Licensing Technical Report

# Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel

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Licensing Technical Report

## Department of Energy Acknowledgement and Disclaimer

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## Abstract

This report describes the acceptability of SA-965 Grade FXM-19 austenitic stainless steel base metal and E/ER209 or E/ER240 weld filler metal for use in the NuScale Power Module (NPM) lower reactor pressure vessel (RPV).

The RPVs in operating pressurized water reactors (PWRs) in the United States are made of ferritic materials. The Nuclear Regulatory Commission (NRC) regulations in Title 10 of the Code of Federal Regulations (CFR) Section 50.60 (10 CFR 50.60); 10 CFR 50.61; 10 CFR 50, Appendix G; and 10 CFR 50, Appendix H, either refer specifically to or utilize data for ferritic materials only. These regulations support compliance with General Design Criterion (GDC) 14, GDC 15, GDC 31, and GDC 32.

Since there are no regulatory data or guidance available for austenitic stainless steel RPVs, the NPM RPV beltline cannot be evaluated using the current regulations.

This report summarizes the known data relating to FXM-19 base metal and E/ER209 or E/ER240 weld filler metal. The results of the literature review support exemptions from 10 CFR 50.60 and 10 CFR 50.61. This report also summarizes NuScale’s position on the reactor vessel surveillance program (RVSP) requirements in 10 CFR 50, Appendix H, and GDC 32.

## Executive Summary

This report describes the acceptability of SA-965 Grade FXM-19 austenitic stainless steel base metal and E/ER209 or E/ER240 weld filler metal for use in the NuScale Power Module (NPM) lower reactor pressure vessel (RPV).

The RPVs in operating pressurized water reactors (PWRs) in the United States are made of ferritic materials. The Nuclear Regulatory Commission (NRC) regulations in Title 10 of the Code of Federal Regulations (CFR) Section 50.60 (10 CFR 50.60); 10 CFR 50.61; 10 CFR 50, Appendix G; and 10 CFR 50, Appendix H, either refer specifically to or utilize data for ferritic materials only. These regulations support compliance with General Design Criterion (GDC) 14, GDC 15, GDC 31, and GDC 32.

NuScale evaluated data on austenitic stainless steel, including SA-965 Grade FXM-19 base metal, E/ER209 or E/ER240 weld filler metal, and Type 3XX austenitic stainless steel for comparison purposes. The data show that austenitic stainless steels have superior ductility and are less susceptible to the effects of neutron and thermal embrittlement than ferritic materials. In addition, the data and methodology in 10 CFR 50.61; 10 CFR 50, Appendix G; and 10 CFR 50, Appendix H, are not applicable to non-ferritic materials.

In addition to assessing material properties and the safety of austenitic stainless steel in the lower RPV, NuScale assessed the beltline of the RPV. Since the lower RPV is austenitic stainless steel and thus less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, it should not be considered the RPV beltline. The upper RPV, which is made of ferritic steel, is not within the RPV beltline; the 57 effective full-power year (EFPY) peak design fluence for the upper RPV is less than the 10 CFR 50, Appendix H, reactor vessel surveillance program (RVSP) threshold of 1E+17 n/cm$^2$, E > 1 MeV, and thus is not subject to supplementary fracture toughness and reactor vessel surveillance program (RVSP) requirements to address the effects of neutron embrittlement.

Though the NPM design does not use the methodology in 10 CFR 50.61; 10 CFR 50, Appendix G; and 10 CFR 50, Appendix H, it does meet the requirements of GDC 14, GDC 15, GDC 31, and GDC 32. The requirements of GDC 14, GDC 15, and GDC 31 are met by ensuring that the NPM lower RPV is constructed of a material that has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement compared to ferritic materials, which increases the integrity and safety of the RCPB. Because the austenitic stainless steel used in the lower RPV is less susceptible to the effects of neutron and thermal embrittlement compared to ferritic materials, and because the ferritic materials in the RPV are below the 10 CFR 50, Appendix H, RVSP threshold, the RPV does not need an RVSP to ensure adequate fracture toughness; therefore, the portion of GDC 32 requiring an “appropriate” material surveillance program is satisfied without an RVSP.

The US600 design used austenitic stainless steel for the lower containment vessel (CNV) because its material properties are less susceptible to the effects of neutron and thermal embrittlement than ferritic materials.

In Section 6.1.1.4.2 of the US600 design final safety evaluation report, the NRC stated:

> *The staff finds the use of SA-965, Grade FXM-19, and its associated weld filler metals acceptable for use in the lower portion of the CNV, as the calculated fluence to the CNV is lower than what is expected to cause embrittlement, and the selection of SA-965, Grade FXM-19, an austenitic stainless steel, is resistant to radiation embrittlement.*

In Section 6.2.7.4 of the final safety evaluation report, the NRC stated:

> *Within the ASME Code, detailed fracture toughness requirements are placed on ferritic materials, as nonferritic materials exhibit sufficient inherent fracture toughness that additional requirements are deemed unnecessary. For example, the austenitic stainless steel used for the CNV lower shell, SA-965, FXM-19, was explicitly chosen specifically for its superior fracture toughness and resistance to neutron embrittlement.*

The results of this report confirm that austenitic stainless steels and compatible weld filler metals are likewise acceptable for use in the lower RPV without additional fracture toughness requirements because they have superior ductility and are less susceptible to the effects neutron and thermal embrittlement than ferritic materials.

# 1.0 Introduction

## 1.1 Purpose

The NuScale Power Module (NPM) lower reactor pressure vessel (RPV) is made of SA-965 Grade FXM-19 austenitic stainless steel base metal and uses E/ER209 or E/ER240 weld filler metal. The use of austenitic stainless steel benefits overall plant safety because it has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials.

Nuclear Regulatory Commission (NRC) regulations in Title 10 of the Code of Federal Regulations (CFR) Section 50.60 (10 CFR 50.60); 10 CFR 50.61; 10 CFR 50, Appendix G; and 10 CFR 50, Appendix H, while applicable to all materials, were developed for and only include data and methods for ferritic materials. Four known RPVs used austenitic stainless steel RPVs, and reactor vessel internals (RVIs) in operating light water reactors are made of austenitic stainless steel. NuScale assessed available literature for FXM-19 base metal and bounding materials for the weld filler metals to evaluate the safety of using austenitic stainless steel in the lower RPV. NuScale identified four RPVs and RVIs made of austenitic stainless steel. The available literature demonstrates the acceptability of austentic stainless steel for the lower RPV of the NPM, without additional fracture toughness or material surveillance requirements. Therefore, the US460 standard design supports exemptions from 10 CFR 50.60 and 10 CFR 50.61.

