5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

This chapter of the U.S. Nuclear Regulatory Commission's (NRC's) safety evaluation report (SER) provides the NRC staff evaluation of the North Anna 3 Combined License (COL) reactor coolant system (RCS) and connected systems of the Economic Simplified Boiling-Water Reactor (ESBWR) design including 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 also provides information on the North Anna 3 RCS and pressure-containing appendages out to and including isolation valves. This grouping of components is characterized as the RCPB and is defined in Title 10 *Code of Federal Regulations* (10 CFR) 50.2, "Definitions."

5.1 Summary Description

Section 5.1, "Summary Description," of the North Anna 3 COL Final Safety Analysis Report (FSAR), Revision 8, incorporates by reference with no departures or supplements Section 5.1, "Summary Description," of Revision 10 of the Design Control Document (DCD) for the ESBWR, referenced in Appendix E to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that 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 E, Section VI.B.1, all nuclear safety issues relating to the summary description have been resolved.

5.2 Integrity of Reactor Coolant Pressure Boundary

Section 5.2, "Integrity of Reactor Coolant Pressure Boundary," of the North Anna 3 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

Section 5.2.1.1 of the North Anna 3 COL FSAR, Revision 8, addresses the American Society of Mechanical Engineers (ASME) code edition and addenda to be used at North Anna 3 in order to demonstrate compliance with the NRC regulations in 10 CFR 50.55a, "Codes and standards."

5.2.1.1.2 Summary of Application

Section 5.2.1.1 of the North Anna 3 COL FSAR, Revision 8, incorporates by reference Section 5.2.1.1 of the certified ESBWR DCD, Revision 10, referenced in 10 CFR Part 52, Appendix E. In addition, in FSAR Section 5.2.1.1, the applicant provides the following:

Supplemental Information

- STD SUP 5.2-2

In FSAR Section 5.2.1.1, the applicant provided supplemental information that preservice inspection (PSI) and In-Service Inspection (ISI) of the RCPB are conducted in accordance with the applicable edition and addenda of the ASME Boiler and Pressure Vessel Code (BPV Code)

Section XI, which is required by 10 CFR 50.55a. FSAR Section 5.2.1.1 also states the following:

> 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, "Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor." In addition, the relevant requirements of the Commission regulations for compliance with 10 CFR 50.55a, and the associated acceptance criteria, are in Section 5.2.1.1 of NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," (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 the staff's review of the information in the North Anna 3 COL application (COLA). 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. Furthermore, NRC regulations in 10 CFR 50.55a, as they relate 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, the staff followed the guidance in Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," June 2007, in evaluating North Anna 3 COL FSAR Section 5.2.1.1 for compliance with NRC regulations.

5.2.1.1.4 Technical Evaluation

As documented in NUREG–1966, the staff reviewed and approved Section 5.2 of the certified ESBWR DCD. The staff reviewed Section 5.2.1.1 of the North Anna 3 COL FSAR and checked the referenced ESBWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ESBWR DCD appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information 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.

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

Supplemental Information

- STD SUP 5.2-2

The FSAR incorporates by reference Section 5.2.1.1 of the ESBWR DCD Tier 2, which refers to Table 3.2-1, "Classification Summary," and Table 3.2-3, "Quality Group Designations – Codes and Industry Standards," of the ESBWR DCD for the ASME Code applied to components in the ESBWR design with respect to Section III of the ASME BPV Code.

In Request for Additional Information (RAI) 05.02.01.01-1(Agencywide Documents Access and Management System (ADAMS) Accession No. ML082110133), dated July 28, 2008, the staff requested that the applicant 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 its response to this RAI dated September 11, 2008 (ADAMS Accession No. ML082610417), the applicant stated that the FSAR would be revised to provide references to the appropriate sections that discuss compliance with ASME BPV Code Section XI and the ASME OM Code, "Operation and Maintenance of Nuclear Power Plants." As a result, 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 testing (PST) and in-service 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. The staff 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 is therefore acceptable. Therefore, RAI 05.02.01.01-1 is resolved and closed.

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 staff's finding related to information incorporated by reference is in NUREG–1966. The 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 E, Section VI.B.1, all nuclear safety issues relating to compliance with 10 CFR 50.55a 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 Section 5.2.1.1 the SRP, and other NRC RGs. The staff's review concludes that the applicant has presented adequate information in the North Anna 3 COL FSAR to meet the requirements in 10 CFR 50.55a.

5.2.1.2 Applicable Code Cases

5.2.1.2.1 Introduction

Section 5.2.1.2, “Applicable Code Cases,” of the North Anna 3 COL FSAR, Revision 8, addresses the applicable Code Cases for the ASME BPV Code and the ASME OM Code. This section also addresses NRC RGs that indicate the acceptance of ASME Code Cases with or without conditions. In general, ASME develops a Code Case based on inquiries from the nuclear industry associated with code clarifications, modifications, or alternatives to the code. All Code Cases will remain valid and available for use until annulled by the ASME. ASME Code Cases acceptable to the staff are published in RG 1.84, Revision 35, “Design and Fabrication Code Case Acceptability, ASME Section III, Division 1”, RG 1.147,Revision 16, “In-Service 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), 10 CFR 50.55a(b)(5) and 10 CFR 50.55(b)(6), respectively.

