5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

The reactor coolant system (RCS) and connected systems include those systems and components that contain or transport fluids coming from or going to the reactor core. These systems form a major portion of the reactor coolant pressure boundary (RCPB). This chapter provides information regarding the RCS and pressure-containing appendages out to and including isolation valves. This grouping of components is defined as the RCPB, as defined in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.2.

5.1 Summary Description

This section of the Fermi 3 combined license (COL) Final Safety Analysis Report (FSAR) incorporates by reference, with no departures or supplements, Subsection 5.1, "Summary Description," of the Economic Simplified Boiling-Water Reactor (ESBWR) design control document (DCD) Revision 9, referenced in 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Appendix [X], "Design Certification Rule for the Economic Simplified Boiling Water Reactor," with no departures or supplements. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the application and checked the referenced DCD. The staff's review confirmed there is no outstanding information, related to this section. Pursuant to the 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix [X], Section VI.B.1, all nuclear safety issues relating to the summary description have been resolved.

5.2 Integrity of Reactor Coolant Pressure Boundary

This section of the FSAR discusses measures employed to provide and maintain the integrity of the RCPB.

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR 50.55a

5.2.1.1.1 *Introduction*

This subsection of the Fermi 3 COL FSAR, Revision 3, addresses the American Society of Mechanical Engineers (ASME) Code edition and addenda to be used at Fermi 3, in order to show compliance with NRC regulations in 10 CFR 50.55a.

5.2.1.1.2 *Summary of Application*

Section 5.2 of the Fermi 3 COL FSAR, Revision 3, incorporates by reference Section 5.2 of the certified ESBWR DCD, Revision 9. In addition, in FSAR Subsection 5.2.1.1, the applicant provides the following:

Supplemental Information

- STD SUP 5.2-2

In FSAR Revision 3, Subsection 5.2.1.1, the applicant provides supplemental information that preservice inspection (PSI) and inservice inspection (ISI) of the RCPB are conducted in accordance with the applicable edition and addenda of the ASME Boiler and Pressure Vessel (BPV) Code, Section XI, required by 10 CFR 50.55a. FSAR Subsection 5.2.1.1 also states that:

> As described in DCD Section 3.9.6 for pumps and valves, and in DCD Section 3.9.3.7.1 for dynamic restraints, preservice and in-service testing of RCPB components is in accordance with the edition and addenda of the ASME OM Code required by 10 CFR 50.55a.

5.2.1.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1966, the Final Safety Evaluation Report (FSER) related to the certified ESBWR DCD.

In addition, the relevant requirements of the Commission regulations for the compliance with the 10 CFR 50.55a, and the associated acceptance criteria, are described in Subsection 5.2.1.1 of NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)."

In particular, NRC regulations in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and Part 52 provide the regulatory basis for NRC staff review of the information provided in the Fermi 3 COL application. For example, NRC regulations in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 1, "Quality standards and records," require that nuclear power plant structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Further, NRC regulations in 10 CFR 50.55a, as related to the establishment of the minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of nuclear power plant components, require conformance with appropriate editions of published industry codes and standards.

Also, NRC staff followed the guidance in Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," June 2007, in evaluating Fermi 3 COL FSAR, Subsection 5.2.1.1 for compliance with NRC regulations.

5.2.1.1.4 *Technical Evaluation*

As documented in NUREG–1966, NRC staff reviewed and approved Section 5.2 of the certified ESBWR DCD. The staff reviewed Section 5.2 of the Fermi 3 COL FSAR, Revision 3, and checked the referenced ESBWR DCD to ensure that the combination of the information in the ESBWR DCD and the information in the COL FSAR appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information contained in the application and the information incorporated by reference address the relevant information related to this section.

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[1] See "*Finality of Referenced NRC Approvals*," in SER Section 1.2.2, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

Section 1.2.3 of this safety evaluation report (SER) provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER with open items issued for the North Anna application were equally applicable to the Fermi COL application, the staff undertook the following reviews:

- The staff compared the North Anna 3 COL FSAR, Revision 1, to the Fermi COL FSAR. In performing this comparison, the staff considered changes made to the Fermi COL FSAR (and other parts of the COL application, as applicable) resulting from requests for additional information (RAIs) and open and confirmatory items identified in the North Anna SER with open items.

- The staff confirmed that the applicant endorsed all responses to RAIs identified in the corresponding standard content (the North Anna SER) evaluation.

- The staff verified that the site-specific differences were not relevant to this section.

The staff has completed its review and found the evaluation performed for the North Anna standard content to be directly applicable to the Fermi COL application. This standard content material is identified in this SER by use of italicized, double indented formatting.

The staff reviewed the information in the Fermi 3 COL FSAR as follows:

Supplemental Information

The following portion of this technical evaluation section is reproduced from Subsection 5.2.1.1 of North Anna Unit 3 SER (ML091730304):

- STD SUP 5.2-2

  In request for additional information (RAI) 05.02.01.01-1, NRC staff requested that Dominion address the application of other sections of the ASME BPV Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) in its implementation of the ESBWR reactor design. In response to this RAI, by letter dated September 11, 2008, the applicant stated that the FSAR would be revised to provide references to the appropriate sections that discuss compliance with the ASME BPV Code, Section XI, and the ASME OM Code. As a result, Revision 1 of FSAR Section 5.2.1.1 states that the [PSI] *and ISI of the RCPB will be conducted in accordance with the applicable edition and addenda of the ASME BPV Code, Section XI, required by 10 CFR 50.55a as described in FSAR Section 5.2.4. FSAR Section 5.2.1.1 also states that preservice and inservice testing (IST) of the RCPB components will be in accordance with the edition and addenda of the ASME OM Code required by 10 CFR 50.55a as described in DCD Section 3.9.6, for pumps and valves and DCD Section 3.9.3.7.1, for dynamic restraints. NRC staff has verified these revisions and finds that the reference to the applicable sections of the ESBWR DCD for the* *application of appropriate ASME Code editions and addenda is consistent with NRC regulations, and therefore is acceptable. Therefore, this RAI is closed.*

Revision 3 of the Fermi 3 COL FSAR, Subsection 5.2.1.1 is consistent with these statements provided in the North Anna 3 FSAR, however, the above quoted text is missing that portion of the text that refers to “PSI”, which has been inserted in brackets. Therefore, NRC staff finds that the reference to the applicable sections of the ESBWR DCD for the application of appropriate ASME Code editions and addenda meets 10 CFR 50.55a requirements, and the guidance in NUREG-0800, and therefore is acceptable.

5.2.1.1.5 *Post Combined License Activities*

There are no post COL activities related to this section.

5.2.1.1.6 *Conclusion*

The NRC staff’s finding related to information incorporated by reference is in NUREG–1966. NRC staff reviewed the application and checked the referenced DCD. The staff’s review confirmed that the applicant has addressed the required information, and no outstanding information is expected to be addressed in the Fermi 3 COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix [X], Section VI.B.1, all nuclear safety issues relating to this section that were incorporated by reference have been resolved.

In addition, the staff compared the additional COL supplemental information in the application to the relevant NRC regulations, the guidance in Subsection 5.2.1.1 of NUREG–0800, and other NRC RGs. The staff’s review concluded that the applicant has presented adequate information in the Fermi 3 COL FSAR to meet the requirements of the Codes and Standards Rule (10 CFR 50.55a).

5.2.1.2 Applicable Code Cases

5.2.1.2.1 *Introduction*

This subsection addresses the ASME BPV Code and ASME “Operation and Maintenance of Nuclear Power Plants” (OM Code) Code Cases that are applicable to the Fermi 3 COL FSAR, Revision 3. This section also addresses NRC RGs that indicate the acceptance of ASME Code Cases with or without conditions. In general, a Code Case is developed by ASME based on inquiries from the nuclear industry associated with Code clarification, modification or alternative to the Code. All Code Cases will remain valid and available for use until annulled by the ASME. ASME Code Cases acceptable to the NRC staff are published in RG 1.84, “Design and Fabrication Code Case Acceptability, ASME Section III, Division 1,” RG 1.147, “Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1,” and RG 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code,” in accordance with requirements of 10 CFR 50.55a(b)(4), (5) and (6), respectively.

5.2.1.2.2 *Summary of Application*

Section 5.2, “Integrity of Reactor Coolant Pressure Boundary,” of the Fermi 3 COL FSAR, Revision 3 incorporates by reference Subsection 5.2.1.2, “Applicable Code Cases,” of the ESBWR DCD, Tier 2 (Revision 9), without supplemental information or departures.

5.2.1.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1966, the FSER related to the certified ESBWR DCD. NRC regulations in 10 CFR Part 50 and Part 52 provide the regulatory basis for the NRC staff review of the information in the Fermi 3 COL application. For example, 10 CFR Part 50, Appendix A, GDC 1 requires that nuclear power plant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Further, NRC regulations in 10 CFR 50.55a, as related to the establishment of the minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of nuclear power plant components, require conformance with appropriate editions of published industry codes and standards.

As one acceptable means of meeting the applicable NRC regulations, RG 1.84 lists ASME BPV Code, Section III Code Cases related to design, fabrication, materials, and testing, which are acceptable with applicable conditions for implementation at nuclear power plants. RG 1.147 lists ASME BPV Code, Section XI Code Cases, which are acceptable with applicable conditions for use in the ISI of nuclear power plant components and their supports. RG 1.192 lists Code Cases related to the ASME OM Code for operation and maintenance of nuclear power plant components, which are acceptable with applicable conditions for implementation at nuclear power plants.