The US460 standard design complies with General Design Criterion (GDC) 14, GDC 15, GDC 31, and GDC 32.

## 1.2 Scope

This report applies to the US460 standard design with an austenitic stainless steel lower RPV and supports the Standard Design Approval Application (SDAA).

## 1.3 Abbreviations

**Table 1-1 Abbreviations**

| Term | Definition |
|---|---|
| ASME | American Society of Mechanical Engineers |
| ATR | Advanced Test Reactor |
| BPVC | Boiler Pressure Vessel Code |
| CASS | cast austenitic stainless steel |
| CFR | Code of Federal Regulations |
| dpa | displacements per atom |
| EFPY | effective full-power years |
| EPRI | Electric Power Research Institute |
| FN | ferrite number |
| GDC | General Design Criterion |
| INL | Idaho National Laboratory |
| LWR | light water reactor |
| NPM | NuScale Power Module |
| NRC | Nuclear Regulatory Commission |
| PTS | pressurized thermal shock |
| PWR | pressurized water reactor |
| RCPB | reactor coolant pressure boundary |
| RPV | reactor pressure vessel |
| $RT_{NDT}$ | nil-ductility reference temperature |
| $RT_{NDT(u)}$ | unirradiated nil-ductility reference temperature |
| $RT_{PTS}$ | reference temperature for pressurized thermal shock |
| RVI | reactor vessel internals |
| RVSP | reactor vessel surveillance program |

# 2.0 Background

The US460 standard design uses austenitic stainless steel and compatible weld filler metals in the lower RPV because austenitic stainless steels have superior ductility and lower susceptibility to the effects of neutron and thermal embrittlement compared to ferritic materials. This increases the integrity and safety of the reactor coolant pressure boundary (RCPB). By NRC regulations, the beltline is the region of the RPV that directly surrounds the effective height of the active core and is predicted to experience sufficient neutron radiation such that it is the most limiting material with regard to radiation damage. For the US460 standard design, the lower RPV contains the beltline; however, the lower RPV is made of austenitic stainless steel, which can withstand the most severe radiation damage in the RPV better than ferritic materials. Therefore, use of austenitic stainless steel and compatible weld filler metals increases the overall safety of the RPV.

NuScale conducted a literature review of data related to SA-965 Grade FXM-19 base metal and to E/ER209 or E/ER240 weld filler metals, as well as a comparison to 3XX austenitic stainless steel properties.

## 2.1 Regulatory Requirements

### 2.1.1 General Design Criterion 14 - Reactor Coolant Pressure Boundary

General Design Criterion 14 (Reference 5.1.5) requires:

> *The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.*

### 2.1.2 General Design Criterion 15 - Reactor Coolant System Design

General Design Criterion 15 (Reference 5.1.6) requires:

> *The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.*

### 2.1.3 General Design Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

General Design Criterion 31 (Reference 5.1.7) requires:

> *The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.*

### 2.1.4 General Design Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

General Design Criterion 32 (Reference 5.1.8) requires:

> *Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.*

### 2.1.5 10 CFR 50.60 - Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation

10 CFR 50.60 (Reference 5.1.1) requires that licensed light water reactors (LWRs) meet the fracture toughness and material surveillance program requirements for ferritic materials in the RCPB set forth in 10 CFR 50, Appendix G (Reference 5.1.3), and in 10 CFR 50, Appendix H (Reference 5.1.4). Proposed alternatives to the requirements described in Reference 5.1.3 or Reference 5.1.4 or portions thereof may be used when the NRC grants an exemption under 10 CFR 50.12.

### 2.1.6 10 CFR 50, Appendix G - Fracture Toughness Requirements

10 CFR 50, Appendix G (Reference 5.1.3), specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the RCPB of LWRs to provide adequate margins of safety during any condition of normal operation.

Conditions of normal operation include anticipated operational occurrences and system hydrostatic tests to which the pressure boundary may be subjected over its service lifetime. 10 CFR 50, Appendix G, requires that beltline materials be tested in accordance with 10 CFR, Appendix H, the results of which are utilized in establishing the fracture toughness requirements for those materials.

### 2.1.7 10 CFR 50, Appendix H - Reactor Vessel Material Surveillance Program Requirements

10 CFR 50, Appendix H (Reference 5.1.4), requires that licensees establish and maintain a material surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors. The materials in the reactor vessel beltline region undergo exposure to neutron irradiation and to the thermal environment.

### 2.1.8 10 CFR 50.61 - Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events

10 CFR 50.61 (Reference 5.1.2) requires the pressurized thermal shock (PTS) event screening for the RPV beltline region of pressurized water reactors (PWRs). Pressurized thermal shock events are events or transients in PWRs causing severe overcooling (thermal shock) concurrent with or followed by significant pressure within the RPV.

## 3.0 Evaluation of Austenitic Stainless Steel Properties

### 3.1 NPM Lower Reactor Pressure Vessel and Beltline

The NPM lower RPV is made of SA-965 Grade FXM-19 austenitic stainless steel and contains two SA-965 Grade FXM-19 forgings joined by one circumferential weld. The permitted weld filler metal for the lower RPV circumferential weld is SFA 5.4, E209 or SA5.4, E240 and SA 5.9 E209 or SA 5.9 E240 (E/ER209 or E/ER240). Figure 3-1 shows the configuration of the NPM lower RPV.

**Figure 3-1 NPM Lower Reactor Pressure Vessel Pressure-Retaining Materials**

{{

}}2(a),(c),ECI

### 3.2 NPM Lower Reactor Pressure Vessel Austenitic Stainless Steel Characteristics

SA-965 Grade FXM-19 is a nitrogen-strengthened austenitic stainless steel with a nominal composition of 22Cr-13Ni-5Mn. The unified numbering system designation is S20910. This material is known as XM-19 or Nitronic 50 in literature.

Table 3-1 compares the chemical composition of SA-965 Grade FXM-19 base metal and E/ER209 or E/ER240 weld filler metal for the lower RPV with Type 3XX austenitic stainless steels. Table 3-2 shows the tensile requirements and that SA-965 Grade FXM-19 and E/ER209 or E/ER240 are stronger than Type 3XX austenitic stainless steels due to elevated levels of manganese and nitrogen.