5.2.1.2.2 Summary of Application

Section 5.2.1.2 of the North Anna 3 COL FSAR, Revision 8, incorporates by reference Section 5.2.1.2 of the certified ESBWR DCD, Revision 10, referenced in 10 CFR Part 52, Appendix E, 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. In addition, the relevant requirements of the Commission regulations for the applicable code cases, and the associated acceptance criteria, are in Section 5.2.1.2 of the SRP. The NRC regulations in 10 CFR Part 50 and 10 CFR Part 52 provide the regulatory basis for the staff’s review of the information in the North Anna 3 COLA. 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. Furthermore, NRC regulations in 10 CFR 50.55a, that are 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 the ASME BPV Code Section III Code Cases related to design, fabrication, materials, and testing that are acceptable with applicable conditions for implementation at nuclear power plants. RG 1.147 lists ASME BPV Code Section XI Code Cases that are acceptable, with the 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 that are acceptable with applicable conditions for implementation at nuclear power plants.

The staff followed the guidance in SRP Section 5.2.1.2 and RG 1.206 to evaluate North Anna 3 FSAR Section 5.2.1.2 for compliance with NRC regulations.

5.2.1.2.4 Technical Evaluation

As documented in NUREG–1966, the staff reviewed and approved Section 5.2.1.2 of the certified ESBWR DCD. The staff reviewed Section 5.2.1.2 of the North Anna 3 COL FSAR and checked the referenced DCD to ensure that the combination of the information in the COL FSAR and the information in the ESBWR DCD represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the relevant information related to applicable Code Cases.

The North Anna 3 FSAR incorporates by reference Section 5.2.1.2 of the ESBWR DCD Tier 2, without departures or supplemental information. In ESBWR DCD, Tier 2, Section 5.2.1.2 indicates that the various ASME Code 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, Section 5.2.1.2 also notes that RG 1.84 and RG 1.147 provide a list of ASME Code design and fabrication Code Cases that the NRC has generically approved.

In RAI 05.02.01.02-1, dated July 28, 2008 (ADAMS Accession No. ML082110133), the staff requested that the applicant discuss the use of any Code Cases related to the ASME BPV Code and the OM Code not listed in ESBWR DCD, Tier 2, Table 5.2-1. In the response to this RAI dated September 11, 2008 (ADAMS Accession No. ML082610417), the applicant stated that no ASME BPV Code Section III or Section XI Code Cases, other than those listed in Table 5.2-1 of the ESBWR DCD, had been identified as necessary. The applicant stated that other Code Cases approved by the NRC in RG 1.147 might be used during the development and implementation of the PSI and ISI Programs. ESBWR DCD, Tier 2, Section 3.9.3.7.1b, "Inspection, Testing, Repair, and/or Replacement of Snubbers," references RG 1.192 for the use of Code Cases (such as Code Case OMN-13) applicable to IST of dynamic restraints. ESBWR DCD, Tier 2, Section 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. In addition, the applicant stated that other Code Cases approved by the NRC in RG 1.192 might be used during the development and implementation of the PST and IST Programs for North Anna 3. In the RAI response, the applicant indicated that the FSAR would be revised to reference RG 1.192 in Section 5.2.1.2. Subsequently, the ESBWR DCD was revised to include RG 1.192 in the list of RGs to be used in meeting the requirements of GDC 1 and 10 CFR 50.55a. The staff finds that the description of the planned use of ASME Code Cases in ESBWR DCD, Tier 2, Section 5.2.1.2, is consistent with the applicable NRC regulations and RGs. Therefore, RAI 05.02.01.02-1 is closed without the need to revise Section 5.2.1.2 of the North Anna 3 FSAR.

In RAI 05.02.01.02-2, dated July 28, 2008 (ADAMS Accession No. ML082110133), the staff requested that the applicant discuss its compliance with the requirements regarding the use of annulled Code Cases specified in 10 CFR 50.55a(b)(4), (5), and (6). In the response to this RAI dated September 11, 2008 (ADAMS Accession No. ML082610417), the applicant stated that the design, fabrication, and construction of safety-related components will be 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 applicant also noted that Section 5.2.1.1 of the ESBWR DCD specifies that the ESBWR meets the relevant requirements of 10 CFR 50.55a. The applicant added that these requirements include the 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 PSI/ISI and PST/IST of safety-related components, the applicant stated 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 staff finds that the plans described by the applicant for using ASME Code Cases at North Anna 3 meet the applicable NRC regulations. Therefore, RAI 05.02.01.02-2 is resolved and closed.

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 staff's finding related to information incorporated by reference is in NUREG–1966. The 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 E, 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 FSAR section addresses the safety and relief valves and the portion of the reactor protection system that ensures overpressure protection for the RCPB during operation at power.

Section 5.2.2, "Overpressure Protection," of the North Anna 3 COL FSAR, Revision 8, incorporates by reference Section 5.2.2, "Overpressure Protection," of the certified ESBWR DCD, Revision 10, referenced in 10 CFR Part 52, Appendix E, with no departures or supplements. The staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The staff's review confirmed that 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 E, 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 FSAR section 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 5.2.3, "Reactor Coolant Pressure Boundary Materials," of the North Anna 3 COL FSAR, Revision 8, incorporates by reference Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," of the certified ESBWR DCD, Revision 10, referenced in 10 CFR Part 52, Appendix E, with no departures or supplements. The staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review. The staff's review confirmed that 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 E, Section VI.B.1, all nuclear safety issues relating to the RCPB materials have been resolved.