NRC staff followed the guidance in SRP Subsection 5.2.1.2, “Applicable Code Cases,” and RG 1.206 in evaluating Fermi 3 COL FSAR, Subsection 5.2.1.2 for compliance with NRC regulations.

5.2.1.2.4 *Technical Evaluation*

As documented in NUREG–1966, NRC staff reviewed and approved Subsection 5.2.1.2 of the certified ESBWR DCD Tier 2. The staff reviewed Subsection 5.2 of the Fermi 3 COL FSAR, Revision 3, and checked the referenced ESBWR DCD to ensure that the combination of the information in the ESBWR DCD and the information in the COL FSAR appropriately represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information contained in the application and the information incorporated by reference addresses the relevant information related to this section.

In letters dated February 16, 2009, July 19, 2010, and September 21, 2010, Detroit Edison notified the NRC that it had assumed the role of R-COL applicant for the ESBWR design and that it adopted the RAI responses related to FSAR Subsection 5.2.1.2 provided by Dominion Power for the previous R-COL plant (North Anna Unit 3 ESBWR). The NRC staff review of these RAIs as they relate to Fermi 3 COL FSAR Subsection 5.2.1.2 is provided below.

Fermi 3 COL FSAR, Subsection 5.2.1, “Compliance with Codes and Code Cases,” incorporates by reference ESBWR DCD, Tier 2, Subsection 5.2.1.2 without departures or supplemental information. ESBWR DCD, Tier 2, Subsection 5.2.1.2 indicates that the various ASME Code

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1 See “*Finality of Referenced NRC Approvals*,” in SER Section 1.2.2, for a discussion on the staff’s review related to verification of the scope of information to be included in a COL application that references a design certification.

Cases that may be applied to components in the ESBWR design are listed in ESBWR DCD, Tier 2, Table 5.2-1. ESBWR DCD, Tier 2, Subsection 5.2.1.2 also notes that RG 1.84 and RG 1.147 provide a list of ASME Code design, fabrication and inspection code cases that have been generically approved by the NRC staff.

In **RAI 05.02.01.02-1** for the previous R-COLA plant, the NRC staff requested that Dominion discuss the use of any Code Cases not listed in ESBWR DCD, Tier 2, Table 5.2-1, related to the ASME BPV Code and OM Code**.** Detroit Edison adopted Dominion's RAI response dated September 11, 2008, that stated that no ASME BPV Code, Section III or Section XI, Code Cases other than those listed in ESBWR DCD, Tier 2, Table 5.2-1 had been identified as necessary. The RAI response indicated that other Code Cases approved by the NRC in RG 1.147 might be used during development and implementation of the preservice and inservice inspection programs. ESBWR DCD, Tier 2, Subsection 3.9.3.7.1b, "Inspection, Testing, Repair, and/or Replacement of Snubbers," references RG 1.192 for use of Code Cases (such as Code Case OMN-13) applicable to inservice testing of dynamic restraints. ESBWR DCD, Tier 2, Subsection 3.9.6.6, "10 CFR 50.55a Relief Requests and Code Cases," indicates that the IST program for the ESBWR does not use any ASME Code Cases. The RAI response states that other Code Cases approved by the NRC in RG 1.192 might be used during the development and implementation of the preservice testing and IST programs. The RAI response provided a planned revision to the FSAR to reference RG 1.192 in Subsection 5.2.1.2. subsequently, Revision 6 to ESBWR DCD, Tier 2, Subsection 5.2.1.2 included RG 1.192 in addition to RGs 1.84 and 1.147 for the use of ASME Code Cases. ESBWR DCD, Tier 2, Subsection 5.2.1.2 also states that the use of the ASME OM Code, including the application of any OM Code Cases with the conditions and restrictions of RG 1.192, is described in DCD Tier 2, Section 3.9. Although an FSAR revision was considered in the RAI response, the NRC staff finds Fermi 3 COL FSAR, Subsection 5.2.1.2 to be acceptable without specific discussion of ASME OM Code Cases in light of the consideration of those code cases in Revision 9 to the ESBWR DCD. Therefore, **RAI 05.02.01.02-1** is resolved.

ESBWR DCD, Tier 2, Subsection 5.2.1.2 states that annulled cases are considered active for equipment that has been contractually committed to fabrication prior to the annulment. In **RAI 05.02.01.02-2** for the previous R-COLA plant, the NRC staff requested that Dominion discuss its compliance with the requirements regarding the use of annulled Code Cases specified in 10 CFR 50.55a(b)(4), (5), and (6). Detroit Edison adopted Dominion's RAI response dated September 11, 2008, that stated that design, fabrication, and construction of safety-related components has been conducted in accordance with ASME Code requirements specified in ESBWR DCD, Tier 2, Table 3.2-1, "Classification Summary," and Table 3.2-3, "Quality Group Designations – Codes and Industry Standards." The RAI response also noted that ESBWR DCD, Tier 2, Subsection 5.2.1.1 specifies that the ESBWR meets the relevant requirements of 10 CFR 50.55a. The RAI response stated that this includes application of any limitations and modifications to the applicable Code edition and addenda as may be specified in 10 CFR 50.55a, including any limitations regarding the use of annulled Code Cases. With respect to preservice and inservice inspection and testing of safety-related components, the RAI response indicated that the applicable edition and addenda of the ASME Code as identified in 10 CFR 50.55a is used, subject to the limitations and modifications specified in 10 CFR 50.55a, including those limitations specified in 10 CFR 50.55a(b)(4), (5), and (6) regarding the use of Code Cases. The NRC staff finds the plans for use of ASME Code Cases described in the RAI response to meet the applicable NRC regulations. Therefore, **RAI 05.02.01.02-2** is resolved.

Therefore, based on the above, the staff finds it acceptable for the applicant to incorporate by reference the ESBWR DCD.

5.2.1.2.5 *Post Combined License Activities*

There are no post COL activities related to this section.

5.2.1.2.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG–1966. The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information, and that there is no outstanding information expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix [X], Section VI.B.1, all nuclear safety issues relating to this section that were incorporated by reference have been resolved.

5.2.2 Overpressure Protection

This subsection of the FSAR addresses the safety and relief valves (SRVs) and the portion of the reactor protection system that ensures overpressure protection for the RCPB during operation at power.

Section 5.2, "Integrity of Reactor Coolant Pressure Boundary" of the Fermi 3 COL FSAR, Revision 3, incorporates by reference, Subsection 5.2.2, "Overpressure Protection," of the certified ESBWR DCD, Revision 9, referenced in 10 CFR Part 52, Appendix [X], with no departures or supplements. NRC staff reviewed the application and checked the referenced DCD to ensure no issues relating to this subsection remains for review.The staff's review confirmed that there is no outstanding issue related to this subsection. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix [X], Section VI.B.1, all nuclear safety issues relating to the overpressure protection have been resolved.

5.2.3 Reactor Coolant Pressure Boundary Materials

This subsection of the FSER addresses information related to the materials selection, fabrication, and processing of RCPB piping and components, as well as the compatibility of RCPB materials with the reactor coolant.

Section 5.2, "Integrity of Reactor Coolant Pressure Boundary" of the Fermi 3 COL FSAR, Revision 3, incorporates by reference Subsection 5.2.3, "Reactor Coolant Pressure Boundary Materials," of the certified ESBWR DCD, Revision 9, Referenced in 10 CFR Part 52, Appendix [X], with no departures or supplements. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this subsection remains for review.[1] The staff's review confirmed that there is no outstanding issue related to this subsection. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix [X], Section VI.B.1, all nuclear safety issues relating to the RCPB materials have been resolved.

5.2.4 Preservice and Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

5.2.4.1 *Introduction*

This subsection of the FSAR discusses components that are part of the RCPB, which must be designed to permit periodic inspection and testing of important areas and features to assess their structural and leak tight integrity. ISI programs are based on the requirements of 10 CFR 50.55a, "Codes and Standards," in that Code Class 1 components, as defined in Section III of the ASME BPV Code, meet the applicable inspection requirements set forth in Section XI of the ASME Code, "Rules for Inservice Inspection of Nuclear Power Plant Components."

5.2.4.2 *Summary of Application*

Section 5.2 of the Fermi 3 COL FSAR, Revision 3, incorporates by reference Section 5.2 of the certified ESBWR DCD, Revision 9. In addition, in FSAR Subsection 5.2.4 and Subsection 5.2.6 (COL Items), the applicant provides the following information:

<u>*COL Items*</u>

STD COL 5.2-1-A

In FSAR Subsection 5.2.4, "Preservice and Inservice Inspection and Testing of Reactor Coolant Pressure Boundary," the applicant states that:

- All Class 1 austenitic or dissimilar metal welds are included in the referenced certified design.

The initial ISI Program incorporates the latest edition and addenda of the ASME BPV Code approved in 10 CFR 50.55a(b) on the date 12 months before initial fuel loading. Additionally, the applicant provides information to address Class 1 austenitic or dissimilar metal welds and preservation of accessibility during construction to enable the performance of ISI examinations during the operational phase.