## Table 3-1 Austenitic Stainless Steel Chemical Compositions

| Material | Chemical Composition Requirement (weight percent)(1) | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | **C** | **Mn** | **P** | **S** | **Si** | **Ni** | **Cr** | **Mo** | **NB+Ta** | **N** | **V** | **Cu** |
| FXM-19 | 0.06(2) | 4.0-6.0 | 0.045 | 0.03 | 1.0 | 11.5-13.5 | 20.5-23.5 | 1.5-3.0 | 0.10-0.30 | 0.20-0.40 | 0.10-0.30 | -- |
| SA-965, F304 | 0.08 | 2.0 | 0.045 | 0.03 | 1.0 | 8.0-11.0 | 18.0-20.0 | -- | -- | -- | -- | -- |
| SA-965, F316 | 0.08 | 2.0 | 0.045 | 0.03 | 1.0 | 10.0-14.0 | 16.0-18.0 | 2.0-3.0 | -- | -- | -- | -- |
| SFA 5.4, E209 | 0.06(2) | 4.0-7.0 | 0.04 | 0.03 | 1.0 | 9.5-12.0 | 20.5-24.0 | 1.5-3.0 | -- | 0.10-0.30 | 0.10-0.30 | 0.75 |
| SFA 5.9, ER209 | 0.05(2) | 4.0-7.0 | 0.03 | 0.03 | 0.9 | 9.5-12.0 | 20.5-24.0 | 1.5-3.0 | -- | 0.10-0.30 | 0.10-0.30 | 0.75 |
| SFA 5.4, E240 | 0.06(2) | 10.5-13.5 | 0.04 | 0.03 | 1.0 | 4.0-6.0 | 17.0-19.0 | 0.75 | -- | 0.10-0.30 | -- | 0.75 |
| SFA 5.9, ER240 | 0.05(2) | 10.5-13.5 | 0.03 | 0.03 | 1.0 | 4.0-6.0 | 17.0-19.0 | 0.75 | -- | 0.10-0.30 | -- | 0.75 |
| SFA 5.4, E308 | 0.08 | 0.5-2.5 | 0.04 | 0.03 | 1.00 | 9.0-11.0 | 18.0-21.0 | 0.75 | -- | -- | -- | 0.75 |
| SFA 5.9, ER308 | 0.08 | 1.0-2.5 | 0.03 | 0.03 | 0.30-0.65 | 9.0-11.0 | 19.5-22.0 | 0.75 | -- | -- | -- | 0.75 |

(1) Values are maximum unless there is a range.

(2) For the lower RPV, the maximum carbon content is limited to 0.04 percent for the base metal and for the E/ER209 or E/ER240 weld filler metal.

**Table 3-2 Austenitic Stainless Steel Room Temperature Tensile Requirements**

| Material | Minimum Yield Strength | | Minimum Tensile Strength | | Minimum Total Elongation |
|---|---|---|---|---|---|
| | MPa | ksi | MPa | ksi | percent |
| SA-965 FXM-19 | 380 | 55 | 690 | 100 | 30 |
| SA-965 F304 and F316 | 205 | 30 | 485 | 70 | 30 |
| SFA 5.4 E209 and E240;<br>SFA 5.9 ER209 and ER240 | -- | -- | 690 | 100 | 15 |
| SFA 5.4 E308;<br>SFA 5.9 ER308 | -- | -- | 550 | 80 | 30 |

## 3.3 Austenitic Stainless Steel Literature Search Results

NuScale searched available literature for data on austenitic stainless steel properties to support the statement that austenitic stainless steel has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials.

Two studies contain tensile properties of irradiated FXM-19 (Reference 5.2.1 and Reference 5.2.2). Reference 5.2.1 contains data from FXM-19 irradiated in the RBT6 reactor, and Reference 5.2.2 contains data from the Idaho National Laboratory (INL) Advanced Test Reactor (ATR); the ATR study reports fracture toughness properties of irradiated FXM-19 (Reference 5.2.2).

While there are studies of irradiated FXM-19, there are no studies that assess FXM-19 at fluence levels relevant to the NPM lower RPV; however, there is extensive data for irradiated Type 3XX austenitic stainless steels because Type 3XX base metal, weld filler metal, and equivalent casting are used as structural materials in RVIs in operating LWRs. The Electric Power Research Institute (EPRI) Material Reliability Program reviews fracture toughness data on behalf of PWR owners and reports fracture toughness data at different fluences and temperatures (Reference 5.2.3). In addition, Argonne National Laboratory, on behalf of PWR owners and the NRC, reviewed irradiated fracture toughness data for Type 3XX austenitic stainless steels and summarized the results in NUREG/CR-7027 (Reference 5.2.4).

Thermal embrittlement or thermal aging embrittlement is a time- and temperature-dependent process whereby a material undergoes microstructural changes leading to decreased ductility and degradation of toughness and impact properties. According to Reference 5.2.3, wrought austenitic stainless steels are not subject to thermal embrittlement at PWR operating temperatures; however, cast austenitic stainless steel (CASS) and austenitic stainless steel welds are potentially susceptible because they contain residual delta ferrite.

Four LWRs not regulated by the NRC had or have RPVs made from austenitic stainless steel, as shown in Table 3-3. Information pertinent to RPV neutron embrittlement was found for the MH-1A and for the ATR.

**Table 3-3 Light Water Reactors with Reactor Pressure Vessels Made from Type 3XX Austenitic Stainless Steel**

| Reactor Name | Reactor Type | Operator | Years Active | Capacity | RPV Material |
|---|---|---|---|---|---|
| PM-1(1) | PWR(1) | US Air Force(1) | 1962 - 1968(1) | 1.25 MWe(1) | Type 347(2) |
| PM-3A(3) | PWR(3) | US Navy(3) | 1962 - 1972(3) | 1.75 MWe(3) | Type 347(2) |
| MH-1A(4) | PWR(4) | US Army(4) | 1967 - 1977(4) | 10 MWe(4) | Type 316(2) |
| ATR(5) | PWR(5) | Department of Energy (INL)(5) | 1967 - present(5) | 250 MWt(5) | Type 304(6) |

(1) Reference 5.2.5
(2) Reference 5.2.6
(3) Reference 5.2.7
(4) Reference 5.2.8
(5) Reference 5.2.9
(6) Reference 5.2.10

## 3.4 Evaluation of Data on Austenitic Stainless Steel to Address Regulations

### 3.4.1 10 CFR 50.60

10 CFR 50.60 (Reference 5.1.1) requires all LWRs to meet the fracture toughness and material surveillance program requirements for the RCPB in 10 CFR 50, Appendix G (Reference 5.1.3), and 10 CFR 50, Appendix H (Reference 5.1.4).