5.2.4 Preservice and In-Service Inspection and Testing of Reactor Coolant Pressure Boundary

5.2.4.1 Introduction

Section 5.2.4 of the North Anna 3 COL 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 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 In-Service Inspection of Nuclear Power Plant Components."

5.2.4.2 Summary of Application

Section 5.2.4 of the North Anna 3 COL FSAR, Revision 8, incorporates by reference Section 5.2.4 of the certified ESBWR DCD, Revision 10, referenced in 10 CFR Part 52, Appendix, Section VI.B.1. In addition, the applicant provided the following information in FSAR Section 5.2.4:

COL Items

- STD COL 5.2-1-A — Preservice and In-Service Inspection Program Description

The applicant provided additional information in FSAR Section 5.2.4 and Sections 5.2.4.3.4, 5.2.4.6, and 5.2.4.11 in order to fully describe the PSI and ISI Programs; including the applicable ASME Code Edition and Addenda, the certification of nondestructive examination (NDE) personnel as amended by 10 CFR 50.55a, system leakage tests as amended by 10 CFR 50.55a, and the PSI and ISI Program implementation milestones.

- STD COL 5.2-3-A — Preservice and In-Service Inspection NDE Accessibility Plan Description

The applicant provided additional information in FSAR Section 5.2.4 and Section 5.2.4.2 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.

Supplemental Information

- STD SUP 5.2-1

The applicant provided supplemental information in FSAR Section 5.2.4.6 to describe the relevant Technical Specification (TS) sections that address system pressure tests and RCS pressure-temperature (P-T) limits.

License Condition

- Part 10, License Condition 3.6 — Operational Program Readiness

In Section 3.6 of Part 10, "Tier 1/ITAAC/Proposed License Conditions," of the COLA, the applicant proposed an operational program readiness license condition.

5.2.4.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is in NUREG–1966. In addition, the relevant requirements of the Commission regulations for the RCPB in-service inspections and testing, and the associated acceptance criteria, are in Section 5.2.4 of the SRP.

The regulatory basis for accepting the COL items (STD COL 5.2-1-A, STD COL 5.2-3-A) is in GDC 32, "Inspection of reactor coolant pressure boundary," as it relates to the periodic inspection and testing of the RCPB; and 10 CFR 50.55a, as it relates to the requirements for testing and inspecting 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 the Commission policy for fully describing an operational program. Moreover, the regulatory basis for accepting STD SUP 5.2-1 is 10 CFR 50.55a.

5.2.4.4 Technical Evaluation

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

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

COL Items

- STD COL 5.2-1-A — Preservice and In-Service Inspection Program Description

ESBWR DCD COL Item 5.2-1-A states that the COL applicant is responsible "for providing a full description of the preservice and in-service inspection programs and augmented inspection programs by supplementing, as necessary, the information in Section 5.2.4 and to provide the milestones for their implementation." To address this COL item, the applicant provided additional information in FSAR Section 5.2.4 and Sections 5.2.4.3.4, 5.2.4.6, and 5.2.4.11 in order to provide a full description of the North Anna 3 PSI and ISI Programs.

In FSAR Section 5.2.4, the applicant stated that "the initial in-service 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 in-service 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 (10 CFR 50.55a) 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 in FSAR Section 5.2.4 is acceptable because it is in compliance with the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(b).

In FSAR 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)." 10 CFR 50.55a(b)(2)(xviii) imposes a modification on the use of the latest edition and addenda of the Code incorporated by reference into 10 CFR 50.55a by requiring that Level I and Level II NDE personnel be recertified on a 3-year interval in lieu of the 5-year interval specified in Section XI, IWA-2314. Given that the initial ISI program will be in accordance with the latest edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a, the information provided in the FSAR Section 5.2.4.3.4 is acceptable because it is in compliance with 10 CFR 50.55a(b).

In FSAR Section 5.2.4.6 the applicant stated 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)(xxvi) imposes a limitation on the use of the 2001 Edition through the latest edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a by requiring that the provisions of IWA-4540(c) from the 1998 Edition of Section XI for pressure testing Class 1, 2, and 3 mechanical joints be applied. Given that the initial ISI program will be in accordance with the latest edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a, the information provided in the FSAR Section 5.2.4.6 is acceptable because it is in compliance with 10 CFR 50.55a(b).

In FSAR Section 5.2.4.11, the applicant stated that DCD Section 5.2.4 "fully describes the Preservice and In-Service Inspection and Testing Programs for the RCPB and that the implementation milestones for the Preservice and In-Service 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 13.4-201 and found that the implementation milestones for the PSI/ISI operational programs are listed.

In North Anna 3 COL FSAR, Part 10, Section 3.6, the applicant has also provided a proposed license condition related to the PSI/ISI operational program which includes the programs listed in Table 13.4-201.

The staff finds implementation milestones are acceptable because they are in accordance with the requirements of ASME Section XI and 10 CFR 50.55a. The staff also finds that the proposed license condition is acceptable because it is in accordance with SECY-05-0197. As discussed in SECY-05-0197, a COL applicant should provide schedules for the implementation of operational programs in order to support the planning for and conducting of NRC inspections. Therefore, the staff will include such license condition in the North Anna 3 COL.