In FSAR Subsection 5.2.4.3.4, "Qualification of Personnel and Examination Systems for Ultrasonic Examination," the applicant states that:

Certification of NDE personnel shall be in accordance with ASME Section XI, IWA-2300, as modified by 10 CFR 50.55a(b)(2)(xviii).

- STD COL 5.2-3-A

In FSAR Subsection 5.2.4.2, "Accessibility," the applicant states:

> During the construction phase of the project, anomalies and construction issues are addressed using change control procedures that require that changes to approved design documents including field changes and modifications are subject to the same review and approval process as the original design.

> Control of accessibility for inspectability and testing during licensee design activities affection Class 1 components is provided via procedures for design control and plant modifications.
>
> Ultrasonic techniques (UT) will be the preferred NDE method for all PSI and ISI volumetric examinations; radiographic techniques (RT) will be used as a last resort only if UT cannot achieve the necessary coverage. The same NDE method used during PSI will be used for ISI to the extent possible to assure a baseline point of reference. If a different NDE method is used for ISI than was used for PSI, equivalent coverage will be achieved as required by code.

- STD SUP 5.2-1

In FSAR Subsection 5.2.4.6, "System Leakage and Hydrostatic Pressure Tests," the applicant states that:

> Regardless of which test method is chosen, system leakage and hydrostatic pressure tests will meet all requirements of ASME Code Section XI, IWA-5000 and IWB-5000 for Class I components, including the limitation of 10 CFR 50.55a(b)(2)(xxvi).
>
> System pressure tests and correlated technical specification requirements are provided in the plant Technical Specifications 3.4.4, "RCS Pressure and Temperature (P/T) Limits," and 3.10.1, "Inservice Leak and Hydrostatic Testing Operation."

In FSAR Subsection 5.2.4.11, "COL Information for Preservice and Inservice Inspection and Testing of Reactor Coolant Pressure Boundary," the applicant states that:

> DCD Subsection 5.2.4 fully describes the Preservice and Inservice Inspection and Testing Programs for the RCPB. The implementation milestones for the Preservice and Inservice Inspection and Testing Programs are provided in Section 13.4.

5.2.4.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1966, the FSER related to the certified ESBWR DCD.

In addition, the relevant requirements of the Commission regulations for the compliance with the 10 CFR 50.55a, and the associated acceptance criteria, are in Subsection 5.2 of NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," (the Standard Review Plan [SRP]).

In particular, NRC regulations in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and Part 52 provide the regulatory basis for NRC staff review of the information provided in the Fermi COL applications. For example, NRC regulations in 10 CFR Part 50, Appendix A, "General Design Criterion for Nuclear Power Plants," General Design Criterion (GDC) 1, "Quality standards and records," require that nuclear power plant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Further, NRC regulations in 10 CFR 50.55a, as related to the establishment of the minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of nuclear power plant components, require conformance with appropriate editions of published industry codes and standards.

Also, NRC staff followed the guidance in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," June 2007, in evaluating Fermi 3 COL FSAR, Subsection 5.2.1.1 for compliance with NRC regulations.

The regulatory basis for acceptance of the supplemental COL information items are established in GDC 32 found in Appendix A to 10 CFR Part 50, as it relates to periodic inspection and testing of the RCPB; and 10 CFR 50.55a, as it relates to the requirements for testing and inspecting of the Code Class 1 components as specified in Section XI of the ASME BPV Code. In addition, SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," provides Commission policy for fully describing an operational program.

5.2.4.4 *Technical Evaluation*

As documented in NUREG–1966, NRC staff reviewed and approved Subsection 5.2.4 of the certified ESBWR DCD. The staff reviewed Subsection 5.2.4 of the Fermi 3 COL FSAR, Revision 3, and checked the referenced ESBWR DCD to ensure that the combination of the information in the ESBWR DCD and the information in the COL FSAR appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information contained in the application and the information incorporated by reference address the relevant information related to this section.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER with open items issued for the North Anna application were equally applicable to the Fermi COL application, the staff undertook the following reviews:

- The staff compared the North Anna 3 COL FSAR, Revision 1, to the Fermi COL FSAR. In performing this comparison, the staff considered changes made to the Fermi COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs and open and confirmatory items identified in the North Anna SER with open items.

- The staff confirmed that the applicant endorsed all responses to RAIs identified in the corresponding standard content (the North Anna SER) evaluation.

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[1] See "*Finality of Referenced NRC Approvals*," in SER Section 1.2.2, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the North Anna standard content to be directly applicable to the Fermi 3 COL application. This standard content material is identified in this SER by use of italicized, double indented formatting.

The staff reviewed the information in the Fermi 3 COL FSAR as follows:

<u>*COL Items*</u>

- *STD COL 5.2-1-A Preservice and Inservice Inspection Program Description*

  In Section 5.2.4, the applicant stated that "the initial inservice inspection program incorporates the latest edition and addenda of the ASME BPV Code approved in 10 CFR 50.55a(b) on the date 12 months before initial fuel load." 10 CFR 50.55a(g)(4)(i) states that inservice examinations and pressure tests conducted during the initial 120-month inspection interval must comply with the requirements in the latest edition and addenda of the Code (or Code Cases) incorporated by reference in paragraph (b) of this section on the date 12 months before the date scheduled for initial loading of fuel under a COL under 10 CFR Part 52 of this chapter subject to the limitations and modifications listed in paragraph (b) of this section. The staff finds that the information provided by the applicant meets 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(b). Therefore, the FSAR information is acceptable.

  In Section 5.2.4.3.4, the applicant stated that "certification of NDE personnel shall be in accordance with ASME Section XI, IWA-2300, as modified by 10 CFR 50.55a(b)(2)(xviii)." The applicant also stated in Section 5.2.4.6 that "system leakage and hydrostatic pressure tests will meet all the requirements of ASME Code Section XI, IWA-5000 and IWB-5000 for Class 1 components, including the limitation of 10 CFR 50.55a(b)(2)(xxvi)." 10 CFR 50.55a(b)(2) imposes certain limitations or modifications on the use of this Code. The FSAR information is consistent with the requirements of the ASME Code, Section XI for the certification of NDE personnel and pressure testing of Class 1 components. In addition, the information reflects the appropriate regulations which modify the ASME Code for certification of NDE personnel and pressure testing. Since the FSAR information is consistent with ASME and the regulations, the FSAR information is acceptable.

  In Section 5.2.4.11, the applicant stated that Section 5.2.4 of the ESBWR DCD "fully describes the Preservice and Inservice Inspection and Testing Programs for the RCPB and that the implementation milestones for the Preservice and Inservice Inspection and Testing Programs are provided in FSAR Section 13.4." Since the PSI program uses essentially the same elements of the ISI program and the PSI program requirements are stated under ASME Section XI, the staff concurs with the statement that the PSI/ISI programs are fully described. The staff reviewed Table 13.4 and found that the implementation milestones for the PSI/ISI operational programs are listed. The staff concludes that the information provided under STD COL 5.2-1-A and the supplemental information is acceptable.

- *STD COL 5.2-3-A Preservice and Inservice Inspection NDE Accessibility Plan Description*

  The applicant stated in Section 5.2.4 that all Class 1 austenitic or dissimilar metal welds are included in the referenced certified design. The applicant described in Section 5.2.4.2 how anomalies and construction issues are addressed using change control procedures during the construction phase of the project. Procedures require that changes to approved design documents, including field changes and modifications, are subject to the same review and approval process as the original design. Control of accessibility for inspect ability and testing during licensee design activities affecting Class 1 components is provided via procedures for design control and plant modifications. The applicant explained that ultrasonic techniques (UT) will be the preferred NDE method for all PSI and ISI volumetric examinations; radiographic techniques (RT) will be used as a last resort only if UT cannot achieve the necessary coverage. The same NDE method used during PSI will be used for ISI to the extent possible to assure a baseline point of reference. If a different NDE method is used for ISI than was used for PSI, equivalent coverage will be achieved as required by the Code.

  During normal plant operation, ultrasonic examination is the desired NDE method for austenitic and dissimilar metal welds due to ease in obtaining examination coverage of piping that is filled with water and as low as reasonably achievable considerations. The use of RT is an acceptable replacement for UT and is allowed under ASME Section XI, Table IWB-2500, since the examination technique specified for these welds is volumetric. The information provided by the applicant meets the requirements under 10 CFR 50.55a(g)(3), which requires that plants be designed to enable the performance of inservice examinations. The use of RT as a supplemental examination technique with 100 percent coverage meets the requirements of ASME Section XI, Table IWB-2500. The information provided by the applicant provides reasonable assurance that during construction, controls exist to maintain the accessibility to enable the performance of inservice examinations for austenitic and dissimilar metal welds. The information provided by the applicant meets 10 CFR 50.55a(g)(3) and ASME Section XI. The FSAR information is, therefore, acceptable.

The applicant identified the following commitments to track implementation of the PSI/ISI programs:

1. ISI - Implemented prior to commercial service (COM 13.4-024)
2. PSI - Completion prior to initial plant startup (COM 13.4-026)

The staff concludes that Commitments 13.4-024 and 13.4-026 shall be License Conditions.

<u>*Supplemental Information*</u>

- *STD SUP 5.2-1*

  Under Section 5.2.4.6, the applicant stated that system pressure tests and correlated technical specification requirements are provided in the plant TSs 3.4.4, "RCS P/T

Limits," and 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." The proposed change provides additional information with respect to system pressure testing that is located within the TS.

