electric Power Research Institute (EPRI) report as the primary basis for proposing to increase the ISI interval for the requested Limerick Units 1 and 2 RHR heat exchanger welds and nozzles: non-proprietary EPRI Report 3002018473, "Technical Bases for Examination Requirements for Class 2 BWR Heat Exchanger Nozzle-to-Shell Welds; Nozzle Inside Radius Sections; and Vessel Head, Shell, and Tubesheet-to-Shell Welds," 2020 (included as Enclosure 1 to the supplement dated April 1, 2021) report support the PFM results. The NRC staff reviewed the proposed alternative request for Limerick Units 1 and 2 as a plant-specific alternative. The NRC did not review EPRI Report 18473 for generic use, and this alternative request does not extend beyond the Limerick Units 1 and 2 plant-specific authorization.

The NRC staff's review focused on evaluating the PFM analyses in Section 8.3 of EPRI Report 18473 and verifying whether the deterministic fracture mechanics (DFM) analyses in the

3.2.2 Degradation Mechanisms

In Section 5.0 of the attachment to the submittal, the licensee referred to the evaluation of potential degradation mechanisms in EPRI Report 18473 and concluded that other than corrosion fatigue (also referred to as environmental assisted fatigue in EPRI Report 18473) and mechanical/thermal fatigue, there were no active degradation mechanisms identified that significantly affect the long-term structural integrity of the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2.

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 3.3.7 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 Limerick Units 1 and 2 plant-specific alternative request because: (1) FCG is known to be the dominant crack driving force in ferritic materials such as the RHR heat exchanger welds and nozzles of Limerick Units 1 and 2 (see Section 3.3.5.1 of this SE); and (2) ferritic materials are known to be highly resistant to stress corrosion cracking under the operating conditions of the RHR heat exchangers of Limerick Units 1 and 2.

3.3.3 Overall PFM Approach

The PFM analyses in EPRI Report 18473 were performed with the **PR**obabilistic **O**pti**M**ization of **I**n**S**p**E**ction (PROMISE) Version 2.0 software. The verification and validation (V&V) of the software is discussed in Section 3.3.3.1 of his SE. The software will be referred to as PROMISE from this point forward unless otherwise noted.

The overall PFM approach in EPRI Report 18473 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 3.3.10.4 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 of 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 18473 defines 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 value is then determined on a per year basis and compared to the acceptance criterion of 1E-06 per year.

The NRC staff finds the overall PFM approach acceptable for the plant-specific alternative request for Limerick Units 1 and 2 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 Package 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 RHR heat exchanger welds and nozzles of Limerick Units 1 and 2 because the impact of an RHR heat exchanger vessel failure is less than the impact of a reactor vessel failure on overall risk. The NRC staff noted that the RHR heat exchangers at Limerick Units 1 and 2 are safety significant because of their heat removal function, but they are not in the same level of safety significance as the reactor vessel. In the supplement dated April 1, 2021, the licensee clarified that it has not implemented ASME Code Case N-716-1, which could allow reclassification of RHR heat exchangers as low safety significance. 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 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-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.

3.3.3.1 *Software V&V*

In Section 5.0 of the attachment to the submittal, the licensee stated that the alternative request for Limerick Units 1 and 2 uses PROMISE Version 2.0. The licensee also stated that the previous version of PROMISE, PROMISE Version 1.0, was used in another EPRI Report referenced as the technical basis for an alternative request by Southern Nuclear Operating Company (SNC, ADAMS Accession No. ML20253A311). As part of the review of SNC's alternative request, 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. During the audit, the NRC staff reviewed the V&V plan and the documents for the test cases that were performed to implement the plan. The NRC staff issued the audit summary report by letter dated December 10, 2020 (ADAMS Accession No. ML20258A002). The NRC staff issued its SE of the SNC submittal by letter dated January 11, 2021 (ADAMS Accession No. ML20352A155).

As documented in the audit summary report, the NRC staff requested benchmarking runs with another PFM software, VIPERNOZ, contained in Structural Integrity Associates (SIA) report 1900064.407.R2 (Enclosure 3 in ADAMS Accession No. ML20253A311). Even though the NRC staff 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). While SIA report 1900064.407.R2 was submitted as part of the plant-specific submittal by SNC, the benchmarking runs were performed with generic stresses instead of plant-specific stresses. The NRC staff reviewed the benchmark runs in 1900064.407.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. EPRI performed benchmarking runs with PROMISE Version 2.0 in Section 8.3.3.2 of EPRI Report 18473. The NRC staff noted that with the 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 18473 is not necessary because of the adequate V&V performed for the difference between the two versions, as discussed next.

Because the NRC staff has already reviewed the V&V of PROMISE Version 1.0 as discussed above, the NRC staff determined that for the current alternative request for Limerick Units 1 and 2, only the difference between PROMISE Version 2.0 and PROMISE Version 1.0 needs to be reviewed. In Section 5.0 of the attachment to the submittal, the licensee summarized the difference between PROMISE Version 2.0 and PROMISE Version 1.0: "The only technical difference between the two versions is that in PROMISE Version 1.0, the user-specified examination coverage is applied to all inspections, whereas in PROMISE Version 2.0, the examination coverage can be specified by the user uniquely for each inspection." In the supplement dated April 1, 2021, the licensee described the V&V performed for this difference. 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 above, the NRC staff finds for the Limerick Units 1 and 2 plant-specific alternative request that PROMISE Version 2.0 received adequate V&V, and therefore, is acceptable for use in the licensee's plant-specific alternative request for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2.