Based on the evaluation described above, STD COL 5.2-1-A is acceptable.

- STD COL 5.2-3-A — Preservice and In-Service Inspection NDE Accessibility Plan Description

ESBWR DCD COL Item 5.2-3-A states that the COL applicant is responsible "for developing a plan and providing a full description of its use during construction, PSI, ISI, and during design activities for components that are not included in the referenced certified design, to preserve accessibility to piping systems to enable NDE of ASME Code Class 1 austenitic and dissimilar metal welds during in-service inspection." To address this COL item, the applicant provided additional information in FSAR Sections 5.2.4 and Section 5.2.4.2.

In FSAR Section 5.2.4, the applicant stated that all Class 1 austenitic or dissimilar metal welds are included in the referenced certified design. The applicant described in FSAR 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 inspection 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 in-service 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 will exist to maintain the accessibility to enable the performance of in-service examinations for austenitic and dissimilar metal welds. The information provided by the applicant meets the requirements of 10 CFR 50.55a(g)(3) and ASME Section XI. Based on the evaluation described above, STD COL 5.2-3-A is acceptable.

Supplemental Information

- STD SUP 5.2-1

Under Section 5.2.4.6, the applicant stated that system pressure tests and correlated TS requirements are provided in the plant TS 3.4.4, "RCS Pressure and Temperature P/T Limits," and TS 3.10.1, "In-Service 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

In FSAR Table 13.4-201, the applicant provided the implementation milestones for the PSI and ISI Programs.

The applicant proposed a license condition in Part 10 of the COLA Revision 8 as follows:

3.6 Operational Program Readiness

The licensee shall submit to the Director of NRO, a schedule, no later than 12 months after issuance of the COL, for implementation of the operational programs listed in FSAR Table 13.4-201. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the operational programs in the FSAR table have been fully implemented.

5.2.4.6 Conclusion

The staff's finding related to information incorporated by reference is in NUREG–1966. The 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 E, 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 in North Anna 3 COL FSAR Section 5.2.4 meets the relevant guidelines in SRP Section 5.2.4 and RG 1.206, and is therefore acceptable. The staff further concludes that the North Anna 3 COL FSAR PSI/ISI Programs and implementation milestones are consistent with the policy established in SECY-05-0197. Conformance with these guidelines and the 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

Section 5.2.5 of the North Anna 3 COL FSAR discusses the RCPB leakage detection systems that are designed to detect and, to the extent practical, identify the source of reactor coolant leakage.

5.2.5.2 Summary of Application

Section 5.2.5 of the North Anna 3 COL FSAR, Revision 8, incorporates by reference Section 5.2.5 of the certified ESBWR DCD, Revision 10, referenced in 10 CFR Part 52, Appendix E. In addition, in FSAR Section 5.2.5, the applicant provides the following:

<u>COL Item</u>

- STD COL 5.2-2-A Leak Detection Monitoring

In the ESBWR DCD, Revision 9, STD COL Item 5.2-2-H becomes STD COL 5.2-2-A.

In FSAR Section 5.2.5, the applicant provided additional information to address STD COL 5.2-2-A. The applicant replaced Section 5.2.5.9, "Leak Detection Monitoring," of the ESBWR DCD, Tier 2 with new information stating that operators are provided with procedures for detecting, monitoring, recording, trending, and determining the sources of the RCPB leakage. In addition, 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. In addition, the relevant requirements of the Commission regulations for RCPB leakage detection, and the associated acceptance criteria, are in Section 5.2.5 of the SRP.

The staff's acceptance of the leakage detection design is based on 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.

Also, the staff followed the guidance in RG 1.206 for evaluating the compliance of North Anna 3 COL FSAR Section 5.2.5 with NRC regulations.

5.2.5.4 Technical Evaluation

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

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

COL Item

- STD COL 5.2-2-A Leak Detection Monitoring

In the ESBWR DCD, Revision 9, STD COL Item 5.2-2-H becomes STD COL 5.2-2-A.

The staff identified that the substitution of Tier 2, 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 Tier 2, 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 (ADAMS Accession No. ML081750645) dated June 23, 2008, the staff requested 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 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 COLA.

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 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.

The 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 is committed to be consistent with ESBWR DCD, Tier 2, Section 5.2.5. Tier 2, Section 5.2.5 of the DCD Revision 10, 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 applicant provided the appropriate information in FSAR Revision 9.

FSAR Section 13.5.2.1, “Operating and Emergency Operating Procedures,” states the following:

> Operating procedures are developed at least six months prior to fuel load to allow sufficient time for plant staff familiarization and to allow staff adequate time to review the procedures and to develop operator licensing examinations.

The staff concludes that the above information meets the relevant guidelines in SRP Section 5.2.5 and RG 1.206, Regulatory Positions C.III.1 and C.I.5.2.5, and is thus acceptable. Conformance with these guidelines and with GDC 2 and GDC 30 provides an acceptable basis for satisfying the NRC requirements.