Since the location of additional information regarding pressure testing is at the discretion of the licensee, and, the proposed change under STD COL 5.2-1-A (discussed above) meets the ASME Code and the limitations under 10 CFR 50.55a(b)(2)(xxvi), the staff concludes that the supplemental information as it pertains to pressure testing is acceptable.

5.2.4.5 *Post Combined License Activities*

The following are license conditions regarding the PSI/ISI programs:

1. ISI - Implemented prior to commercial service (COM 13.4-024)

2. PSI - Completion prior to initial plant startup (COM 13.4-026)

Additionally, the staff concludes that Commitments 13.4-024 and 13.4-026 shall be License Conditions.

5.2.4.6 *Conclusions*

The NRC staff's finding related to information incorporated by reference is in NUREG–1966. The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information, and that there is no outstanding information expected to be addressed in the Fermi 3 COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix [X], Section VI.B.1, all nuclear safety issues relating to this section that were incorporated by reference have been resolved.

In addition, the staff concludes that the information provided in Fermi 3 COL FSAR, Revision 3, Subsection 5.2.4, meets the relevant guidelines in SRP Subsection 5.2.4 and RG 1.206, Subsection C.III.1, Chapter 5, C.I.5.2.4, and is thus, acceptable. The staff further concludes that the Fermi 3 COL FSAR, PSI/ISI programs and implementation milestones are consistent with the policy established in SECY-05-0197. Conformance with these guidelines and policy provides an acceptable basis for satisfying in part, the requirements of GDC 32 and 10 CFR 50.55a.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

5.2.5.1 *Introduction*

This subsection of the Fermi 3 COL FSAR, Revision 3, discusses the RCPB leakage detection systems which are designed to detect and, to the extent practical, identify the source of reactor coolant leakage.

5.2.5.2 *Summary of Application*

Section 5.2 of the Fermi 3 COL FSAR, Revision 3, incorporates by reference Section 5.2 of the ESBWR DCD, Revision 9. In addition, in FSAR Subsection 5.2.5 and Subsection 5.2.6 the applicant provides the following:

<u>COL Item</u>

- STD COL 5.2-2-A Leak Detection Monitoring

The applicant provided additional information to address STD COL 5.2-2-A. The applicant replaced Subsection 5.2.5.9, “Leak Detection Monitoring” of the ESBWR DCD with a paragraph that states that operators are provided with procedures and information for detecting, monitoring, recording, trending and determining the sources of reactor coolant pressure boundary leakage. The applicant added that FSAR Section 13.5, “Plant Procedures” provides a description of the plant procedures program and implementation milestones.

5.2.5.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1966, the FSER related to the certified ESBWR DCD.

Also, NRC staff followed the guidance in RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” June 2007, in evaluating Fermi 3 COL FSAR, Subsection 5.2.5 for compliance with NRC regulations.

In addition, the relevant requirements of the Commission regulations for the compliance with the 10 CFR 50.55a, and the associated acceptance criteria, are in Subsection 5.2.5 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition).” Staff acceptance of the leakage detection design is based on its meeting the requirements of the following criteria:

- GDC 2, “Design Basis for Protection Against Natural Phenomena,” as it relates to the capability of the design to maintain and perform its safety function following an earthquake.

- GDC 30, “Quality of Reactor Coolant Pressure Boundary,” as it relates to the detection, identification, and monitoring of the source of reactor coolant leakage.

5.2.5.4 *Technical Evaluation*

As documented in NUREG–1966, NRC staff reviewed and approved Subsection 5.2.5 of the certified ESBWR DCD. The staff reviewed Subsection 5.2.5 of the Fermi 3 COL FSAR, Revision 3, and checked the referenced ESBWR DCD to ensure that the combination of the information in the ESBWR DCD and the information in the Fermi 3 COL FSAR appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information contained in the application and the information incorporated by reference addresses the relevant information related to this section.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER with open items issued for the North Anna application were equally applicable to the Fermi COL application, the staff undertook the following reviews:

- The staff compared the North Anna 3 COL FSAR, Revision 1, to the Fermi 3 COL FSAR. In performing this comparison, the staff considered changes made to the Fermi COL 3 FSAR (and other parts of the COL application, as applicable) resulting from requests for RAIs and open and confirmatory items identified in the North Anna 3 SER with open items.

- The staff confirmed that the applicant endorsed all responses to RAIs identified in the corresponding standard content (the North Anna SER) evaluation.

- The staff verified that the site-specific differences were not relevant to this section.

<u>*COL Item*</u>

In the ESBWR DCD, Revision 9, the following STD COL Item is numbered as STD COL 5.2-2-A.

The following portion of this technical evaluation section is reproduced from Subsection 5.2.5 of North Anna Unit 3 SER (ML091730304):*STD COL 5.2-2-H Leak Detection Monitoring*

> *NRC staff identified that the substitution of Section 5.2.5.9 of the ESBWR DCD with STD COL 5.2-2-H text appears to inappropriately limit the intended scope of the procedures contained in Section 5.2.5.9 of the ESBWR DCD. In addition, inclusion in FSAR, Revision 0 of the STD COL 5.2-2-H text of the examples "sump pump run time, sump level, and condensate transfer rate" without inclusion of "radioactivity," also appears to inappropriately limit the scope of the procedures. In* ***RAI 05.02.05-1****, the staff requested the applicant to clarify the following:*
>
> *(a) Revise the FSAR to clarify the scope of procedures relative to TSs. In addition to establishing the leakage rates for the limits in the TS, the operators should be able to use the procedures to identify and monitor the unidentified leakage at a level much lower than the TS limit so that the operator can monitor leakage, evaluate trends, determine the source of leakage, and evaluate potential corrective actions. This level to provide operators an early alert to initiate actions prior to the TS limit should be*

[1] See "*Finality of Referenced NRC Approvals*," in SER Section 1.2.2, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

established as an alarm. The alarm level being established in an approved revision of the ESBWR DCD, Section 5.2.5 is acceptable for the COL application.

(b) *Confirm the procedure scope addresses the conversion of different parameter indications to include all three detection instrumentation in TS Limiting Condition for Operation 3.3.4.1, and clarify STD COL 5.2.2-H accordingly. The procedures should include indications from 1) the drywell floor drain high conductivity water sump monitoring system, 2) drywell air coolers condensate flow monitoring system, and 3) drywell fission product monitoring system.*

In the letter, dated August 8, 2008, the applicant responded to ***RAI 05.02.05-1****. In the response, the applicant revised FSAR Section 5.2.5.9 and STD COL 5.2.2-H to clarify that the procedures will fully address the topics described in Items (a) and (b) of the* ***RAI*** *and will be consistent with Section 5.2.5 of the ESBWR DCD, Revision 5. The revised FSAR Section 5.2.5.9 and STD COL 5.2.2-H states as follows:*

"Operators are provided with procedures for detecting, monitoring, recording, trending, and determining the sources of RCPB leakage. Examples of parameters that are monitored are sump pump run time, sump level, condensate transfer rate, and process chemistry/radioactivity.

The procedures are used for converting different parameter indications for identified and unidentified leakage into common leak rate equivalents (volumetric or mass flow) and leak rate rate-of-change values, including indications from: 1)the drywell floor drain high conductivity water sump monitoring system, 2) the drywell air coolers condensate flow monitoring system, and 3) the drywell fission product monitoring system.

The procedures are used to monitor leakage at levels well below Technical Specifications limits and provide guidance for evaluating potential corrective action plans to prevent the plant from exceeding a Technical Specifications limit.

An unidentified leakage rate-of-change alarm provides an early alert to the operators to initiate corrective actions prior to reaching a Technical Specifications limit."

NRC staff reviewed the applicant's response to the above ***RAI****. The staff found that the response addresses all the concerns identified in the* ***RAI,*** *and that the applicant committed to be consistent with ESBWR DCD, Tier 2, Revision 5, Section 5.2.5. DCD Revision 5, Section 5.2.5 includes an alarm that annunciates if a step increase in the unidentified leak rate occurs ("reference DCD Section 5.2.5.4, Limits for Reactor Coolant Leakage Rates within the Drywell.") The standard design and procedures will enable the operators to monitor leakage at levels well below TS limits, and initiate actions to prevent the plant from exceeding a TS limit. Based* *on the above, the staff finds **RAI 05.02.05-1** resolved and the staff confirmed the appropriate information is provided in FSAR Revision 1.*

The applicant identified the following commitment to track implementation of the operating and emergency operating procedures:

1. Operating procedures are developed at least six months prior to fuel load to allow sufficient time for plant staff familiarization and to allow NRC staff adequate time to review the procedures and to develop operator licensing examinations. (COM 13.5-002)

The staff concludes that the information above, meets the relevant guidelines in SRP Section 5.2.5 and RG 1.206, Section C.III.1, Chapter 5, C.I.5.2.5, and is thus acceptable. Conformance with these guidelines, GDC 2, "Design Basis for Protection Against Natural Phenomena," and GDC 30, "Quality of Reactor Coolant Pressure Boundary," provide an acceptable basis for satisfying the requirements.