Current operating LWRs regulated by the NRC have RPVs made of carbon and low-alloy steels (ferritic materials), and the regulations and guidance contain data and procedures pertaining only to ferritic materials. Reference 5.1.1 requires compliance with the fracture toughness requirements in 10 CFR 50, Appendix G (Reference 5.1.3), and with the RVSP requirements in 10 CFR 50, Appendix H (Reference 5.1.4). Since the data and requirements for Reference 5.1.4 and Reference 5.1.4 only apply to ferritic materials, the NPM lower RPV must comply with the intent of the regulations in order to demonstrate compliance with GDC 14, GDC 15, and GDC 31.

A survey of data on austenitic stainless steel and its tensile strength and fracture toughness after irradiation, as well as the effects of neutron and thermal embrittlement on austenitic stainless steel, is in Section 3.4.1.1 and Section 3.4.1.2.

From the regulations, the RPV beltline is the region of the RPV that directly surrounds the effective height of the active core and is predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage. 10 CFR 50, Appendix H (Reference 5.1.4), requires monitoring for neutron and thermal embrittlement of ferritic materials in the RPV, including base metal, weld metal, and the heat-affected zone of the RCPB beltline during the RPV design life. Reference 5.1.4 requires a RVSP for the ferritic materials whose design life peak fluence exceeds 1E+17 n/cm$^2$, E > 1 MeV. The 57 effective full-power year (EFPY) peak fluence for the NPM lower

RPV beltline base metal (inside surface) is {{ }}2(a),(c),ECI; the 57 EFPY peak fluence for the NPM lower RPV beltline weld filler metal (inside surface) is {{ }}2(a),(c),ECI, E > 1 MeV. Since the beltline is in the lower RPV, which is made of austenitic stainless steel, the region of the RPV containing ferritic materials that experiences the highest fluence is evaluated for an RVSP. Therefore, the top surface of the upper RPV lower flange is evaluated. The 57 EFPY peak fluence for the top surface of the lower flange is {{ }}2(a),(c),ECI, E > 1 MeV. Therefore, the design life peak fluence for the upper RPV is less than the threshold value of 1E+17 n/cm$^2$, E > 1 MeV.

#### 3.4.1.1 Fracture Toughness Evaluation

Because Type 304 heavy reflectors surround the NPM fuel assemblies in the RVIs, the 57 EFPY peak fluence for the NPM lower RPV beltline base metal (inside surface) is {{ }}2(a),(c),ECI, E > 1 MeV, and the 57 EFPY peak fluence for the NPM lower RPV beltline weld filler metal (inside surface) is {{ }}2(a),(c),ECI, E > 1 MeV. The 57 EFPY peak fluence values convert to {{ }}2(a),(c),ECI for the NPM lower RPV base metal and {{ }}2(a),(c),ECI for the NPM lower RPV weld filler metal. Figure 3-2 and Figure 3-3 show the effect of neutron irradiation on the tensile properties of FXM-19 that was irradiated in light water moderated research reactors or test reactors.

Reference 5.2.1 reflects data from solution-annealed FXM-19 specimens irradiated and tested at 572 degrees F to 0.0007 dpa, 0.007 dpa, and 0.05 dpa. The uniform elongation declined slightly at 0.05 dpa but remained high (greater than 40 percent), and the total elongation did not change. Therefore, neutron embrittlement of solution-annealed FXM-19 was minor after irradiation to 0.05 dpa, which bounds the 57 EFPY peak fluence for the NPM lower RPV.

Reference 5.2.2 reports data from mill-annealed FXM-19 specimens irradiated at 550 degrees F and tested at 572 degrees F; however, no unirradiated specimens were tested at ATR, so the unirradiated data in Figure 3-2 is used for comparison. The ATR specimens were irradiated to 0.076 dpa. The uniform elongation increased by about 28 percent, and the total elongation increased by about 10 percent. Uniform elongation remained very high (greater than 30 percent) at 0.076 dpa. Therefore, neutron embrittlement of solution-annealed FXM-19 was minor after irradiation to 0.076 dpa, which bounds the 57 EFPY peak fluence for the NPM lower RPV.

Figure 3-2 and Figure 3-3 show the effect of neutron irradiation on the tensile properties of FXM-19 that was irradiated in light water moderated research reactors or test reactors. Table 3-4 summarizes the fracture toughness test results of the mill-annealed FXM-19 irradiated at the ATR, corresponding to Figure 3-3. The irradiated and unirradiated fracture toughness specimens were in the L-T orientation with respect to the original major working direction and were tested at 550 degrees F. Irradiation to 0.08 dpa caused only 4 percent reduction in average plane-strain fracture-toughness values ($K_{JQ}$ or $K_{Jc}$) at 550 degrees F. Therefore, neutron embrittlement of mill-annealed FXM-19 after irradiation to 0.08 dpa was insignificant. It is noted that 0.08 dpa is much greater than the 57 EFPY peak neutron dose of the lower RPV.

Although a neutron dose above 0.08 dpa is far beyond the lower RPV design life fluence, the mill-annealed FXM-19 still possessed high plane-strain fracture toughness exceeding 200 MPa√m after irradiation to 1.4 dpa, which is greater than the 57 EFPY peak neutron dose of the lower RPV.