5.2.5.5 Post Combined License Activities

The applicant proposed a license condition in Part 10 of the COLA Revision 8 as follows:

> 3.6 Operational Program Readiness

The licensee shall submit to the Director of NRO, a schedule, no later than 12 months after issuance of the COL, for implementation of the operational programs listed in FSAR Table 13.4-201. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the operational programs in the FSAR table have been fully implemented.

5.2.5.6 Conclusion

The staff’s finding related to information incorporated by reference is in NUREG–1966. The 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 E, 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 supplemental information in the COLA to the relevant NRC regulations, the guidance in Section 5.2.5 of the SRP, and other NRC RGs. The staff’s review concluded that the applicant has presented adequate information in the North Anna 3 COL FSAR to meet the requirements of GDC 2 and GDC 30, and the guidance in RG 1.206 and SRP Section 5.2.5.

5.3 <u>Reactor Vessel</u>

5.3.1 Reactor Vessel Materials

5.3.1.1 Introduction

Section 5.3.1, “Reactor Vessel Materials,” of the North Anna 3 COL FSAR, Revision 8, addresses the reactor vessel (RV) material specifications, including weld materials, special processes used to manufacture and fabricate 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 RV fasteners.

5.3.1.2 Summary of Application

Section 5.3.1 of the North Anna 3 COL FSAR, Revision 8, incorporates by reference Section 5.3.1 of the certified ESBWR DCD, Revision 10, referenced in 10 CFR Part 52, Appendix E. In addition, in FSAR Section 5.3.1, the applicant provides the following:

COL Items

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

The applicant provided additional information in FSAR Section 5.3.1.8 in order to fully describe the North Anna 3 RVSP and its implementation.

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

This COL item is discussed in SER Section 5.3.2, “Pressure-Temperature Limits.”

- NAPS COL 5.3-2-A

In FSAR Section 5.3.1.6, the applicant states a need to delete the parenthetical statement in the first sentence of the first paragraph in ESBWR DCD, Tier 2, Section 5.3.1.6. This statement refers to DCD Section 5.3.1.8, the content of which is completely replaced with new information in FSAR Section 5.3.1.8 by the resolution of STD COL 5.3-2-A.

License Conditions

- Part 10, License Condition 3.5.7 Fuel Loading
- Part 10, License Condition 3.6 Operational Program Readiness

5.3.1.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is in NUREG–1966. In addition, the relevant requirements of the Commission regulations for RV materials, and the associated acceptance criteria, are in Section 5.3.1 of the SRP.

In particular, the regulatory basis for the acceptance of the RVSP information (COL Items NAPS COL 5.3-2-A and STD COL 5.3-2-A) is established in:

- 10 CFR Part 50, Appendix A, GDC 32, as it relates to the RVSP
- 10 CFR 50.60, “Acceptance criteria for fracture prevention measures for light-water nuclear power reactors for normal operation,” as it relates to compliance with the requirements of 10 CFR Part 50, Appendices G and H
- 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” as it relates to materials testing and acceptance criteria for fracture toughness
- 10 CFR Part 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” as it relates to the RVSP
- SECY-05-0197, as it relates to fully describing an operational program

Also, the staff followed the guidance in RG 1.206 for evaluating the compliance of North Anna 3 COL FSAR Section 5.3.1 with NRC regulations.

5.3.1.4 Technical Evaluation

As documented in NUREG–1966, the staff reviewed and approved Section 5.3.1 of the certified ESBWR DCD. The staff reviewed Section 5.3 of the North Anna 3 COL FSAR, Revision 8, and checked the referenced ESBWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ESBWR DCD appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the relevant information related to this section.

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

<u>*COL Items*</u>

- STD COL 5.3-2-A and NAPS COL 5.3-2-A Reactor Vessel Materials Surveillance Program

ESBWR DCD COL Item 5.3.2-A states that the COL applicant will "develop a description of the RV material surveillance program and milestones per Section 5.3.1.8." To address this COL item, the applicant provided STD COL 5.3-2-A and NAPS COL 5.3-2-A in order to fully describe the North Anna 3 RVSP and its implementation.

After reviewing the information provided in North Anna 3 COL FSAR Revision 0, Section 5.3.1, including the information referenced in DCD Tier 2, 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 (ADAMS Accession No. ML082030183), dated July 21, 2008, in order to complete this review.

In RAI 05.03.01-1, 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 the response to RAI 05.03.01-1 dated September 3, 2008 (ADAMS Accession No. ML082520378), the applicant 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; the applicant also agreed to update the FSAR. The staff determined that the applicant's response appropriately addressed the issue in RAI 05.03.01-1. The staff reviewed FSAR Section 5.3.1.8 and confirmed that the information described in the response to RAI 05.03.01-1 has been included in Revision 1 of the FSAR. Therefore, the staff finds that the applicant has adequately addressed this issue and RAI 05.03.01-1 is resolved and closed.

In FSAR Revision 8, Section 5.3.1.8, the applicant describes in detail the preparation of the surveillance capsule specimens; the number and type of specimens; the location of the specimen capsules in the core beltline region; and the reporting of test results. The staff finds that the information in FSAR Section 5.3.1.8 is acceptable because it is in accordance with ASTM International (ASTM) E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," and 10 CFR Part 50, Appendix H.