5.2.5.5 *Post Combined License Activities*

The applicant identified the following commitment to track implementation of the operating and emergency operating procedures:

1. Operating procedures are developed at least six months prior to fuel load to allow sufficient time for plant staff familiarization and to allow NRC staff adequate time to review the procedures and to develop operator licensing examinations. (COM 13.5-002)

5.2.5.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG–1966. NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information, and no outstanding information is expected to be addressed in the Fermi 3 COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix [X], Section VI.B.1, all nuclear safety issues relating to this section that were incorporated by reference have been resolved.

In addition, the staff compared the additional Fermi 3 COL supplemental information in the application to the relevant NRC regulations, the guidance in Subsection 5.2.5 of NUREG–0800, and other NRC RGs. The staff's review concluded that the applicant has presented adequate information in the Fermi 3 COL FSAR to meet the requirements of GDC 2 and 30

5.3 <u>Reactor Vessel</u>

5.3.1 Reactor Vessel Materials

5.3.1.1 *Introduction*

This subsection of the Fermi 3 COL FSAR addresses the reactor vessel material specifications, including weld materials, special processes used for manufacture and fabrication of components, special methods for NDE, special controls and special processes used for ferritic steels and austenitic stainless steels, fracture toughness, reactor vessel materials surveillance program (RVSP), and reactor vessel fasteners.

5.3.1.2 *Summary of Application*

Section 5.3, "Reactor Vessel" of the Fermi 3 COLA FSAR incorporates by reference Section 5.3 of the ESBWR DCD, Revision 9. In addition, the applicant provides the following:

COL Items

- STD COL 5.3-2-A

  The description of the reactor vessel material surveillance program is provided in DCD Subsection 5.3.1.6, and is supplemented as follows:

  A complete reactor vessel material surveillance program will be developed as described above in accordance with the implementation schedule provided in Section 13.4.

- STD COL 16.0-1-A 5.6.4-1

  In FSAR Revision 3, Section 5.3, the applicant provides supplemental information related to subsection 5.3.1.5 "Fracture Toughness Compliance with 10 CFR Part 50, Appendix G", that states:

  > The pressure-temperature limit curves are developed in accordance with the Pressure and Temperature Limits Report, as discussed in the Technical Specifications Subsection 5.6.4 [START COM 5.03-002] Prior to fuel load, the pressure-temperature limit curves will be updated to reflect plant-specific material properties, if required. [END COM 5.03-002].

The staff's evaluation of this COL Item is in subsection 5.3.2 of this SER.

Supplemental Information

- STD SUP 5.3-1

In FSAR Revision 3, Section 5.3, the applicant provides supplemental information related to subsection 5.3.3.6 "Operating Conditions", that states:

> Development of plant operating procedures is addressed in Section 13.5. These procedures require compliance with the Technical Specifications. The Technical Specifications (which are developed by the methodology also identified in the Technical Specifications) are intended to ensure that the P-T limits identified in DCD Subsection 5.3.2 are not exceeded during normal operating conditions and anticipated plant transients.

5.3.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1966, the FSER related to the certified ESBWR DCD.

In addition, the relevant requirements of the Commission regulations for the compliance with 10 CFR 50.55a, and the associated acceptance criteria, are described in Subsection 5.3.1 of NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)."

Also, NRC staff followed the guidance in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," June 2007, in evaluating Fermi 3 COL FSAR, Subsection 5.3.1 for compliance with NRC regulations.

In addition, the regulatory basis for the acceptance of the RVSP Information is established in:

- GDC 32 found in Appendix A to 10 CFR Part 50, as it relates to the RVSP
- 10 CFR 50.60, as it relates to compliance with the requirements of 10 CFR Part 50, Appendix G
- 10 CFR Part 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness
- 10 CFR Part 50, Appendix H, as it relates to the RVSP
- 10 CFR 50.55a, as it relates to the requirements for testing and inspecting Code Class 1 components of the RCPB as specified in Section XI of the ASME Code
- SECY-05-0197, as it relates to fully describing an operational program

In addition, the relevant requirements of the Commission regulations for the compliance with 10 CFR 50.55a, and the associated acceptance criteria, are described in Subsection 5.3.1 of

NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)."

5.3.1.4 *Technical Evaluation*

As documented in NUREG–1966, NRC staff reviewed and approved Subsection 5.3.1 of the certified ESBWR DCD. The staff reviewed Subsection 5.3.1 of the Fermi 3 COL FSAR, Revision 3, and checked the referenced ESBWR DCD to ensure that the combination of the information in the ESBWR DCD and the information in the COL FSAR appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed

[1] See "*Finality of Referenced NRC Approvals*," in SER Section 1.2.2, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification

that the information contained in the application and the information incorporated by reference addresses the relevant information related to this section.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER with open items issued for the North Anna application were equally applicable to the Fermi COL application, the staff undertook the following reviews:

- The staff compared the North Anna 3 COL FSAR, Revision 1, to the Fermi COL FSAR. In performing this comparison, the staff considered changes made to the Fermi COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs and open and confirmatory items identified in the North Anna SER with open items.

- The staff confirmed that the applicant endorsed all responses to RAIs identified in the corresponding standard content (the North Anna SER) evaluation.

- The staff verified that the site-specific differences were not relevant.

<u>*COL Item*</u>

The following portion of this technical evaluation section is reproduced from Subsection 5.3.1 of North Anna Unit 3 SER (ML091730304):

- *STD COL 5.3-2A Materials and Surveillance Capsule*

NRC staff reviewed STD COL 5.3-2-A related to reactor vessel materials included under Section 5.3.1 of North Anna 3 COL FSAR, Revision 0. Specifically, the applicant provided STD COL 5.3-2-A as a substitution for Section 5.3.1.8 of the ESBWR DCD, Revision 5, in order to fully describe their RSVP and its implementation.

After reviewing the information provided in North Anna 3 COLA FSAR, Revision 0, Section 5.3.1, including the information referenced in the DCD Section 5.3.1, the staff found that the COL applicant had not met the minimum guidelines in RG 1.206 for a description of the RVSP and its implementation. The staff determined that more information was needed to fully describe the RVSP in accordance with SECY-05-0197 to reach a resolution for this COL item. Thus, the staff requested additional information in ***RAI 5.03.01-1*** *in order to complete this review.*

In ***RAI 5.03.01-1****, the staff requested that the applicant provide additional information on the preparation of the surveillance capsule specimens, the surveillance capsule locations, and the number and type of specimens in each capsule associated with the RVSP.*

In a letter dated September 3, 2008, the applicant responded to ***RAI 5.03.01-1*** *and described in detail the preparation of the capsule* *specimens, the number and type of specimens, and the location of the specimen capsules in the core beltline region, and applicant agreed to update their FSAR. The staff determined that the applicant's response appropriately addressed RAI 5.03.01-1. The staff reviewed Revision 1 to FSAR Section 5.3.1 and confirmed that the information described in the response to RAI 5.03.01-1 was included in the FSAR. Therefore the staff finds that the applicant has adequately addressed this issue and RAI 5.03.01-1 is resolved.*

The applicant stated that, a summary technical report, including test results, is submitted as specified in 10 CFR 50.4, for the contents of each capsule withdrawn, within one year of the date of capsule withdrawal unless an extension is granted by the Director, Office of Nuclear Reactor Regulation. The Fermi 3 COL FSAR includes a commitment that states:

> If the test results indicate a change in the Technical Specifications is required, the expected date for submittal of the revised Technical Specification will be provided with the report. (COM 5.3.001)

The applicant also identifies the following license condition:

The applicant identified in FSAR Section 13.4, Table 13.4-201, that the Reactor Vessel Material Surveillance Program, required by 10 CFR 50.60 and Appendix H to 10 CFR Part 50, is to be required by a license condition. FSAR Section 5.3.1.8 states that a complete reactor vessel material surveillance program will be developed prior to fuel load. The staff finds it acceptable to require the Reactor Vessel Material Surveillance Program by a license condition and will include such license condition in the Fermi COL.

The staff concludes that the information above, meets the relevant guidelines in SRP Section 5.3.1 and RG 1.206, Section C.III.1, Chapter 5, C.I.5.3.1, and is thus acceptable. Conformance with these guidelines, GDC 32 found in Appendix A to 10 CFR Part 50, as it relates to the RVSP, 10 CFR 50.60, 10 CFR Part 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness, 10 CFR Part 50, Appendix H, 10 CFR 50.55a, as it relates to the requirements for testing and inspecting Code Class 1 components of the RCPB as specified in Section XI of the ASME Code, 10 CFR Part 50, Appendix G, and SECY-05-0197, provide an acceptable basis for satisfying the requirements.

Supplemental Information:

- STD SUP 5.3-1

In FSAR Revision 3, Section 5.3, the applicant provides supplemental information related to subsection 5.3.3.6 "Operating Conditions", that states:

> Development of plant operating procedures is addressed in Section 13.5. These procedures require compliance with the Technical Specifications. The Technical Specifications (which are developed by the methodology also identified in the Technical Specifications) are intended to ensure that the P-T limits identified in DCD

Subsection 5.3.2 are not exceeded during normal operating conditions and anticipated plant transients.