3.3.4 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, fracture toughness, stress, crack growth rate, and ISI schedule, all of which may be further defined by sub-parameters (such as the exponential term in the crack growth rate). Analysts typically use two sensitivity tools to understand the effects of the input parameters. Sensitivity analyses help identify the major contributors to the final PoF value, and sensitivity studies help in determining the impact of each parameter to the final PoF value.

In Section 8.3.4.2 of EPRI Report 18473, EPRI performed sensitivity analyses to determine the dominant parameters that contribute to the probability of leak and rupture for a representative RHR heat exchanger vessel weld. The results of these sensitivity analyses are in Tables 8-11 and 8-12 of the report. For probability of leakage, EPRI determined that the most dominant contributor is FCG rate coefficient, and for probability of rupture, EPRI determined that the most dominant contributor is fracture toughness.

The NRC staff reviewed the overall results of the sensitivity analyses in EPRI Report 18473 with respect to the Limerick Units 1 and 2 plant-specific alternative request. 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 fracture toughness since applied SIF and fracture toughness (represented by K_{IC}) are the two main parameters in the governing expression in LEFM: applied SIF is greater than (<) K_{IC}. Thus, the NRC staff finds that fracture toughness being the dominant contributor to probability of rupture to be reasonable. The NRC staff noted that even though applied SIF did not come out as the dominant contributor in the sensitivity analyses in EPRI Report 18473, it is one of the significant parameters reflected in the parameter of stress in the sensitivity studies in EPRI Report 18473, as discussed in the next paragraph.

In Section 8.3.4.3 of EPRI Report 18473, EPRI performed sensitivity studies on the following parameters: stress, fracture toughness, crack size distribution, and FCG rate. EPRI concluded that all four are significant contributors to the PFM results. As with the sensitivity analyses results for probability of leakage, the NRC staff finds the overall result of the sensitivity studies on FCG rate reasonable for the Limerick Units 1 and 2 plant-specific alternative request since FCG rate is a measure of how fast the postulated crack would grow to 80 percent of the wall thickness. Similarly, as with the sensitivity analyses results for probability of rupture, the NRC staff finds the overall result of the sensitivity studies on stress and fracture toughness reasonable for the Limerick Units 1 and 2 plant-specific alternative request since these are the parameters that directly affect the governing expression in LEFM. The NRC staff noted that, unlike the other EPRI PFM analyses for steam generator welds and pressurizer welds[1], crack size distribution was reported as a significant parameter in EPRI Report 18473 because the

[1] These other EPRI PFM analyses are in non-proprietary EPRI report number 3002014590 (ADAMS Accession No. ML19347B107) and non-proprietary EPRI report number 3002015905 (ADAMS Accession No. ML21021A271).

crack size distribution used in the analysis (solid line in Figure RAI-230-1 of the supplement dated April 1, 2021) was compared to a much better distribution (the “Chapman” curves in Figure RAI-230-1). Further discussion of the crack size distribution is in Section 3.3.8 of this SE.

During the audit of PROMISE, 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 was a generic observation of the NRC staff on the PFM methodology.

The sensitivity analyses, sensitivity studies, and the NRC staff’s observations on the PROMISE software thus identified the following significant parameters or aspects of the PFM analyses in EPRI Report 18473 that warrant a close evaluation: stress analysis, fracture toughness, FCG rate coefficient (or simply FCG rate), crack size distribution, and effect of ISI schedule and examination coverage. The NRC staff discussed and closely evaluated each in the next five sections of this SE. The NRC staff also evaluated other parameters or aspects of the analyses in Section 3.3.10 of this SE.

3.3.5 Stress Analysis

3.3.5.1 *Selection of components and materials*

In Appendix A of the attachment to the submittal, the licensee evaluated the plant-specific applicability of the components and materials selected and analyzed in EPRI Report 18473 to the RHR heat exchanger welds of Limerick Units 1 and 2. The licensee showed that Limerick Units 1 and 2 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 EPRI Report 18473, which the NRC staff evaluated below.

In Section 4.6 of EPRI Report 18473, EPRI discussed the selection of representative RHR heat exchanger locations for analysis. In selecting the locations, EPRI considered the variation in design in the operating fleet, as documented in Section 4.4 of the report. This included geometry, operating characteristics, materials, and design requirements. EPRI concluded that variations in the design of RHR heat exchangers are not significant, and that the most important design parameter is the radius-to-thickness (R/t) ratio of the RHR shell or affixed nozzles. Table 4-1 of EPRI Report 18473 summarizes the variation in the RHR heat exchanger dimensional parameters identified. Table 4-3 of the report shows the R/t ratio evaluated. Examples of RHR heat exchanger designs were also provided in Figures 4-1 and 4-2 of EPRI Report 18473. The licensee provided figures detailing the RHR heat exchanger designs of Limerick Units 1 and 2 in Figures A-1 through A-4 of the submittal attachment. The NRC staff reviewed the figures and representative EPRI dimensional parameters and found them to be sufficiently similar to produce useful results.

EPRI noted that axial and hoop stresses in the RHR heat exchanger locations were designed based on a fixed ratio relative to the design pressure, and consequently the R/t ratio would relate reliably to the modeled pressure-induced stress. Consequently, a sensitivity study on modeled pressure would relate most directly to the sensitivity of the model to R/t, providing assurance that the modeled geometry would bound a subject geometry. Results relating to this

were discussed in Section 8.3.4 of EPRI Report 18473. Weld residual stresses (WRS) are discussed in EPRI Report 18473 as well, for example in Section 8.2.2.3.2 of the report. EPRI analogized the WRS likely to be found in the RHR heat exchanger vessel to a thick-walled vessel (likely to be higher than in the relatively thin walled RHR design). Results of a sensitivity study on the WRS are reported in Table 8-18 of EPRI Report 18473, which documents a small effect in the analysis outcome relative to WRS assumptions.