The implementation milestone for the RVSP is provided in FSAR Section 13.4. In Table 13.4-201, the applicant has stated that the RVSP is to be implemented prior to fuel load and required by a license condition. In addition, in North Anna 3 COL, Part 10, the applicant has provided the following proposed license conditions related to the RVSP:

> The licensee shall implement the Reactor Vessel Materials Surveillance Program prior to initial fuel load. (North Anna 3 COL, Part 10, Section 3.5.7)
>
> The licensee shall submit to the Director of NRO, a schedule, no later than 12 months after issuance of the COL, for implementation of the operational programs listed in FSAR Table 13.4-201. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the operational programs in the FSAR table have been fully implemented. (North Anna 3 COL, Part 10, Section 3.6)

Based on the information described above, the staff finds the applicant's proposed license conditions are acceptable because they provide assurance that the operational program will be implemented with specific milestones consistent with policy established in SECY-05-0197. The staff finds that the COL applicant has met the minimum guidelines provided in RG 1.206 regarding the description of the RVSP and its implementation and that the applicant has provided a sufficient level of detail to "fully describe" its RVSP as an operational program. On this basis, the COL information items are acceptable.

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

The staff's evaluation of STD COL 16.0-1-A 5.6.4-1 is in Section 5.3.2 of this SER.

5.3.1.5 Post Combined License Activities

In FSAR Table 13.4-201, the applicant described the implementation milestone for the RVSP.

As discussed above, the staff has identified the following license conditions:

In Section 3.5.7 of Part 10 of the COLA, Revision 8, the applicant identifies the following license conditions:

> The licensee shall implement each operational program prior to initial fuel load:
>
> - Reactor Vessel Material Surveillance Program

In Section 3.6 of Part 10 of the COLA, the applicant identifies the following license condition:

> The licensee shall submit to the Director of NRO, a schedule, no later than 12 months after issuance of the COL, for implementation of the operational programs listed in FSAR Table 13.4-201. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the operational programs in the FSAR table have been fully implemented.

5.3.1.6 Conclusion

The staff's finding related to information incorporated by reference is in NUREG–1966. The 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 E, 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 information in the COLA to the relevant NRC regulations, the guidance in Section 5.3.1 of the SRP, and other NRC RGs. The staff's review concludes that the applicant has adequately addressed STD COL 5.3-2-A, in accordance with the acceptance criteria in SRP Section 5.3.1, the guidance in RG 1.206 and is consistent with the policy established in SECY-05-0197. Conformance with these guidelines and the policy provides an acceptable basis for satisfying the requirements of 10 CFR Part 50, Appendices G and H. The applicant's additional information is therefore acceptable.

5.3.2 Pressure-Temperature Limits

5.3.2.1 Introduction

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

5.3.2.2 Summary of Application

Section 5.3.2 of the North Anna 3 COL FSAR, Revision 8, incorporates by reference Section 5.3.2 of the certified ESBWR DCD, Revision 10, referenced in 10 CFR Part 52, Appendix E. In addition, in FSAR Section 5.3.1.5, the applicant provided the following:

<u>COL Item</u>

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

In ESBWR DCD, Revision 9, COL Item 16.0-1-H 5.6.4-1 becomes STD COL 16.0-1-A 5.6.4-1.

In FSAR Section 5.3, the applicant provides supplemental information in Section 5.3.1.5, "Fracture Toughness Compliance with 10 CFR Part 50, Appendix G," which states:

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

In addition, the applicant has provided technical report NEDC-33441P, "GE Hitachi Nuclear Energy Methodology for the Development of Economic Simplified Boiling Water Reactor (ESBWR) Reactor Pressure Vessel Pressure-Temperature Curves," Revision 6. This report is referenced in North Anna 3 TS Section 5.6.4 as providing the analytical methods used to determine the RCS pressure and temperature limits.

5.3.2.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is in NUREG–1966. In addition, the regulatory basis for the acceptance of STD COL 16.0-1-A 5.6.4-1 is in 10 CFR Part 50, Appendix G, which provides the requirements for P-T limits.

5.3.2.4 Technical Evaluation

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

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

COL Item

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

ESBWR DCD, Tier 2, Section 5.3.1.5 states that the COL applicant, in accordance with the 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 pressure and temperature limit report (PTLR) submittal for NRC to review. To address this item, the North Anna 3 COL FSAR, Revision 1, Section 5.3.1.5, states that "the pressure-temperature limit curves are developed in accordance with the PTLR as discussed in North Anna Unit 3 TSs (Part 4 of the COLA) Section 5.6.4." In Section 5.6.4, the applicant states that the PTLR methodology is scheduled for submittal to the NRC in the second quarter of 2009. This was tracked as Open Item 5.3.2-2.

In addition, the staff identified the need for the applicant to address the submittal of plant-specific P-T limits in the FSAR. Therefore, in RAI 05.03.02-1(ADAMS Accession No. ML091480213), dated May 28, 2009, the staff requested that the FSAR be revised to provide a commitment to submit the P-T limits using plant-specific material properties before fuel loading. This RAI was tracked as Open Item 5.3.2-1.

The resolution of the Open Items is described in the sections below.