In STD SUP 5.3-1, the COL applicant added information to FSAR Subsection 5.3.3.6, “Operating Conditions,” to state that the development of plant operating procedures is addressed in Section 13.5. These procedures require compliance with the technical specifications (TS). The TS (which are developed by the methodology also identified in the TS) are intended to ensure that the pressure and temperature (P-T) limits identified in DCD Subsection 5.3.2 are not exceeded during normal operating conditions and anticipated plant transients. The staff finds that STD SUP 5.3-1 is acceptable because it is in accordance with the recommendations of RG 1.206, Subsection C.I.5.3.2.2 which states that the FSAR should include a commitment that plant operating procedures will ensure that the P-T limits will not be exceeded during any foreseeable upset condition.

5.3.1.5 *Post Combined License Activities*

The applicant identifies the following license condition:

1. A complete reactor vessel material surveillance program will be developed prior to fuel load.

5.3.1.6 *Conclusion*

The NRC staff’s finding related to information incorporated by reference is in NUREG–1966. NRC staff reviewed the application and checked the referenced DCD. The NRC staff’s review confirmed that the applicant has addressed the required information, and no outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix [X], Section VI.B.1, all nuclear safety issues relating to this section that were incorporated by reference have been resolved.

The staff concludes that the applicant’s proposed resolution to the COL Item STD COL 5.3-2-A and information provided in STD SUP 5.3-1 meet the relevant acceptance criteria of SRP Subsection 5.3.1, and the guidance in RG 1.206 and is thus, acceptable. Conformance with GDC 32 provides an acceptable basis for satisfying the requirements of Appendices G and H to 10 CFR Part 50.

5.3.2.1.1 Pressure-Temperature Limits

5.3.2.1.2 Introduction

This subsection of the Fermi 3 COL FSAR, Revision 3, discusses P/T limits which are required as a means of protecting the reactor vessel during startup and shut down to minimize the possibility of fast fracture. The methods outlined in Appendix G of Section XI of the ASME Code are employed in the analysis of protection against non-ductile failure. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the adjusted reference temperature, which includes a reference nil ductility temperature (NDT) shift, initial RT_{NDT}, and margin.

5.3.2.2.1 Summary of Application

Section 5.3 of the Fermi 3 COL FSAR, Revision 3, incorporates by reference Section 5.3 of the ESBWR DCD, Revision 9 without any departures. In addition, in FSAR Subsection 5.3.1.5, the applicant provides the following:

<u>COL Item</u>

- STD COL 16.0-1-A 5.6.4-1 Pressure-Temperature Limit Curves

The applicant stated that the P/T limit curves are developed in accordance with the pressure-temperature limiting report (PTLR) as discussed in TS Subsection 5.6.4.

5.3.2.2.2 Regulatory Basis

The regulatory basis of the information incorporated by reference is in NUREG–1966, the FSER related to the certified ESBWR DCD.

In addition, the relevant requirements of the Commission regulations for the compliance with the 10 CFR 50.55a, and the associated acceptance criteria, are described in Section 5.3 of NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)."

In particular, NRC regulations in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and Part 52 provide the regulatory basis for NRC staff review of the information provided in the Fermi 3 COL applications. For example, NRC regulations in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 1, "Quality standards and records," require that nuclear power plant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Further, NRC regulations in 10 CFR 50.55a, as related to the establishment of the minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of nuclear power plant components, require conformance with appropriate editions of published industry codes and standards.

Also, NRC staff followed the guidance in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," June 2007, in evaluating Fermi 3 COL FSAR Subsection 5.3.2 for compliance with NRC regulations.

Appendix G to 10 CFR Part 50 provides the regulatory basis regarding P/T limits.

5.3.2.2.3 Technical Evaluation

As documented in NUREG–1966, NRC staff reviewed and approved Subsection 5.3.2 of the certified ESBWR DCD. The staff reviewed Subsection 5.3.2 of the Fermi 3 COL FSAR, Revision 3, and checked the referenced ESBWR DCD to ensure that the combination of the information in the ESBWR DCD and the information in the COL FSAR appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information contained in the application and the information incorporated by reference addresses the relevant information related to this subsection.

By a letter dated March 3, 2011 (ML1106700900), Detroit Edison submitted reports NEDO-33441 and NEDC-33441P, "GE Hitachi Nuclear Energy Methodology for the Development of Economic Simplified Boiling Water Reactor (ESBWR) Reactor Pressure Vessel Pressure-Temperature Curves," Revision 5. These reports were prepared by GE-Hitachi (GEH) and submitted in support of the Fermi 3, Reference Combined License (R-COL) Application to address an ESBWR DCD COL Item, which states that the COL applicant, in accordance with ESBWR TS Chapter 16, Section 5.6.4, will furnish bounding P-T curves either as part of the TS or as part of a PTLR submittal for NRC review. As such, the purpose of this report is to provide the bounding P-T limits and the associated methodology for the development of the PTLR using the criteria of Generic Letter (GL) 96-03. The first part of the staff's review was to ensure that the information provided in the proposed PTLR and the revised TS pages are in accordance with the guidance in GL 96-03, "Relocation of Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits." The second part of the staff's review was to verify that the proposed P-T limits have been developed appropriately using the methodology provided in GE Report NEDC-33441P, Revision 5 (hereafter referred to as the ESBWR PTLR).

NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the acceptability of a facility's proposed PTLR based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; Appendix H to 10 CFR Part 50; RG 1.99, Revision 2 GL 92-01, Revision 1; GL 92-01; Revision 1, Supplement 1; SRP Section 5.3.2; and GL 96-03. Appendix G to 10 CFR Part 50 requires that facility P-T limits for the RPV be at least as conservative as those obtained by applying the linear elastic fracture mechanics methodology of Appendix G to Section XI of the ASME Code. Appendix H to 10 CFR Part 50 establishes requirements related to facility RPV material surveillance programs. RG 1.99, Revision 2 contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation. GL 92-01, Revision 1 requested that licensees submit the RPV data for their plants to the staff for review, and GL 92-01, Revision 1, Supplement 1 requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. SRP Section 5.3.2 provides an acceptable method for determining the P-T limits for ferritic materials in the beltline of the RPV based on the ASME Code, Section XI, Appendix G methodology.

The most recent version of Appendix G to Section XI of the ASME Code which has been endorsed in 10 CFR 50.55a, and therefore, by reference in 10 CFR Part 50, Appendix G, is the 2007 Edition through the 2008 Addenda of the ASME Code. The P-T limit methodology based on this edition of Appendix G to Section XI of the ASME Code (the ASME Code, Section XI, Appendix G methodology) incorporates the provisions of ASME Code Cases N-588 and N-640.

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1 See "*Finality of Referenced NRC Approvals*," in SER Section 1.2.2, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

Additionally, Appendix G to 10 CFR Part 50 imposes minimum head flange temperatures when system pressure is at or above 20 percent of the preservice hydrostatic test pressure.

GL 96-03 addresses the technical information necessary for a licensee's implementation of a PTLR. GL 96-03 establishes the information, which must be included in: (1) an acceptable PTLR methodology (with the P-T limit methodology as its subset) and, (2) the PTLR itself. TSTF-419 provides additional guidance, which provides an alternative format for documenting the implementation of a PTLR in the "Administrative Controls" section of a facility's TS.

The Fermi 3 COL TSs contain all of the necessary provisions required for the implementation and control of a PTLR. The Fermi 3 TSs are provided in Part 4 of the R-COL application. The relevant TS requirements include the TS definition of the PTLR (TS Section 1.1); the TS limiting conditions of operation (LCO) for the reactor coolant system P-T limits (LCO 3.4.4), including LCO Action Statements, Surveillance Requirements, and related applicability criteria; and the necessary administrative controls governing the PTLR content and reporting requirements (TS 5.6.4). All of the TS pages related to the implementation and control of a PTLR are acceptable to the staff.

The staff has determined that the contents of the PTLR conform to the staff's technical criteria for PTLRs, as defined in Attachment 1 of GL 96-03. The staff has also determined that the PTLR has satisfied the requirements of 10 CFR Part 50, Appendix G. Furthermore, the staff has determined that the PTLR is compatible with the TSs and the PTLR-related TS provisions meet the technical criteria of GL 96-03. The staff notes that the PTLR provides generic, not plant-specific, heatup and cooldown P-T curves based on bounding material properties and projected fluence. To address the submittal of plant specific P-T limits, the COL applicant has provided the following commitment:

1. Prior to fuel load, the pressure-temperature limit curves will be updated to reflect plant-specific material properties, if required. (COM 5.03-002)

The staff finds that this approach is consistent with the guidelines of GL 96-03 and is therefore acceptable. Based on this evaluation, the staff concludes that the PTLR is acceptable for establishing limiting P-T limit curves and related input parameters. Finally, per GL 96-03, any subsequent changes in the methodology used to develop the P-T limits must be approved by the NRC.

As discussed in Section 1.0 of the PTLR, this report describes the methodology used to develop the P-T limits and provides specific P-T curves for the reactor vessel. Accordingly, the PLTR utilizes generic inputs for reactor vessel (RV) beltline material chemistry, initial nil-ductility reference temperature (RT_{NDT}) values, and projected neutron fluence, to determine the P-T limit curves. These generic inputs are intended to be bounding for the design and represent the maximum allowable limits on the input parameters. Therefore, these generic inputs will be substantiated for use in the Fermi 3 COL PTLR in order to verify that actual plant-specific RV beltline properties remain bounded by the generic inputs contained in the PTLR.

Attachment 1 of GL 96-03 contains seven technical criteria (PTLR Criteria) that the contents of PTLRs should conform to if P-T limits are to be located in a PTLR. The staff's evaluations of the contents of the ESBWR PTLR against the seven criteria in Attachment 1 of GL 96-03 are given in the subsections that follow.