**Figure 3-2 Typical Stress-Strain Curves of Solution-Annealed FXM-19 Irradiated in RBT6**

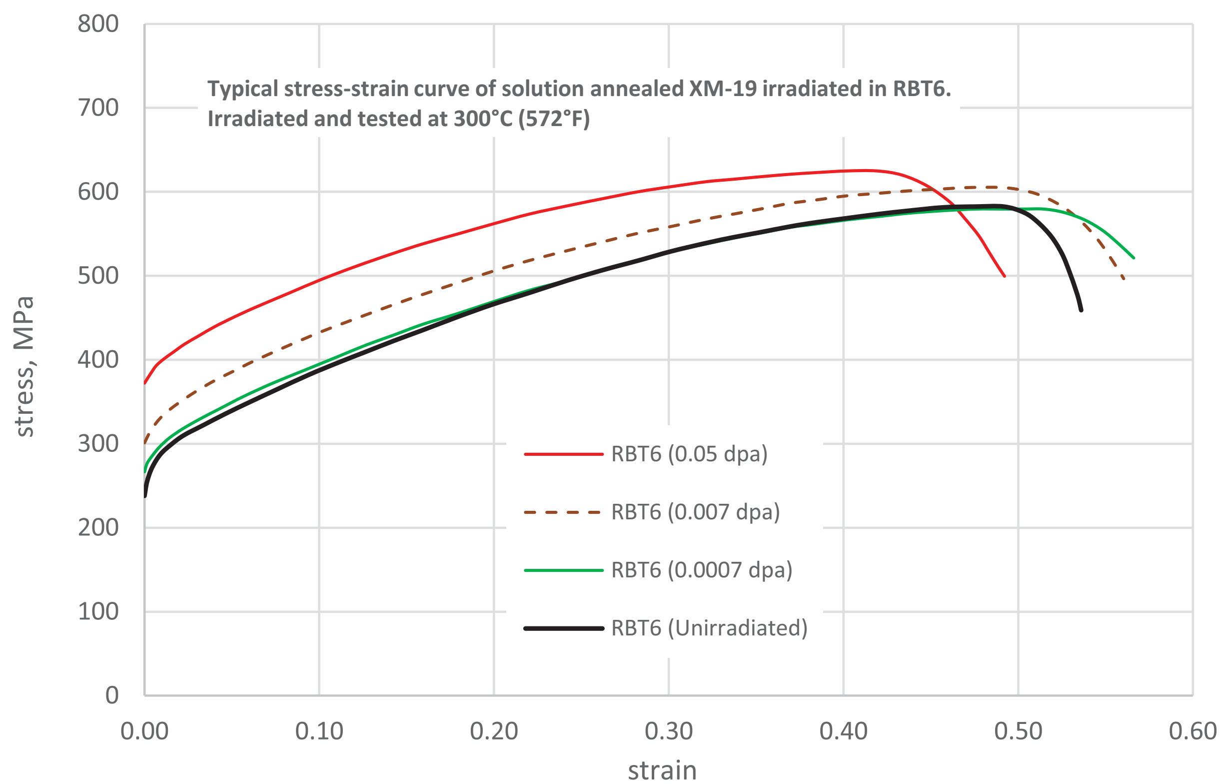


**Figure 3-3 Typical Stress-Strain Curves of Mill-Annealed FXM-19 Irradiated in the Advanced Test Reactor**

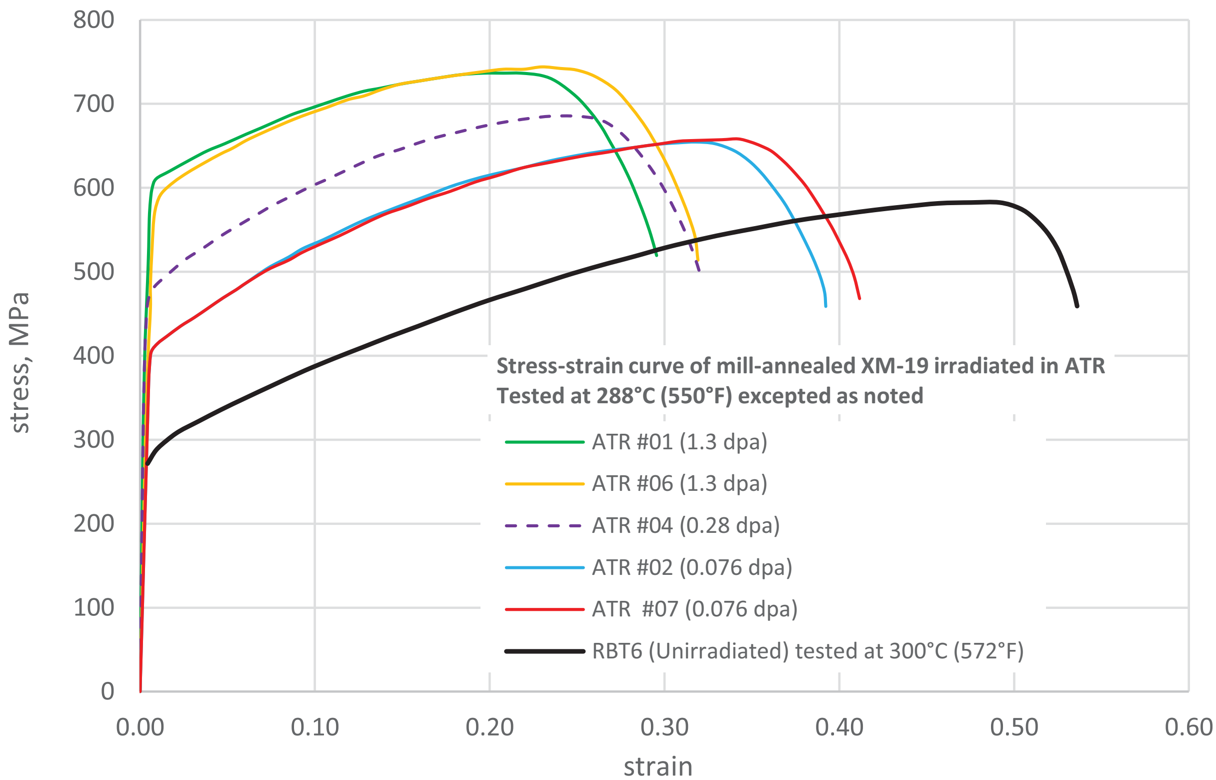


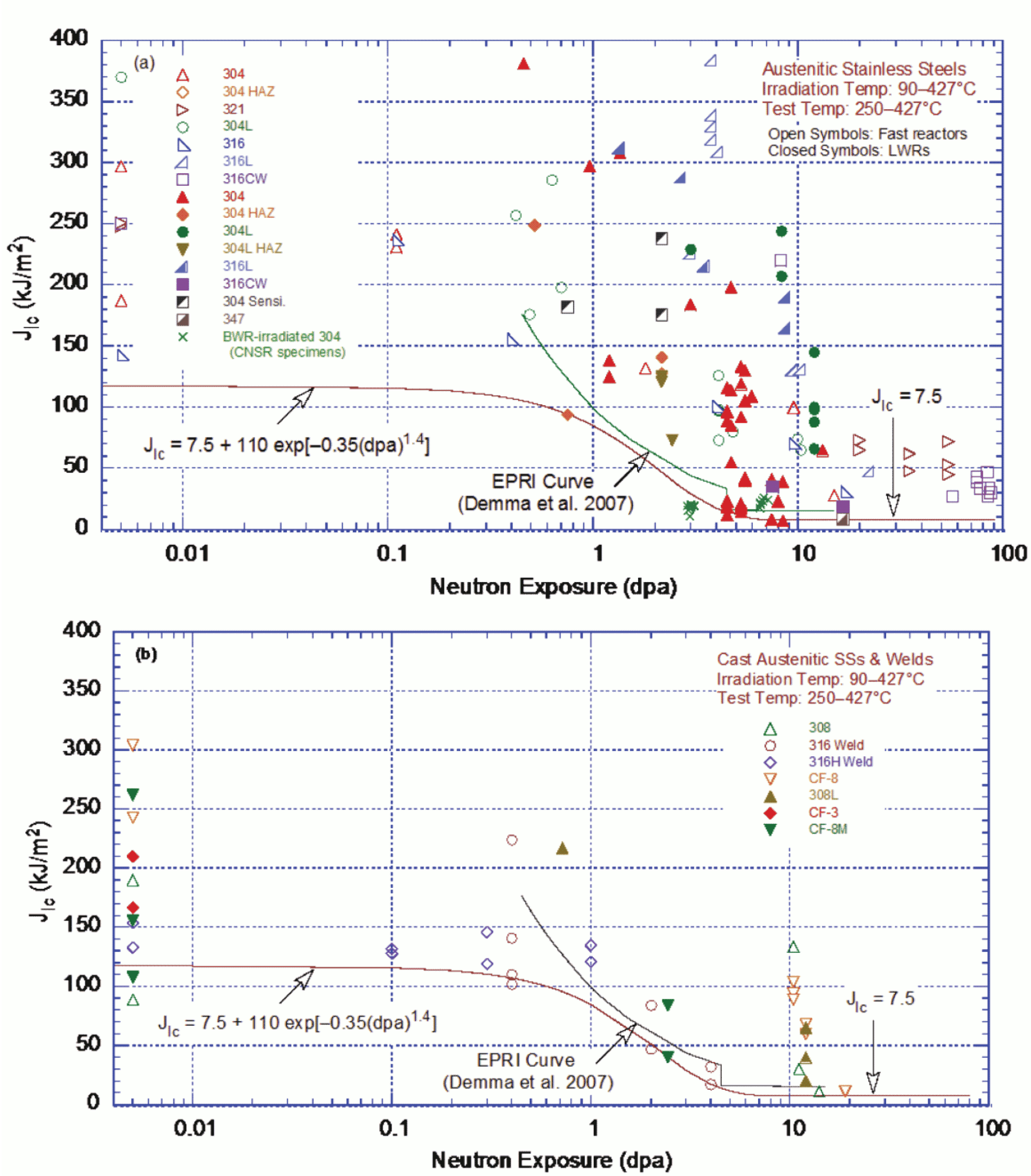
**Table 3-4 Fracture Toughness of Unirradiated and Irradiated Mill-Annealed FXM-19**

| FXM-19 Fracture Toughness Specimen(1) | Specimen Size | Average Irradiation Temperature (degrees F) | Fluence | $J_Q$ or $J_{Ic}$ | $K_{JQ}$ or $K_{Ic}$ | | |
|---|---|---|---|---|---|---|---|
| | | | dpa | kJ/m$^2$ | MPa√m | ksi√in(2) | Percent Change |
| Unirradiated C747 | 1T-CT | Unirradiated | 0 | 388 | 296 | 269 | -- |
| Unirradiated C748 | 1T-CT | Unirradiated | 0 | 354 | 283 | 258 | -- |
| Average of two unirradiated specimens | | | | 371 | 290 | 263 | -- |
| Irradiated in ATR 10A0001A02 | 0.4T-CT | 621 | 0.08 | 312 | 265 | 241 | -- |
| Irradiated in ATR 10A0001A07 | 0.4T-CT | 642 | 0.08 | 377 | 291 | 265 | -- |
| Average of two specimens irradiated to 0.08 dpa | | | | 345 | 278 | 253 | -4 |
| Irradiated in ATR 10A0001B01 | 0.4T-CT | 624 | 0.29 | 231 | 230 | 209 | -- |
| Irradiated in ATR 10A0001B02 | 0.4T-CT | 662 | 0.29 | 310 | 266 | 242 | -- |
| Average of two specimens irradiated to 0.29 dpa | | | | 271 | 248 | 226 | -14 |
| Irradiated in ATR 10A0001D05 | 0.4T-CT | 631 / 507(3) | 1.47 | 303 | 262 | 238 | -- |
| Irradiated in ATR 10A0001D01 | 0.4T-CT | 626 / 502(3) | 1.43 | 203 | 215 | 196 | -- |
| Irradiated in ATR 10A0001D04 | 0.4T-CT | 574 / 507(3) | 1.41 | 251 | 237 | 216 | -- |
| Average of three specimens irradiated from 1.41 dpa to 1.47 dpa | | | | 252 | 238 | 217 | -18 |

(1) Irradiated values are from Table 4-6 of Reference 5.2.2. Unirradiated values of the same FXM-19 heat is from Table 1 of Reference 5.2.11. Values listed are from specimens tested in the L-T orientation.