Resolution of Standard Content Open Items

To address Open Item 5.3.2-2, the applicant submitted Technical Report NEDC-33441P, "GE Hitachi Nuclear Energy Methodology for the Development of Economic Simplified Boiling Water Reactor (ESBWR) Reactor Pressure Vessel Pressure-Temperature Curves," Revision 6, in a letter dated December 6, 2013 (ADAMS Accession No. ML13346A654) (hereafter referred to as the ESBWR PTLR). This report was prepared by GE-Hitachi (GEH) and was submitted in support of the North Anna 3 COLA to address the COL Item described above. 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 in Generic Letter (GL) 96-03, "Relocation of

Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits.”

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. 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 ESBWR PTLR.

5.3.2.4.1 Summary of the Regulatory Requirements for the Submittal of a PTLR

The NRC established requirements in 10 CFR Part 50 to protect the integrity of the RCPB 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, “Radiation Embrittlement of Reactor Vessel Materials”; GL 92-01, Revision 1,“Reactor Vessel Structural Integrity, 10 CFR 50.54(f)”; GL 92-01; Revision 1, Supplement 1,“Reactor Vessel Structural Integrity”; 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. In GL 92-01, Revision 1, Supplement 1, the staff 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 (ASME Code Section XI, Appendix G methodology) incorporates the provisions of ASME Code Cases N-588, “Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels Section XI, Division 1”; and N-640, “Alternative Reference Fracture Toughness for Development of P-T Limit Curves Section XI, Division 1.” Additionally, Appendix G to 10 CFR Part 50 imposes minimum head flange temperatures when the 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 to implement a PTLR. GL 96-03 establishes the information that must be included in (1) an acceptable PTLR methodology (with the P-T limit methodology as its subset), and (2) the PTLR itself. Technical specification task force (TSTF)-419 provides additional guidance, which includes an alternative format for documenting the implementation of a PTLR in the “Administrative Controls” section of a facility’s TS.

5.3.2.4.2 Evaluation of the North Anna 3 COL Technical Specification Requirements for Implementation and Control of a PTLR

The North Anna 3 COL TS contain all of the necessary provisions required for the implementation and control of a PTLR. The North Anna 3 TS are in Part 4 of the COLA. 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, SRs, 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.

5.3.2.4.3 Evaluation of the ESBWR Generic PTLR Contents and Methodology against the Seven Criteria for PTLR Contents in Attachment 1 of GL 96-03

As discussed in Section 1.0 of the ESBWR PTLR, this report describes the methodology used to develop the P-T limits and provides specific P-T curves for the RV. Accordingly, the PLTR utilizes generic inputs for the RV beltline material chemistry, the initial nil-ductility reference temperature (RT_{NDT}) values, and the 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 North Anna 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 in the subsections that follow.

5.3.2.4.3.1 PTLR Criterion 1

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,"Reactor coolant pressure boundary"; GDC 30; and GDC 31,"Fracture prevention of reactor coolant pressure boundary," 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 provides 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, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." 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.

5.3.2.4.3.2 PTLR Criterion 2

The requirements for designing and implementing RV material surveillance programs are found in NRC regulation 10 CFR Part 50, Appendix H. The rule requires that RV material surveillance programs for operating reactors comply with the specifications of ASTM E 185. The rule requires that the program design and the surveillance capsule withdrawal schedules for the programs must meet the edition of ASTM E 185 that is current on the issue date of the ASME

Code that the RV was purchased; although the rule permits more recent versions up through the 1982 version of ASTM E 185 to be used.

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 placed on the docket by the licensee requesting approval of the PTLR for its units. This criterion assures that the 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 amount exceeds the three capsules specified in ASTM E 185-82, because the predicted transition temperature shift is less than 55.6 degrees Celsius (°C) (100 degrees Fahrenheit [°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.

5.3.2.4.3.3 PTLR Criterion 3

PTLR Criterion 3 states that the low temperature overpressure protection system lift- setting limits for the power operated relief valves developed using NRC-approved methodologies may be included in the PTLR. This criterion is not applicable to the ESBWR design and is thus not applicable to the North Anna 3 COL.

5.3.2.4.3.4 PTLR Criterion 4

The P-T limits for operating reactors are generated using a method that accounts for the effects of neutron embrittlement on the fracture toughness of the RV beltline materials in accordance with 10 CFR Part 50, Appendix G. 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. RG 1.99, Revision 2, provides the staff's recommended methodologies for calculating the 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 (T is the vessel beltline thickness).

To ensure compliance with the requirements of 10 CFR Part 50, Appendix G, PTLR Criterion 4 states that the 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 in Section 3 of the PTLR. In PTLR Section 3.4, the applicant describes how the procedures outlined in RG 1.99, Revision 2, were 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 273.9 °C (525 ^{o}F). The staff notes that for the procedures of this RG to be valid for nominal irradiation temperatures less than 273.9 °C (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 0.56 °C (1 ^{o}F) increase in the ΔRT_{NDT} for each 0.56 °C (1 ^{o}F) decrease in irradiation temperature below 287.8 °C (550 ^{o}F). This method will be validated for North Anna 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 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. The staff noted that the applicant had not calculated the ART value at the 3/4T location, which is relevant to the heatup P-T limit calculation, because the ART value at 1/4T is assumed to be bounding for heatup and cooldown. The staff verified that the applicant's assumption is valid.

Based on the evaluation described above, the staff finds that the procedure used to calculate the ART values is consistent with the guidance of RG 1.99, Revision 2. The procedure is 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.