In Section 8.3.4.3.2 of EPRI Report 18473, EPRI performed sensitivity studies on the impact of stress by determining what multipliers to the base case stresses (those determined from the stress analysis) would lead to exceedance of the acceptance criterion for PoF of 1E-06 per year. The result of the study showed that stress multipliers ranging from 1.2 for cases with preservice inspection (PSI) only to 2.5 for several scenarios are necessary to lead to simulation of failure by leakage or rupture. The NRC staff noted that results listed in Table 8-17 of the report improved markedly with ISI inspections, particularly when ISI is conducted at 20, 30, and or 40 years after initial operation. The fact that the licensee has conducted subsequent ISI (Appendix B of the attachment to the submittal) provides ample margin in the PFM analysis results in EPRI Report 18473 relative to the accuracy of the modeled stresses.

Based on the discussion above, the NRC staff finds that the selection of component and materials in EPRI Report 18473 is acceptable. Since the licensee has shown in Appendix A of the attachment to the submittal that Limerick Units 1 and 2 meet the component configuration and material criteria, the NRC staff finds that, in terms of component configuration and materials, the analyses in EPRI Report 18473 are applicable to the requested RHR heat exchanger welds of Limerick Units 1 and 2.

3.3.5.2 *Selection of transients*

The licensee evaluated the applicability of the transients selected in EPRI Report 18473 to the RHR heat exchangers of Limerick Units 1 and 2 (Appendices A and C of the attachment to the submittal). The licensee stated that the cycles in Table 5-4 of the EPRI report meet or exceed the 60-year projected cycles for Limerick Units 1 and 2 that were based on the surveillance test (i.e., thermal transient monitoring of the reactor vessel). The NRC staff confirmed that the RHR heat exchangers of Limerick Units 1 and 2 were well within the modeled transient parameters of EPRI Report 18473. The acceptability of meeting the transient criterion, however, depends on the acceptability of the transient selection described in EPRI Report 18473, which the NRC staff evaluated below.

In Section 5.2 of EPRI Report 18473, EPRI cited thermal cycle diagrams and RHR heat exchanger specifications to identify the appropriate RHR transients to include in the analysis. Because of the expected variations in the operating characteristics of RHR heat exchangers, EPRI increased temperature ramp rates, changes in temperature and pressure, and/or the number of transient cycles to ensure that the selected transients for analysis were bounding. The NRC staff reviewed Section 5.2 of EPRI Report 18473 and the RHR heat exchanger specifications the licensee provided in Enclosures 2 and 3 of the April 1, 2021, supplement and confirmed the types of transients applicable to RHR heat exchangers. The NRC staff then reviewed the comparison of the number of analyzed transients in Table 5-4 of EPRI Report 18473 and the corresponding number of events to date applicable to the RHR heat exchangers at Limerick Units 1 and 2, which the licensee provided in Appendix D of the attachment to the submittal. The NRC staff confirmed from this comparison that the selected transients in Table 5-4 of EPRI Report 18473 bound those of the Limerick Units 1 and 2 RHR heat exchangers.

In Table D-1 of the attachment to the submittal, the licensee listed the fuel pool cooling assist, suppression pool cooling, and steam condensing transients as not applicable to the RHR heat exchangers of Limerick Units 1 and 2. In the supplement dated April 1, 2021, the licensee confirmed that the RHR heat exchangers of Limerick Units 1 and 2 have not been used for suppression pool cooling nor for containment spray, both of which are faulted events. In addition, the plant-specific characteristics of the shutdown cooling transient were milder than those of the shutdown cooling transient analyzed in EPRI Report 18473. The maximum transient temperature, maximum transient pressure, and number of cycles were much lower than those analyzed in the report. The minimum transient pressure was identical. The NRC staff determined that these observations on the plant-specific operational characteristics of the Limerick Units 1 and 2 RHR heat exchangers provide additional assurance that the selected transients in EPRI Report 18473 bound those of the Limerick Units 1 and 2 RHR heat exchangers.

Based on the review of Appendices A and C of the attachment to the submittal, the NRC staff finds that Limerick Units 1 and 2 met the transient criteria in EPRI Report 18473; and based on the discussion above, the NRC staff finds that the transient loads selected in EPRI Report 18473 are acceptable for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2.

3.3.5.3 *Other operating loads*

In Section 5.2 of EPRI Report 18473, EPRI stated that attached piping loads are not considered in the analysis. EPRI stated that the nozzles were substantially thicker than the piping sections and consequently would experience lower bending stresses due to pipe weight. EPRI concluded that because of this difference in thickness, resultant stresses would be overwhelmed by pressure and thermal stresses and consequently would not materially impact the analysis results. EPRI further concluded that this simplification would be addressed through the sensitivity studies in Section 8 of EPRI Report 18473. The NRC staff finds the assumption of not including attached piping loads acceptable for the Limerick Units 1 and 2 plant-specific alternative request because the sensitivity studies provides results that bound the assumption.

In Section 8.2.2.3.2 of EPRI Report 18473, EPRI discussed the cosine distribution assumed for the through-wall residual stress due to welding, or the WRS as discussed earlier. EPRI noted that the cosine distribution used for WRS is the same distribution the NRC staff found acceptable in proprietary report BWRVIP-108. This report is the NRC-approved, PFM-based technical basis for reduction of number of nozzles inspected in boiling water reactor pressure vessels, on which some of the inputs for the current EPRI Report were based. The NRC staff's December 19, 2007 SE of BWRVIP-108 is found in ADAMS Accession No. ML073600374. The licensee further noted that the WRS in the circumferential direction would have a different distribution, but that using the cosine distribution would be bounding due to its higher inner surface stresses. To determine the effect of WRS in the circumferential direction, the licensee conducted a sensitivity study regarding using the cosine distribution compared with a more realistic uniform distribution, as shown in Table 8-18 of EPRI Report 18473. The results of this sensitivity study indicated that the alternate uniform stress distribution for the circumferential direction led to only a marginal change in probabilities of rupture and leakage. Based on the discussion above, NRC staff finds that the WRS in EPRI Report 18473 is adequate for the expected WRS state for the RHR heat exchanger welds of Limerick Units 1 and 2.