(2)1 MPa√m = 0.910 ksi√in
(3) First cycle irradiation temperature and second cycle irradiation temperature, respectively.

Argonne National Laboratory reviewed irradiated fracture toughness data of Type 3XX austenitic stainless steels and summarized the results in Reference 5.2.4. Table 3-4 shows the fracture toughness as a function of fluence level with an irradiation temperature and test temperature range of 482 degrees F to 800 degrees F. Based on the data in Table 3-4, Reference 5.2.4 proposes a 0.5 dpa threshold fluence for austenitic stainless steel base metal and a 0.3 dpa threshold fluence for austenitic stainless steel weld filler metal; the threshold fluence is the fluence below which irradiation has little to no effect on fracture toughness.

**Figure 3-4 Fracture Toughness of Irradiated Type 3XX Austenitic Stainless Steel**



FXM-19 typically contains 0.3 percent nitrogen, which is higher than the nitrogen level of Type 3XX austenitic stainless steels. Solid solution strengthening with nitrogen is also used for Type 3XX grades with a nitrogen range of 0.10 percent to 0.16 percent. There is no evidence in the literature to suggest that higher manganese and nitrogen content in FXM-19 compared to Type 3XX austenitic stainless steels contributes to heightened effects of neutron embrittlement.

Table 3-5 compares the fracture toughness of FXM-19 from Reference 5.2.2 (shown in Table 3-4) with the Reference 5.2.4 fracture toughness of Type 3XX austenitic stainless steel. The irradiated FXM-19 values from Reference 5.2.2 are well above the lower bound of the Reference 5.2.4 data. Figure 3-5 also shows that the FXM-19 fracture toughness of mill-annealed FXM-19 responds to neutron exposure similarly to Type 3XX austenitic stainless steels.