PTLR Criterion 1 states that the PTLR contents should include the neutron fluence values that are used in the calculations of the adjusted reference temperature (ART) values for the P-T limit calculations. Accurate and reliable neutron fluence values are required in order to satisfy the provisions GDC 14, 30, and 31 of 10 CFR Part 50, Appendix A, as well as the specific fracture toughness requirements of 10 CFR Part 50, Appendix G. ESBWR PTLR Section 3.3, "Predicted Fluence," states that the fluence analysis for the ESBWR is based on the NRC-approved methodology provided in GE Licensing Topical Report NEDC-32983P-A, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations." In addition, the applicant provided the peak RV neutron fluence values projected to 60 years of facility operation in Section 3.3 of the ESBWR PTLR. The staff determined that these 60-year neutron fluence values were calculated using an NRC-approved methodology that is consistent with the guidelines in RG 1.190. The inclusion of valid peak RV neutron fluence values calculated using a neutron fluence methodology that is in conformance with RG 1.190 fulfills the provisions of PTLR Criterion 1. Therefore, the staff determined that PTLR Criterion 1 is satisfied.

10 CFR Part 50, Appendix H provides the staff's requirements for designing and implementing RV material surveillance programs. The rule requires that RV material surveillance programs for operating reactors comply with the specifications of American Society for Testing and Materials (ASTM) Standard Procedure E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." The rule requires that the program design and the surveillance capsule withdrawal schedules for the programs must meet the edition of E 185 that is current on the issue date of the ASME Code to which the RV was purchased, although the rule permits more recent versions of E 185 to be used, up through the 1982 version.

To ensure conformance with these requirements, PTLR Criterion 2 states that the PTLR should either provide the RV surveillance capsule withdrawal schedule or provide references, by title and number, for the documents containing the RV surveillance capsule withdrawal schedule. The criterion also states that the PTLR should reference, by title and number, any applicable surveillance capsule reports that have been placed on the docket by the licensee requesting approval of the PTLR for its units. This criterion assures that the adjusted RT_{NDT} (ART) calculations will appropriately follow the RV material surveillance program requirements of 10 CFR Part 50, Appendix H. A discussion of the RV material surveillance program is provided in Section 7.0 of the PTLR. Section 7.0 states that the material surveillance program complies with Appendix H to 10 CFR Part 50 and ASTM E 185-82. The surveillance program description states that four capsules are provided to consider the 60-year design life of the vessel. This exceeds the three capsules specified in ASTM 185-82, since the predicted transition temperature shift is less than 55.6°C (100°F) at the inside of the vessel. The capsule withdrawal schedule is also provided in this section, and it is stated that each surveillance capsule will be tested in accordance to 10 CFR Part 50, Appendix H. The applicant also states that the results of the material surveillance program will be used to verify the ΔRT_{NDT} values in accordance with RG 1.99, Revision 2, and the P-T limits will be adjusted, as necessary, based on these results. The staff reviewed the recommended surveillance capsule withdrawal schedule and determined that it is in accordance with the specifications of ASTM E 185-82. On this basis, the staff determined that the provisions of PTLR Criterion 2 are satisfied.

PTLR Criterion 3 states that the Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs) developed using NRC-approved methodologies may be included in the PTLR. This criterion is not applicable to the ESBWR design, and thus not applicable to the Fermi 3 COL.

10 CFR Part 50, Appendix G requires that the P-T limits for operating reactors be generated using a method that accounts for the effects of neutron embrittlement on the fracture toughness of RV beltline materials. For P-T limits, the effects of neutron embrittlement on the fracture toughness of RV beltline materials is defined in terms of the shift in the RT_{NDT} values resulting from neutron irradiation over a given period of facility operation. The final ART value for a material resulting from neutron embrittlement over a certain period of facility operation is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the shift in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin term. NRC RG 1.99, Revision 2, provides the staff's recommended methodologies for calculating ART values used for P-T limit calculations. ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Revision 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Revision 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence, and the calculation procedures. Appendix G to Section XI of the ASME Code requires that licensees determine the ART at the 1/4T and 3/4T locations.

To ensure compliance with the requirements of 10 CFR Part 50, Appendix G, PTLR Criterion 4 states that PTLR contents should identify the limiting materials and limiting ART values at the 1/4T and 3/4T locations in the wall of the RV. The ART values and all inputs for the ART calculations, including RV beltline material chemistry values, initial RT_{NDT} values (Table 3-1), and peak RV beltline neutron fluence projections at 60-years are provided in Section 3 of the PTLR. In PTLR Section 3.4, the applicant describes how the procedures outlined in RG 1.99, Revision 2, are applied to determine the ΔRT_{NDT} and ART values. In this section, the applicant states that the nominal irradiation temperature in the beltline region is less than 525 ^{o}F. The staff notes that for the procedures of this RG to be valid for nominal irradiation temperatures less than 525^{o}F, a correction factor shall be used to compensate for greater embrittlement. To address this issue, the applicant proposed to utilize a correction factor equal to a 1^{o}F increase in ΔRT_{NDT} for each 1^{o}F decrease in irradiation temperature below 550^{o}F. This method will be validated for Fermi 3 using the results of the materials surveillance capsule program. The staff determined that this approach is acceptable because it provides a conservative estimate of the additional effect of irradiation on the beltline region at lower temperatures and that the applicant will verify the applicability of the assumption upon receipt of the surveillance capsule data.

The ART calculations and margin term values for the RV beltline materials are provided in Section 3.5. These values are determined for a 60-year design life. Based on the ART calculations, the applicant has identified the shell forging as limiting material to be used for the derivation of the P-T limits. To evaluate the proposed P-T limits for the RV, the staff confirmed the applicant's selection of the shell forging as the limiting beltline material and performed an independent calculation of the ART values provided in the report using the RG 1.99, Revision 2, methodology. It is noted that the applicant did not calculate the ART value at the 3/4T location, which is relevant to the heatup pressure-temperature limit calculation; because it is assumed that the ART value at 1/4T is bounding for heatup and cooldown.

Based on the evaluation described above, the staff found that the procedure used to calculate the ART values was consistent with the guidance of RG 1.99, Revision 2, and was, therefore, acceptable. Also, the PTLR clearly identifies the limiting materials and limiting ART values at the 1/4T location. Therefore, the staff determined that the provisions of PTLR Criterion 4 are satisfied.

Section IV.A.2 of 10 CFR Part 50, Appendix G requires that the P-T limits for operating reactors and the minimum temperatures established for the stressed regions of RVs (i.e., for the RV flange and stud assemblies) be met for all conditions. The rule also requires that the P-T limits for operating reactors must be at least as conservative as those that would be generated if the methods of analysis in the ASME Code, Section XI, Appendix G were used to generate the P-T limit curves. Table 1 of 10 CFR Part 50, Appendix G provides a summary of the required criteria for generating the P-T limits for operating reactors.

To ensure that PTLRs are in compliance with the above requirements, PTLR Criterion 5 states that the PTLR contents should provide the P-T limit curves for heatup and cooldown operations, core critical operations, and pressure testing conditions for operating light-water reactors. Table 4-2 of the PTLR includes P-T limit data for heatup and cooldown operations, core critical operations, and hydrostatic and pressure testing. The P-T Limit curves corresponding to these data points are provided in Figure 4-1 of the PTLR. In Section 5.0, the applicant also provided P-T Limit data and the corresponding curves for several non-beltline components including the closure head flanges and the main steam, feedwater, standby liquid control, and core DP nozzles. This meets the provisions of PTLR Criterion 5, which specifies that the PTLR include the P-T limit curves for reactor heatup, cooldown, critical operations, and pressure testing conditions.

The staff also performed independent analyses to verify the P-T limits curves for heat-up and cool-down operations, core critical operations, and hydrostatic pressure and leak testing provided in the PTLR. Based on this independent verification, the staff determined that the applicant's proposed P-T limits were developed in accordance with ASME Code, Section XI, Appendix G and therefore satisfy the requirements of 10 CFR Part 50, Appendix G. Hence, the applicant's proposed P-T limit curves are acceptable for operation of the RV.

Section IV.A.2 of 10 CFR Part 50, Appendix G requires that the P-T limits for operating reactors and the minimum temperature requirements for the highly stressed regions of the RVs (i.e., for the RV flange and stud assemblies) be met for all conditions. Table 1 of 10 CFR Part 50, Appendix G provides required the criteria for meeting the minimum temperature requirements for the highly stressed regions of the RV.

PTLR Criterion 6 states that the minimum temperature requirements of 10 CFR Part 50, Appendix G shall be incorporated into the P-T limit curves, and the PTLR shall identify minimum temperatures on the P-T limit curves such as the minimum boltup temperature and the hydrotest temperature. The staff determined that the curves were in compliance with the minimum temperature requirements of 10 CFR Part 50, Appendix G. Furthermore, the PTLR clearly identifies the minimum boltup temperature and hydrotest temperature in Section 6.0. Therefore, the staff determined that the provisions of PTLR Criterion 6 are satisfied.