The NRC noted that the RHR heat exchanger vessel design analyzed in EPRI Report 18473 is cladded, but there are RHR heat exchanger vessel designs that are not cladded. Therefore, the NRC staff evaluated the effects on loads both with and without cladding. EPRI included the effect of cladding in the analyses in EPRI Report 18473 using Equation 8-1 of the report. The NRC staff determined that using Equation 8-1 of EPRI Report 18473 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 18473 with the finite element analysis (FEA) stresses described in Section 7 of EPRI Report 18473 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 3.3.5.4 of this SE) and second in Equation 8-1 of EPRI Report 18473, 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 RHR heat exchanger welds and nozzles analyzed in EPRI Report 18473. 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 3.3.8 of this SE). This distribution is shown in Equation 8-3 of EPRI Report 18473. 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 RHR heat exchanger vessel in EPRI Report 18473 is 0.0625 inch. Since most of the postulated flaw depth is greater than the clad thickness of 0.0625 inch, the crack tip is in the ferritic base metal that has a lower fracture toughness compared to the stainless steel cladding, and therefore, correctly accounting for the effect of differential thermal expansion between the cladding and base metal could have an impact on the final probability of rupture values. The NRC staff calculated the total applied SIF for a 0.1-inch deep flaw due to thermal stress, pressure stress, and clad residual stress at a temperature of 70 °F (degrees Fahrenheit) when the value of Equation 8-1 of EPRI Report 18473 is maximum. The resulting total applied SIF is 38 ksi√in (kilopounds per square inches root-inch). The NRC staff noted that including the pressure stress is conservative since pressure is typically kept at a minimum at low temperatures. Based on this conservative calculation, 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 impact on the final probability of rupture values since the total applied SIF is less than the 60 ksi√in performed in one of the sensitivity studies on toughness in EPRI Report 18473.

The NRC staff noted that if cladding was not modeled, the through-wall stress due to thermal transients analyzed in EPRI Report 18473 could be tensile instead of compressive (see Section 3.3.5.4 of this SE). However, the NRC staff noted that because the transients of the Limerick Units 1 and 2 RHR heat exchangers are well-bounded by the selected transients in EPRI Report 18473 as discussed in Section 3.3.5.2 of this SE, the effect of this tensile stress would have minimal impact for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2.

Based on the discussion above, the NRC staff finds the treatment of other loads acceptable for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2.

3.3.5.4 *Finite element analyses*

In Section 7 of EPRI Report 18473, EPRI discussed the FEA to determine stresses due to internal pressure and thermal transients for the selected geometries discussed in Section 3.3.5.1 of this SE. 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 and found them acceptable. For instance, the NRC staff noted the average hoop stress due to a 1,000 psi [pounds per square inch] internal pressure at the RHR heat exchanger shell in Figure 7-5 of EPRI Report 18473 is approximately 34,000 psi. The NRC staff calculated a hoop stress of 1000(R/t) = 1000(31.5) = 31,500 psi (R/t = 31.5 for the modeled RHR heat exchanger shell from Table 4-3 of EPRI Report 18473).

The NRC staff noted that the through-wall stress distribution plots for the thermal transients that have temperature drops analyzed in EPRI Report 18473 show compressive stresses at the inner surface (see Figure 7-9 of EPRI Report 18473, for example). Tensile stresses at the inside surface are typically expected for transients that have temperature drops, such as the shutdown cooling transient, but the presence of cladding can cause a compressive stress on the inside surface at hot conditions. 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. 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 effect of clad residual stress in Section 3.3.5.3 of this SE.

Based on the discussion above, the NRC staff determined that the stresses determined through FEA in EPRI Report 18473 are acceptable for referencing for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2.

3.3.6 <u>Fracture Toughness</u>

In Section 8.2.2.6 of EPRI Report 18473, EPRI assumed a K_{IC} value of 106 ksi√in based on the upper shelf fracture toughness value derived from C-8320 of the ASME Code, Section XI, Appendix C. For the probabilistic portion of the analysis, the K_{IC} value of 106 ksi√in was taken as a mean, with a standard deviation of 5 ksi√in. The NRC staff confirmed that this standard deviation is consistent with the value the NRC staff has accepted in the BWRVIP-108 project. The staff confirmed the material properties cited from C-8320 of the ASME Code, Section XI. More importantly the sensitivity study on fracture toughness samples K_{IC} values down to a mean value of 60 ksi√in and consequently provides additional assurance that the mean value of 106 ksi√in for K_{IC} is adequate.

Based on the discussion above and the discussion in Sections 3.3.5.1 this SE, which confirmed that the materials are acceptable for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2, the NRC staff finds the fracture toughness model in EPRI Report 18473 acceptable for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2.

3.3.7 FCG Rate

In Sections 8.3.2.6 of EPRI Report 18473, 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 noted that the FCG rate in ASME Code, Section XI, A-4300 is applicable to both BWR and pressurized-water reactor (PWR).