Finally, for the NPM, the comparison of 57 EFPY peak fluence values with the threshold values from Reference 5.2.4 is shown in Table 3-5. Table 3-5 supports the conclusion that neutron embrittlement is not a concern for the NPM lower RPV made of austenitic stainless steel.

**Figure 3-5 Comparison of FXM-19 Fracture Toughness with Type 3XX Austenitic Stainless Steel Fracture Toughness**

{{

}}2(a),(c),ECI

**Table 3-5 Comparison of Design Life Peak Fluence with Threshold Fluence**

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| | | | |
| | | | |

}}2(a),(c),ECI

#### 3.4.1.2 Thermal Embrittlement Evaluation

According to Reference 5.2.3, wrought austenitic stainless steels are not subject to thermal embrittlement at PWR operating temperatures. However, CASS and austenitic stainless steel weld filler metals are potentially susceptible to thermal embrittlement because they contain some residual delta ferrite. The NPM lower RPV does not use CASS but does contain austenitic stainless steel weld filler metal (E/ER209 or E/ER240).

Based on the thermal embrittlement data for Type 3XX austenitic stainless steel, Table 3-2 of Reference 5.2.3 lists criteria to help evaluate when there is a potential for synergistic effects between thermal embrittlement and neutron embrittlement of CASS and austenitic stainless steel weld filler metal. The criteria are based on molybdenum content, which is known to increase thermal embrittlement of austenitic stainless steel welds; delta ferrite content, which is known to increase thermal embrittlement of austenitic stainless steel welds; and neutron dose. The following are the criteria for austenitic stainless steel weld filler metal where thermal embrittlement is of concern.

1. end of life neutron dose greater than 0.5 dpa with any molybdenum and delta ferrite content
2. molybdenum content less than 0.50 percent along with greater than 20 percent delta ferrite
3. molybdenum content greater than 0.50 percent along with greater than 14 percent delta ferrite

The 57 EFPY peak life fluence for the NPM lower RPV is well below the end of life neutron dose criterion of 0.5 dpa, so the first criterion is not met.

Welds using E/ER209 filler metal contain 1.5 percent to 3.0 percent molybdenum, and welds using E/ER240 filler metal contain up to 0.75 percent molybdenum, which corresponds to item three above. Based on the revised DeLong diagram in Reference 5.2.12, 14 percent delta ferrite is approximately equivalent to 16 ferrite number (FN). Therefore, the NPM design limits the delta ferrite in the lower RPV weld filler metal (E/ER209 or E/ER240) to 16 FN in order to avoid synergistic effects between thermal embrittlement and neutron embrittlement in the weld filler metal.

#### 3.4.1.3 Material Surveillance Program Evaluation

Austenitic stainless steels do not undergo ductile-to-brittle transition temperature and have higher toughness than ferritic materials used for ASME BPVC Section III Class 1 pressure-retaining components. These factors are why Reference 5.1.9, NB-2311, does not have impact test requirements for austenitic stainless steels, including SA-965 Grade FXM-19 used in the NPM lower RPV.

The requirements for establishing a material surveillance program rely on the nil-ductility reference temperature ($RT_{NDT}$) calculation in Reference 5.1.9, NB-2331. However, Reference 5.1.9, NB-2331, is not applicable to austenitic stainless steels because there are no impact test requirements. The drop-weight test required by Reference 5.1.9, NB-2331, to establish $RT_{NDT}$ is limited to ferritic materials. Because $RT_{NDT}$ cannot be calculated for the SA-965 Grade FXM-19 lower RPV, the requirements in Reference 5.1.1, Reference 5.1.3, and Reference 5.1.4 do not apply to the NPM lower RPV. The NRC endorsed the ASME BPVC in 10 CFR 50.55a.

The US600 (Reference 5.2.14) design lower RPV is designed with SA-508 Grade 3 Class 1 ferritic steel and compatible weld filler metals. Since SA-965 Grade FXM-19 austenitic stainless steel and E/ER209 or E/ER240 weld filler metals have superior ductility and are less susceptible to the effects of neutron and thermal embrittlement than SA-508 Grade 3 Class 1 ferritic steel and its compatible weld filler metals, the probability of rapid failure in the NPM lower RPV is even lower than that of the lower RPV of the approved US600 design. In addition, there have been no reports of rapidly propagating failures of ferritic steel RPVs in LWRs actors worldwide, after accumulating over 16,000 reactor-years of operation as of June 2020 (Reference 5.2.13).

### 3.4.2 10 CFR 50.61

10 CFR 50.61 (Reference 5.1.2) requires protection against PTS events. Pressurized thermal shock events are events or transients in PWRs causing severe overcooling (thermal shock) concurrent with or followed by significant pressure within the RPV.

The PTS screening criterion ($RT_{PTS}$) calculation uses the acceptance criteria in 10 CFR 50.61(b)(2). The PTS screening methodology in Reference 5.1.2 is based on calculating $RT_{PTS}$, which is the $RT_{NDT}$ evaluated for end of design life peak fluence for each of the RPV beltline materials using the Reference 5.1.2 procedures. The Reference 5.1.2 procedures require the use of $RT_{NDT(u)}$, which is the unirradiated reference temperature established by impact testing according to Reference 5.1.9, NB-2331. The NRC endorsed the ASME BPVC in 10 CFR 50.55a. Impact testing is not required for austenitic stainless steels because they do not undergo ductile-to-brittle transition temperature and have higher toughness than ferritic materials. Because $RT_{NDT}$ cannot be calculated for the austenitic stainless steels in the lower RPV, the requirements in Reference 5.1.2 do not apply to the NPM lower RPV. Furthermore, the chemistry factors in Reference 5.1.2 assume use of carbon or low-alloy steel because Equation 4 of Reference 5.1.2 uses a chemistry factor that corresponds to copper and nickel content of ferritic materials. There is no chemistry factor for austenitic stainless steel in Reference 5.1.2.

As noted in Section 3.4.1, the beltline of the NPM lower RPV is not made of ferritic steel; therefore, the region of the RPV containing ferritic materials that experiences the highest fluence is the top surface of the upper RPV lower flange. The upper RPV 57 EFPY peak fluence is {{ }}2(a),(c),ECI, E > 1 MeV. Consequently, the NPM upper RPV does not require PTS screening since the design life peak fluence is less than 1E+17 n/cm$^2$, E > 1 MeV.

## 4.0 Summary and Conclusions

The US460 standard design meets the requirements in GDC 14, GDC 15, GDC 31, and GDC 32. While the requirements in 10 CFR 50.60 and 10 CFR 50.61 cannot be used for the NPM lower RPV because it is made of austenitic stainless steel, the design satisfies the requirements of the GDCs.