RG 1.99, Revision 2, provides the staff's recommended methods for calculating the ART values for RV beltline materials. These ART values are calculated for the 1/4T and 3/4T locations in the vessel wall. The ASME Code, Section XI, Appendix G and 10 CFR Part 50, Appendix G requires that these values be used for the calculations of P-T limit curves for reactors. 10 CFR Part 50, Appendix G also requires that the ART values include the applicable results of the RV material surveillance program of 10 CFR Part 50, Appendix H. ART values for ferritic RV base metal and weld materials increase as a function of accumulated neutron fluence and the quantity of alloying elements in the materials, copper, and nickel in particular. The procedures of the RG specify the use of a CF as a means for quantifying the effect of the alloying elements on the ART values. Furthermore, the RG specifies that a CF be calculated and input into the calculation of the final ART value for each beltline material. The RG cites two possible methods for determining the CF values for the RV beltline base metal and weld materials: (1) Regulatory Position 1.1 in the RG allows the licensee to determine the CF values from applicable tables in the RG as a function of copper and nickel content or, (2) Regulatory Position 2.1 allows the use of applicable RV surveillance data to determine the CF values if the base metal or weld materials are represented in a licensee's RV material surveillance program and if two or more credible surveillance data sets become available for the material in question. The criteria for determining the credibility of the RV surveillance data sets are defined in the RG. In accordance with the requirements of 10 CFR Part 50, Appendix G, the RG states that if the procedure of Regulatory Position 2.1 results in a higher ART value than that given by using the procedure of Regulatory Position 1.1, the surveillance data should be used for determining the CF and ART. If the procedure of Regulatory Position 2.1 results in a lower value for the ART, either procedure may be used for determining the CF and ART.

To ensure that PTLRs are in compliance with the above regulatory requirements and guidelines, PTLR Criterion 7 states that if surveillance data are used in the calculations of the ART values, the PTLR contents should include the surveillance data and calculations of the CF values for the RV base metal and weld materials, as well as an evaluation of the credibility of the surveillance data against the credibility criteria of RG 1.99, Revision 2. However, the PTLR is generic for the design and is based on bounding embrittlement correlations for which surveillance data is not yet available. Therefore, the incorporation of surveillance data and related calculations is currently not applicable to the PTLR. As previously discussed, the CF and ART values in the PTLR were determined using the procedures of Regulatory Position 1.1 in RG 1.99, Revision 2. Therefore, the staff determined that the provisions of PTLR Criterion 7 are satisfied.

The staff has determined that the contents of the PTLR conform to the staff's technical criteria for PTLRs, as defined in Attachment 1 of GL 96-03. The staff has also determined that the PTLR has satisfied the requirements of 10 CFR Part 50, Appendix G. Furthermore, the staff has determined that the PTLR is compatible with the TSs and the PTLR-related TS provisions meet the technical criteria of GL 96-03. The staff notes that the PTLR provides generic, not plant-specific, heatup and cooldown P-T curves based on bounding material properties and projected fluence. To address the submittal of plant specific P-T limits, the COL applicant has provided a commitment (COM 5.03-002) discussed above. The staff finds that this approach is consistent with the guidelines of GL 96-03 and is therefore acceptable. Based on this evaluation, the staff concludes that the PTLR is acceptable for establishing limiting P-T limit curves and related input parameters. Finally, per GL 96-03, any subsequent changes in the methodology used to develop the P-T limits must be approved by the NRC.

5.3.2.2.4 Post Combined License Activities

The applicant identifies the following commitment:

Prior to fuel load, the pressure-temperature limit curves will be updated to reflect plant-specific material properties, if required. (COM 5.03-002)

5.3.2.2.5 Conclusion

The NRC staff's finding related to information incorporated by reference is in NUREG–1966. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information, and no outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix [X], Section VI.B.1, all nuclear safety issues relating to this section that were incorporated by reference have been resolved.

In addition, the staff concludes that the PTLR has satisfied the requirements of 10 CFR Part 50, Appendix G. Furthermore, the staff has determined that the PTLR is compatible with the TSs and the PTLR-related TS provisions meet the technical criteria of GL 96-03.

5.4 Reactor Coolant System Component and Subsystem Design

5.4.1 *Introduction*

NRC staff reviewed Section 5.4 of the Fermi 3 COL FSAR, Revision 3, including the corresponding sections in the referenced DCD. Specifically, the staff verified that the following subsections of the DCD contained information appropriate for incorporation by reference and that any supplemental information to be provided by the COL applicant has been addressed in the COL application:

5.4.1 Reactor Recirculation System
5.4.2 Steam Generators (not applicable to ESBWR)
5.4.3 Reactor Coolant Piping
5.4.4 Main Steamline Flow Restrictors
5.4.5 Nuclear Boiler System Isolation
5.4.6 Isolation Condenser System
5.4.7 Residual Heat Removal System
5.4.8 Reactor Water Cleanup/Shutdown Cooling System
5.4.9 Main Steamlines and Feedwater Piping
5.4.10 Pressurizer (not applicable to ESBWR)
5.4.11 Pressurizer Relief Discharge System (not applicable to ESBWR)
5.4.12 Reactor Coolant System High Point Vents
5.4.13 Safety and Relief Valves and Depressurization Valves
5.4.14 Component Supports
5.4.15 COL Information
5.4.16 References

5.4.2 *Summary of Application*

Section 5.4, "Reactor Coolant System Component and Subsystem Design" of the Fermi 3 COLA FSAR, Revision 3, incorporates by reference Section 5.4 of the certified ESBWR DCD, Revision 9. In addition, the applicant provides the following:

<u>*Supplemental Information:*</u>

- STD SUP 5.4-1

In FSAR Subsection 5.4.8, the applicant stated that operating procedures will provide guidance to prevent severe water hammer caused by mechanisms such as voided lines.

- STD SUP 5.4-2

In FSAR subsection 5.4.12, the applicant stated that human factors analysis of the control room displays and controls for the RCS vents is included as part of the overall human factors analysis of the control room displays and controls described in ESBWR DCD Chapter 18.

- STD SUP 5.4-3

In FSAR Subsection 5.4.12, the applicant stated that operating procedures for the reactor vent system address considerations regarding when venting is needed and when it is not needed, including a variety of initial conditions for which venting may be required. The development of operating procedures is addressed in Section 13.5 of the Fermi 3 COL FSAR.

5.4.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1966, the FSER related to the certified ESBWR DCD.

In addition, the relevant requirements of the Commission regulations, and the associated acceptance criteria, are described in Section 5.4 of NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)."

5.4.4 *Technical Evaluation*

As documented in NUREG–1966, NRC staff reviewed and approved Section 5.4 of the certified ESBWR DCD. The staff reviewed Section 5.4 of the Fermi 3 COL FSAR, Revision 3 and checked the referenced ESBWR DCD to ensure that the combination of the information in the ESBWR DCD and the information in the COL FSAR appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information contained in the application and the information incorporated by reference addresses the relevant information related to this section.

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[1] See "*Finality of Referenced NRC Approvals*," in SER Section 1.2.2, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER with open items issued for the North Anna application were equally applicable to the Fermi COL application, the staff undertook the following reviews:

- The staff compared the North Anna 3 COL FSAR, Revision 1, to the Fermi COL FSAR. In performing this comparison, the staff considered changes made to the Fermi COL FSAR (and other parts of the COL application, as applicable) resulting from requests for RAIs and open and confirmatory items identified in the North Anna SER with open items.

- The staff confirmed that the applicant endorsed all responses to RAIs identified in the corresponding standard content (the North Anna SER) evaluation.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the North Anna standard content to be directly applicable to the Fermi COL application. This standard content material is identified in this SER by use of italicized, double indented formatting.

The staff reviewed the information in the COL FSAR as follows:

Supplemental Information

The following portion of this technical evaluation section is reproduced from Subsection 5.2.1.1 of North Anna Unit 3 SER (ML091730304):

- *STD SUP 5.4-1*

  In FSAR Subsection 5.4.8, the applicant stated that operating procedures will provide guidance to prevent severe water hammer caused by mechanisms such as voided lines.

  The NRC staff finds that supplement STD SUP 5.4-1 is acceptable because water hammer is to be addressed in the plant operating procedures.

- *STD SUP 5.4-2*

  In FSAR subsection 5.4.12, the applicant stated that human factors analysis of the control room displays and controls for the RCS vents is included as part of the overall human factors analysis of the control room displays and controls described in ESBWR DCD Chapter 18.

  The staff found that this information is wholly incorporated in Section 18 of the Fermi 3 COL FSAR, and is thus, the staff concludes that STD SUP 5.4-2 is acceptable.

- *STD SUP 5.4-3*

  In FSAR Subsection 5.4.12, the applicant stated that operating procedures for the reactor vent system address considerations regarding when venting is needed and when it is not needed, including a variety of initial conditions for which venting may be required. The development of operating procedures is addressed in Section 13.5 of the Fermi 3 COL FSAR.

The NRC staff finds that supplement STD SUP 5.4-3 is acceptable because system venting is to be addressed in the plant operating procedures.

5.4.5 *Post Combined License Activities*

There are no post COL activities related to this section.

5.4.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG–1966. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information, and no outstanding information is expected to be addressed in the Fermi 3COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix [X], Section VI.B.1, all nuclear safety issues relating to this section that were incorporated by reference have been resolved.

In addition, the staff compared the supplemental information in the application to the guidance in Section 5.4 of NUREG–0800, and found it acceptable as discussed above.