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 Limerick Units 1 and 2 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 18473, EPRI stated that assuming the A-4300 curve as the median curve 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. The NRC staff determined that because of the amount of available data for ferritic FCG rate, however, the difference between the 50 percent confidence limit on the median and the 95 percent confidence limit on the median would likely be small. Thus, the NRC staff determined for the Limerick Units 1 and 2 plant-specific alternative request that assuming the A-4300 curve 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 xLPR, a PFM software sponsored by the NRC and EPRI. The NRC staff confirmed that the FCG rate in xLPR received adequate V&V. The NRC staff noted that the FCG rate is the rate defined in A-4300 with a statistical distribution around it since the FCG rate is treated as a random parameter. In Section 8.3.4.3.4 of EPRI Report 18473, EPRI performed a sensitivity study on the effect of FCG rate on probability of leakage by replacing the A-4300 FCG rate with the FCG rate used in BWRVIP-108. The result of the study showed that the A-4300 FCG rate led to a higher 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 RHR heat exchanger welds of Limerick Units 1 and 2.

3.3.8 Initial Flaw Depth and Length Distribution

In Section 8.3.2.2 of EPRI Report 18473, EPRI stated that the flaw distribution derived from flaw data from the PVRUF vessel was applied to the RHR heat exchanger vessel in the analyses. NUREG-6471 “Characterization of Flaws in U.S. Reactor Pressure Vessels,” (ADAMS Package Accession No. ML112510316) states that the PVRUF vessel is from an unused PWR vessel and that the PVRUF data are from fabrication flaws in the PVRUF vessel weldment. The NRC staff noted that the PVRUF depth distribution represented by Equation 8-3 of EPRI Report 18473 consisted of mostly small flaws that are inner surface-breaking.

The NRC staff also noted that the nominal thickness of PWR vessels is 8 inches, and that since the PVRUF flaw data are based on a PWR vessel, the data may not be appropriate for much thinner components, such as the RHR heat exchanger vessels of Limerick Units 1 and 2. Also, differences in manufacturing process and examination techniques between PWR vessels and RHR heat exchanger vessels can result in different flaw distributions. The NRC staff needed additional information to determine if the PVRUF flaw data is appropriate for the Limerick Units 1 and 2 RHR heat exchanger vessels.

In the supplement dated April 1, 2021, the licensee assessed the PVRUF flaw data for applicability to the RHR heat exchanger welds and nozzles of Limerick Units 1 and 2. In this assessment, the licensee considered four other initial flaw distributions in addition to the PVRUF and Chapman distributions discussed in Section 8.3.2.2 of EPRI Report 18473. The licensee stated that collectively, the six initial flaw distributions represent distributions that were developed for vessel and piping that consider the relevant geometrical parameter (i.e., thickness), and different materials, manufacturing processes, and examinations for such components. The licensee performed PFM analyses using all six flaw distributions for the limiting base cases, BHX-VC-P1A and BHX-TS-P3A, in EPRI Report 18473 to demonstrate that the probability of rupture values were insensitive to initial flaw depth, and that both probability of rupture and leakage values were below the criterion of 1E-06. The licensee indicated that two of the six flaw distributions are conservative relative to the PVRUF distribution. The NRC staff reviewed the licensee's assessment of the applicability of the PVRUF flaw distribution to the Limerick Units 1 and 2 RHR heat exchanger welds, and finds it acceptable because (1) the licensee analyzed flaw distributions that address the effects due to differences in geometry, materials, manufacturing processes, and examinations; (2) the licensee included initial flaw depth distributions that are conservative relative to the PVRUF distribution; and (3) the probability of rupture values were insensitive to initial flaw depth, and both probability of rupture and leakage values were below the criterion of 1E-06. Thus, the NRC staff determined for the Limerick Units 1 and 2 plant-specific alternative request that applying the PVRUF initial flaw depth distribution to the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2 is acceptable.

In Section 8.3.2.2 of EPRI Report 18473, 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" (ADAMS Accession No. ML040830499) for the log-normal distribution for the flaw length. As the NRC staff observed in the audit summary report for PROMISE, 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 the Limerick Units 1 and 2 plant-specific alternative request and the length distribution is acceptable for the analysis results referenced for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2.

3.3.9 ISI Schedule and Examination Coverage

In Section 5.0 of the attachment to the submittal, the licensee proposes to increase the ISI interval for the requested RHR heat exchanger welds and nozzles to the end of the current operating licenses (October 26, 2044, for Limerick Unit 1 and June 22, 2049, for Limerick Unit 2) from the current 10-year ISI interval required by the ASME Code, Section XI. The licensee stated that this proposed alternative ISI interval equates to an extension of 27 years, 8 months, 25 days for Limerick Unit 1 and 32 years, 4 months, 21 days for Limerick Unit 2 from the end of the third ISI Interval (January 31, 2017) at which time all ASME Code, Section XI, Division 1 requirements were satisfied.

The licensee further explained that both RHR heat exchangers of Limerick Unit 1 were replaced in 1994 and the “B” RHR heat exchanger of Limerick Unit 2 was replaced in 2011. The only RHR heat exchanger from original plant construction is the “A” RHR heat exchanger of Limerick Unit 2, and the licensee stated that it received initial examination (i.e., PSI) plus examinations for three 10-year ISI intervals, denoted by PSI + 10 + 20 + 30. With the proposed ISI interval extensions, this would be PSI + 10 + 20 + 30 + 57 for Limerick Unit 1 and PSI + 10 + 20 + 30 + 62 for Limerick Unit 2. The NRC staff noted that the proposed ISI extensions of PSI + 10 + 20 + 30 + 57 and PSI + 10 + 20 + 30 + 62 are essentially equivalent to PSI + 10 + 20 + 30 + 60 because the difference is plausibly too small to have any impact on the final PFM results. The NRC staff also noted that ISI with replacement RHR heat exchangers would be at least as good PSI + 10 + 20 + 30 + 60 because, given the implementation of ISI in PROMISE as explained in the next paragraph, replacement is essentially “repair” of a postulated flaw, while the outcomes of ISI when it is implemented is either repair of a postulated flaw or non-detection and growth of a postulated flaw that could lead to failure, as described below.