The US460 standard design meets GDC 14 by using austenitic stainless steel in the RPV beltline, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB.

The US460 standard design meets GDC 15 by using austenitic stainless steel in the RPV beltline, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB. The US460 standard design ensures that the RCPB limits are not exceeded during operation.

The US460 standard design meets GDC 31 by using austenitic stainless steel in the RPV beltline, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB.

The US460 design meets item (2) of GDC 32 by using austenitic stainless steel in the RPV beltline, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB. Item 2 of GDC 32 requires an appropriate material surveillance program. An RVSP is not necessary to ensure the safety of the US460 standard design because the austenitic stainless steel used in the lower RPV is less susceptible to the effects of neutron and thermal embrittlement compared to ferritic materials. Therefore, the design satisfies Item 2 of GDC 32 without an RVSP.

The US600 design (Reference 5.2.14) used austenitic stainless steel for the lower containment vessel (CNV) because its material properties are less susceptible to the effects of neutron and thermal embrittlement than ferritic materials. In Section 6.1.1.4.2 of the US600 design final safety evaluation report (Reference 5.2.15), the NRC stated:

> *The staff finds the use of SA-965, Grade FXM-19, and its associated weld filler metals acceptable for use in the lower portion of the CNV, as the calculated fluence to the CNV is lower than what is expected to cause embrittlement, and the selection of SA-965, Grade FXM-19, an austenitic stainless steel, is resistant to radiation embrittlement.*

In Section 6.2.7.4 of the final safety evaluation report (Reference 5.2.15), the NRC stated:

> *Within the ASME Code, detailed fracture toughness requirements are placed on ferritic materials, as nonferritic materials exhibit sufficient inherent fracture toughness that additional requirements are deemed unnecessary. For example, the austenitic*

> *stainless steel used for the CNV lower shell, SA-965, FXM-19, was explicitly chosen specifically for its superior fracture toughness and resistance to neutron embrittlement.*

The results of this report confirm that austenitic stainless steels and compatible weld filler metals are likewise acceptable for use in the lower RPV without additional fracture toughness requirements because they have superior ductility and are less susceptible to the effects of neutron and thermal embrittlement than ferritic materials.

# 5.0 References

## 5.1 Source Documents

5.1.1 U.S. Code of Federal Regulations, “Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation,” Section 50.60, Part 50, Chapter I, Title 10, “Energy,” (10 CFR 50.60).

5.1.2 *U.S. Code of Federal Regulations*, Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events,” Section 50.61, Part 50, Chapter I, Title 10, Energy,” (10 CFR 50.61).

5.1.3 *U.S. Code of Federal Regulations*, Fracture Toughness Requirements,” Appendix G, Part 50, Chapter I, Title 10, Energy,” (10 CFR 50, Appendix G).

5.1.4 *U.S. Code of Federal Regulations*, Reactor Vessel Material Surveillance Program Requirements,” Appendix H, Part 50, Chapter I, Title 10, Energy,” (10 CFR 50, Appendix H).

5.1.5 *U.S. Code of Federal Regulations*, Reactor Coolant Pressure Boundary,” Criterion 14, Appendix A, Part 50, Chapter I, Title 10, Energy,” (10 CFR 50, Appendix A).

5.1.6 *U.S. Code of Federal Regulations*, Reactor Coolant System Design,” Criterion 15, Appendix A, Part 50, Chapter I, Title 10, Energy,” (10 CFR 50, Appendix A).

5.1.7 *U.S. Code of Federal Regulations*, Fracture Prevention of Reactor Coolant Pressure Boundary,” Criterion 31, Appendix A, Part 50, Chapter I, Title 10, Energy,” (10 CFR 50, Appendix A).

5.1.8 *U.S. Code of Federal Regulations*, Inspection of Reactor Coolant Pressure Boundary,” Criterion 32, Appendix A, Part 50, Chapter I, Title 10, Energy,” (10 CFR 50, Appendix A).

5.1.9 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 Edition, Section III, Subsection NB, “Rules for Construction of Nuclear Facility Components,” New York, NY.

## 5.2 Referenced Documents

5.2.1 Pokrovsky, A.S., et al., Effect of Neutron Irradiation on Tensile Properties of Austenitic Steel XM-19 for the ITER Application,” *Journal of Nuclear Materials* (2011): 417:874-877.

5.2.2 Idaho National Laboratory INL/EXT-20-58432, Revision 1, “Irradiation and PIE of Alloys X-750 and XM-19 (EPRI Phase III),” Jackson, J.H., et al., July 2020.

5.2.3 Electric Power Research Institute MRP-175, “Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values,” Palo Alto, CA, 2005: 1012081.

5.2.4 U.S. Nuclear Regulatory Commission, “Degradation of LWR Core Internal Materials due to Neutron Irradiation,” NUREG/CR-7027, December 2010.

5.2.5 Characteristics of PM-1 (Sundance),” *Nucleonics*, 1962. (from Wikipedia)

5.2.6 Karnoski, Jr., P.J., et al., Stainless Steel Reactor Pressure Vessels,” *Nuclear Engineering and Design*, (1970): Volume 11, Issue 3: 347-167.

5.2.7 Adams Atomic Engines, Inc., “PM-3A Design and Construction,” October 1996. (from Wikipedia)

5.2.8 Floating Nuclear Plant Sturgis Dismantled,” *The Maritime Executive*, March 16, 2019. (from Wikipedia)

5.2.9 Idaho National Laboratory, “Advanced Test Reactor User Guide.”

5.2.10 Phillips Petroleum Company, “Reactor Vessel,” Safety Analysis Report, Advanced Test Reactor, Volume 1 of 2, IDO-17021, April 1965: 93-108.

5.2.11 Andresen, P., et al., SCC and Fracture Toughness of XM-19,” *Proceedings of the 18$^{th}$ International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors*.

5.2.12 Long, C.J. and DeLong, W.T., The Ferrite Content of Austenitic Stainless Steel Weld Metal,” *Welding Research Supplement, Welding Journal*, July 1973.

5.2.13 International Atomic Energy Agency, “Operating Experience with Nuclear Power Stations in Member States (2020 Edition),” Vienna, Austria, August 2020.

5.2.14 NuScale Power, LLC, NuScale Standard Plant Design Certification Application, Rev. 5, July 29, 2020 (ML20225A044), Portland, OR.

5.2.15 U.S. Nuclear Regulatory Commission, “NuScale Design Certification Final Safety Evaluation Report, FSER Chapter 6 - Engineered Safety Features,” July 23, 2020 (ML20205L406).