The NRC staff noted the impact of ISI schedule on the PoF values, as shown in Table 8-10 of EPRI Report 18473. In the audit summary report for PROMISE (see Section 3.3.3.1 of this SE), the NRC staff observed 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 probability of detection (POD) curve (see Section 3.3.10.2 of this SE for further discussion of the POD curve). 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 3.3.3.1 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 different 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 3.3.3.1 of this SE). Thus, because of the adequate implementation and benchmarking of ISI, the NRC finds that the PoF values in EPRI Report 18473 adequately included the effect of ISI schedule. Since the licensee referenced the PFM results in EPRI Report 18473 for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2, the NRC staff finds that the licensee adequately included the effect of ISI schedule on the PoF values for the RHR heat exchanger welds and nozzles of Limerick Units 1 and 2.

In Section 8.3.5 of EPRI Report 18473, EPRI stated that it assumed 100 percent inspection of the required volume (i.e., 100 percent examination coverage) of the RHR heat exchanger welds and nozzles analyzed in the report during each of the ISI scenarios that were evaluated. EPRI explained that by performing examinations with 100 percent coverage during PSI, no other examinations are needed for safe plant operation for 80 years. EPRI further stated 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 3.3.11 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, the NRC staff observed 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 an ISI is implemented.

In Appendix B of the attachment to the submittal, the licensee included the examination history of the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2, which shows that examination coverage could be as low as 53 percent. Less than 100 percent examination coverage is discussed in Section 3.3.11 of this SE.

Therefore, because examination coverage was adequately implemented and less than 100 percent examination coverage was considered as discussed above, the NRC finds that the licensee adequately addressed the effect of examination coverage on the PoF values for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2.

3.3.10 Other Considerations

3.3.10.1 *Flaw density*

In Section 8.3.2.2 of EPRI Report 18473, EPRI stated that 1.0 fabrication flaw per weld and 0.001 flaws were assumed for the nozzle inside radius (NIR) section of each nozzle, consistent with the assumptions in BWRVIP-108. As discussed in the December 19, 2007 SE of BWRVIP-108, the 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.

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^3 per weld and the volumes of the subject RHR heat exchanger welds of Limerick Units 1 and 2 estimated from the RHR heat exchanger dimensions in Table A-1 of the attachment to the submittal, the NRC staff determined that 1.0 flaw per weld is adequate for the requested RHR heat exchanger welds of Limerick Units 1 and 2.

In the supplement dated April 1, 2021, the licensee performed sensitivity studies with 0.1 flaw per nozzle in the NIR locations, designated as cases BHX-NR-P8N and BHX-NR-P9N, analyzed in EPRI Report 18473. The resulting probability of leakage and rupture values are below the criterion of 1E-06 per year, as shown in Table RAI-231-1 of the supplement. Based on this discussion, the NRC staff determined that 0.1 flaw per NIR is adequate for the requested RHR heat exchanger nozzles of Limerick Units 1 and 2.

3.3.10.2 *Probability of detection*

In Section 8.3.2.3 of EPRI Report 18473, 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 8-12 in EPRI Report 18473 is the same as the POD curve in BWRVIP-108. The NRC staff noted that the welds and nozzles 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 NRC staff also noted that per I-2200 of ASME Code, Section XI, the required examinations for RHR heat exchangers vessel welds and nozzles are those specified in ASME Code, Section XI, Appendix III, and as such observed that the Appendix VIII-based POD curve used in the EPRI analyses may not be adequate. In the supplement dated April 1, 2021, the licensee provided a basis for the adequacy of the Appendix VIII-based POD curve as it relates to the RHR heat exchanger components and their inspection modelling. The licensee performed sensitivity studies using the POD curve based on Appendix L of ASME Code, Section XI and compared them with the PFM results using the Appendix VIII-based POD curve. The licensee stated that the Appendix L-based POD curve is applicable to ferritic piping per Supplement 3 of Appendix VIII of the ASME Code, Section XI and that it fits the inspection requirements for heat exchangers because the component thickness and the inspection and fabrication practices for heat exchangers are similar to those used for ferritic piping. The licensee applied the licensee's proposed ISI schedule of PSI + 10 + 20 + 30 + 60 and showed that for the limiting cases in EPRI Report 18473, the probability of rupture values using the Appendix L-based POD curve are the same as those using the Appendix VIII-based POD curve. The licensee also showed that for the limiting cases in EPRI Report 18473, the probability of leakage values using the Appendix L-based POD curve increased relative to those using the Appendix VIII-based POD curve.

The NRC staff verified that Supplement 3 of Appendix VIII of the ASME Code, Section XI applies to ferritic piping welds. The NRC staff determined that, with respect to POD curves, it is reasonable to treat the RHR heat exchanger welds and nozzles of Limerick Units 1 and 2 as piping components because the RHR heat exchanger vessels are in the same range of thicknesses as piping components. The NRC staff determined that the licensee's comparison of probability of rupture values between the Appendix VIII-based and Appendix L-based POD curves acceptable because there was no change in probability of rupture values. The NRC staff determined that, even though the probability of leakage values using the Appendix L-based POD curves increased, and in some cases exceeded the criterion of 1E-06 per year, the licensee's comparison of probability of leakage values between the Appendix VIII-based and Appendix L-based POD curves is acceptable because leakage is not component rupture and would be managed by the plant leakage detection system.

Based on the discussion above, the NRC staff determined for the Limerick Units 1 and 2 alternative request that the Appendix VIII-based POD curve is adequate for use in the PFM analyses referenced for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2.

3.3.10.3 *Models*

In Section 8.2.2.5 of EPRI Report 18473, 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 that are similar to the crack models used to analyze postulated flaws in the nozzle-to-shell welds in BWRVIP-108 and BWRVIP-241, for which the NRC staff has approved and issued SEs. Since the RHR heat exchanger nozzle-to-shell welds (Item No C2.21) of Limerick Units 1 and 2 are similar in configuration to the nozzle-to-shell welds analyzed in BWRVIP-108 and BWRVIP-241, the NRC staff finds the Limerick Units 1 and 2 plant-specific alternative request and the cylindrical SIF models in EPRI Report 18473 are appropriate for the postulated flaws in the requested RHR heat exchanger nozzle-to-shell welds

of Limerick Units 1 and 2.

The NRC staff finds the Limerick Units 1 and 2 plant-specific alternative request and the cylindrical SIF models in EPRI Report 18473 appropriate for the postulated flaws in the requested RHR heat exchanger shell circumferential welds (Item Nos. C1.10 and C1.20) of Limerick Units 1 and 2 because these welds are located in the cylindrical portions of the Limerick Units 1 and 2 RHR heat exchanger shell.

In Section 8.2.2.5 of EPRI Report 18473, 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 3.3.3.1 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 Limerick Units 1 and 2 plant-specific alternative request that the NIR crack model in EPRI Report 18473 is appropriate for the postulated flaws in the requested RHR heat exchanger nozzles (Item No. C2.22) of Limerick Units 1 and 2.

3.3.10.4 *Uncertainty*

In Section 8.3.1.2 of EPRI Report 18473, 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 loop and an epistemic loop. In Section 8.3.4.1 of EPRI Report 18473, 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 in PWR water environments). EPRI performed 10 million aleatory realizations and 1 epistemic realization for the PFM analyses. The NRC staff noted that representing all variables as aleatory will result in probabilities that represent the mean of the distribution.

In the audit summary report for PROMISE, 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 in 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 base cases in EPRI Report 18473 had a probability of rupture values more than two orders of magnitude below the acceptance criterion of 1E-06 per year, the NRC staff finds for the Limerick Units 1 and 2 alternative request that the large uncertainty in the low probability results reflected by the large percent error is acceptable.

In Table 8-8 of EPRI Report 18473, EPRI indicated that there were no uncertainties in the transient stresses. The NRC staff finds the Limerick Units 1 and 2 alternative request and treating transient stresses as constant rather than random is acceptable since the transients were selected based on bounding temperature rates and changes, as discussed in Section 3.3.5.2 of this SE.

Based on the discussion above, the NRC staff finds for the Limerick Units 1 and 2 alternative request that the licensee's handling of uncertainty is acceptable for the analysis results referenced for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2.

3.3.10.5 *Convergence*

The NRC staff noted that EPRI did not conduct sensitivity studies on the number of realizations in EPRI Report 18473. Conducting sensitivity studies on number of realizations would determine whether the PFM results in the report were sufficiently converged. The NRC staff also noted the counterintuitive results in Table 8-10 of EPRI Report 18473, in which increasing the number of ISI appeared to increase the probability of leakage values, as shown, for example, in the results for scenarios 3 and 4 of the table. The NRC staff noted that the counterintuitive results could indicate insufficient convergence.

The supplement dated April 1, 2021, showed probability of rupture and leakage values for case BHX-NS-P6A analyzed in EPRI Report 18473 for 10^7 and 10^8 realizations. The results indicate little difference in the probability of rupture and leakage values between 10^7 and 10^8 realizations. To explain the counterintuitive results in Table 8-10 of EPRI Report 18473, the licensee also performed analyses for scenarios 3 and 4 in the table with different random number seeds combined with 10^7 and 10^8 realizations. The results of these analyses showed the expected decrease in probability of leakage values going from scenario 3 to 4 and indicate little difference in the probability of rupture and leakage values between 10^7 and 10^8 realizations.

Based on the discussion above, the NRC staff finds for the Limerick Units 1 and 2 plant-specific alternative request that the PFM results in EPRI Report 18473 were sufficiently converged, and that the number of realizations used in the analyses, 10^7 realizations, is acceptable for the PFM results referenced for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2. 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 3.3.10.4 of this SE.

3.3.10.6 *DFM analysis*

In Section 8.2 of EPRI Report 18473, EPRI performed a DFM analysis with an initial flaw depth of approximately 5.2 percent of the component thickness, and average values of all other parameters considered random in the PFM analysis. In the supplement dated April 1, 2021, the licensee reran the DFM analysis with the corrected initial flaw depth of 14.2 percent (rounded to 15 percent) of the component thickness. All locations analyzed with the 15 percent initial flaw depth in the subject RHR heat exchanger welds and nozzles resulted in many years to reach 80 percent of the component thickness, the least being 465 years, as shown in Table RAI-229-1 of the supplement. No locations reached an applied SIF of greater than the mean fracture toughness of 106 ksi√in used in the PFM analysis divided by a structural factor of 2, (106 ksi√in) ÷ 2 = 53 ksi√in. Based on this discussion, the NRC staff determined for the Limerick Units 1 and 2 plant-specific alternative request that overall, the DFM analysis supports the PFM analysis.

3.3.11 PFM Results Relevant to Alternative Request I4R-24

In Section 5.0 of the attachment 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 8.3.5 of EPRI Report 18473, which states that no other inspections are required for the welds evaluated in the report to maintain safe plant operation for 80 years. The NRC staff does not find the licensee's and EPRI's general conclusion on PSI-only examinations 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.

The NRC staff determined that since the PFM analyses in EPRI Report 18473 were based on representative RHR heat exchanger vessels, the uncertainties on the different parameters (which are different from the sampling uncertainty discussed in Section 3.3.10.4 of this SE) should be taken into account, especially those from the significant parameters of stress and fracture toughness 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 + 10 + 20 + 50 [2], the probability of rupture at 80 years for the limiting location changed from 1.25E-09 per year to 1.43E-06 per year (Table 8-21 of EPRI Report 18473). While the NRC staff acknowledged that this study assumed conservative values for stress and fracture toughness simultaneously (thereby accounting for uncertainties in these two parameters), 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 of the two parameters could easily lead to probability of ruptures greater than 1E-06 per year.

Given the discussion on uncertainty above and in Section 3.3.10.4 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 + 60, would have a much more adverse effect on risk-informed principles, particularly since PSI-only examinations would remove any future condition monitoring needed for risk-informed decision making.

As discussed in Section 3.3.9 of this SE, the licensee is seeking the alternative ISI schedule of PSI + 10 + 20 + 30 + 60. Therefore, the NRC staff determined that PFM results for PSI + 10 + 20 + 30 + 60 are the results relevant to the licensee's proposed alternative.

The NRC staff noted that even though EPRI Report 18473 does not have PoF results for PSI + 10 + 20 + 30 + 60, it has results for PSI + 10 + 20 + 50, which bounds the former since ISI is implemented more times in the former. Therefore, the NRC staff evaluated the PFM results in the sensitivity studies in Section 8.3.4.3 of EPRI Report 18473 relevant to the proposed alternative ISI schedule of PSI + 10 + 20 + 30 + 60 by assessing the results for PSI + 10 + 20 + 50.

As stated earlier, Table 8-21 of EPRI Report 18473 shows that the limiting probability of rupture is 1.43E-06 per year, which exceeds the criterion of 1E-06 per year. This probability of rupture value is for an ISI schedule of PSI + 10 + 20 + 50, which bounds the licensee's proposed alternative of PSI + 10 + 20 + 30 + 60, for the combined sensitivity studies on fracture toughness and stress. The NRC staff noted that that if fracture toughness was set to the base case value of 106 ksi√in with a standard deviation of 5 ksi√in, which the NRC staff found acceptable in Section 3.3.6 of this SE, the only other parameter that needs to be addressed is stress. Table 8-17 of EPRI Report 18473 shows stress multipliers for various ISI scenarios that are needed for the probability of rupture or leakage value to exceed the criterion of 1E-06 per year. For the scenario of PSI + 10 + 20 + 50 (Scenario 11), it takes a stress multiplier of 1.7 for the probability of rupture value at the limiting location, BHX-NS-P6A, to exceed the criterion of 1E-06 per year. As shown in Figure 7-19 of EPRI Report 18473, this limiting location is in the

[2] The supplement dated April 1, 2021 stated that the ISI schedule shown in Table 8-21 of EPRI report 18473 should be PSI + 10 + 20 + 50.

nozzle forging for which the maximum R/t ratio was modeled in the analysis, as indicated in Tables 4-1 and 4-4 of EPRI Report 18473. Since the R/t ratio reflects the stress level and the limiting location was modeled with the maximum R/t ratio, the stress multiplier at the limiting location is 1.0, which would result in a probability of rupture less than the criterion of 1E-06 per year.

The PFM results in EPRI Report 18473 discussed thus far assume 100 percent examination coverage. To address less than 100 percent examination coverage, the NRC staff noted that none of results of the sensitivity studies on fracture toughness (Tables 8-13 and 8-14 of EPRI Report 18473) or sensitivity studies on stress (Tables 8-15 and 8-16 of EPRI Report 18473) exceeded the criterion of 1E-06 per year. The PFM results in these tables are with PSI examinations only, and as such, the NRC staff determined that the results in Tables 8-13 through 8-16 of EPRI Report 18473 adequately address less than 100 percent examination coverage for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2.

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. The PFM results in EPRI Report 18473 discussed above 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 3.3.3 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 this factor of 1.3 has no impact on the NRC staff's discussion of the PFM results in EPRI Report 18473 in the preceding discussion.

Therefore, the NRC staff determined that the PFM analyses in EPRI Report 18473 adequately address uncertainties in the PoF values relevant to the licensee's proposed alternative of PSI + 10 + 20 + 30 + 60 for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2. Based on the discussion above, the NRC staff finds that the proposed alternative of PSI + 10 + 20 + 30 + 60 for the requested RHR heat exchanger welds and nozzles of Limerick Units 1 and 2 would result in a PoF per year that is below the acceptance criterion of 1E-06 per year.

4.0 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 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 I4R-24 at Limerick Generating Station, Units 1 and 2 for the remainder of the fourth 10-year ISI interval and up to the end of the 60-year operating license of each unit (October 26, 2044, for Unit 1) and (June 22, 2049, for Unit 2).

The NRC's authorization of the proposed alternative does not infer or imply the approval of EPRI Report 18473 for generic use.

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 Contributor: David Dijamco

Date: September 2, 2021

SUBJECT: LIMERICK GENERATING STATION, UNIT 1 AND 2 – ISSUANCE OF RELIEF REQUEST I4R-24 ASSOCIATED WITH RESIDUAL HEAT REMOVAL HEAT EXCHANGER CATEGORY C-A AND C-B EXAMINATIONS FOR THE REMAINDER OF THE FOURTH 10-YEAR ISI INTERVAL AND UP TO THE END OF THE 60-YEAR OPERATING LICENSES (EPID L-2020-LLR-0122) DATED SEPTEMBER 2, 2021

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