5.3.2.4.3.5 PTLR Criterion 5

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 ASME Code Section XI, Appendix G were used to generate the P-T limit curves. In 10 CFR Part 50, Appendix G, Table 1 summarizes 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 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 differential pressure (DP) nozzles. These data meet the provisions of PTLR Criterion 5. This criterion 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 limit curves for heatup and cooldown 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 RV operation.

5.3.2.4.3.6 PTLR Criterion 6

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 identifies the required 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.

5.3.2.4.3.7 PTLR Criterion 7

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. ASME Code Section XI, Appendix G and 10 CFR Part 50, Appendix G require these values to be used for calculating 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 RG defines the criteria for determining the credibility of the RV surveillance data sets. 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 obtained by using the procedure of Regulatory Position 1.1, the surveillance data should be used to determine the CF and the 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 the 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.

5.3.2.4.4 Staff Findings on the Acceptability of the PTLR

Based on the evaluation described above, 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 also determined that the PTLR satisfies the requirements of 10 CFR Part 50, Appendix G. Furthermore, the staff determined that the PTLR is compatible with the TS 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 (Open Item 5.3.2-1), the COL applicant provided the following statement in FSAR, Revision 9, Section 5.3.1.5:

> Prior to fuel load, the pressure-temperature limit curves will be updated to reflect plant-specific material properties, if required.

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 finds that STD COL 16.0-1-A 5.6.4-1 is acceptable. As a result, the Phase 2 North Anna 3 SER with Open Items 5.3.2-1 and 5.3.2-2 are resolved.

5.3.2.5 Post Combined License Activities

The staff has noted the following FSAR requirement for North Anna 3:

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

5.3.2.6 Conclusion

The staff's finding related to information incorporated by reference is in NUREG–1966. The 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 E, 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 ESBWR PTLR (NEDC-33441P, Revision 6) is acceptable for use by the North Anna 3 COL for establishing limiting P-T limit curves and related input parameters. Pursuant to North Anna 3 TS requirement 5.6.4c, future COL holders and licensees will be required to provide updated, plant-specific PTLRs to the NRC "upon issuance for each RV neutron fluence period and for any PTLR revision or supplement thereto." Finally, per GL 96-03, any subsequent changes in the methodology used to develop the P-T must be approved by the NRC.

The staff also concludes that the information in STD COL 16.0-1-A 5.6.4-1 meets the relevant acceptance criteria of SRP Section 5.3.2 and the guidance in RG 1.206. Conformance with these guidelines provides an acceptable basis for satisfying the requirements of 10 CFR Part 50, Appendix G.

5.3.3 <u>Reactor Vessel Integrity</u>

5.3.3.1 Introduction

This section of the North Anna 3 COL FSAR discusses all factors related to RV integrity.

5.3.3.2 Summary of Application

Section 5.3.3 of the North Anna 3 COL FSAR, Revision 8, incorporates by reference Section 5.3.3 of the certified ESBWR DCD, Revision 10, referenced in 10 CFR Part 52, Appendix E. In addition, in FSAR Section 5.3.3, the applicant provided the following:

<u>Supplemental Information</u>

- STD SUP 5.3-1

In FSAR Section 5.3.3.6, the applicant provides supplemental information regarding operating procedures intended to ensure that the P-T limits are not exceeded during normal operating conditions or anticipated plant transients.

5.3.3.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is in NUREG–1966. In addition, the relevant requirements of the Commission regulations for RV integrity, and the associated acceptance criteria, are in Section 5.3.3 the SRP and RG 1.206.

5.3.3.4 Technical Evaluation

As documented in NUREG–1966, the staff reviewed and approved Section 5.3.3 of the certified ESBWR DCD. The staff reviewed Section 5.3.3 of the North Anna 3 COL FSAR and checked the referenced ESBWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ESBWR DCD represents the complete scope of information relating to the review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the relevant information related to this section.

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

Supplemental Information

- STD SUP 5.3-1

In FSAR Section 5.3.3.6, the applicant added supplemental information stating that the development of plant operating procedures is addressed in Section 13.5. These procedures require compliance with the TS, which are intended to ensure that the P-T limits identified in DCD Section 5.3.2 are not exceeded during normal operating conditions and anticipated plant transients. The staff finds that STD SUP 5.3-1 acceptable because it is in accordance with the recommendations of RG 1.206, Regulatory Position 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.

5.3.3.5 Post Combined License Activities

There are no post COL activities related to this section.

5.3.3.6 Conclusion

The staff's finding related to information incorporated by reference is in NUREG 1966. The 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 E, Section VI.B.1, all nuclear safety issues relating to this section that were incorporated by reference have been resolved.

The staff also concluded that the information in STD SUP 5.3-1 meets the guidance in RG 1.206 and is therefore acceptable. Conformance with this guidance provides an acceptable basis for satisfying the requirements in 10 CFR Part 50, Appendix G. The staff also concluded that the information in STD SUP 5.3-1 meets the guidance of RG 1.206 and is therefore acceptable. Conformance with this guidance provides an acceptable basis for satisfying the requirements of 10 CFR Part 50, Appendix G.

5.4 <u>Reactor Coolant System Component and Subsystem Design</u>

5.4.1 Introduction

Section 5.4, "Reactor Coolant System Component and Subsystem Design," of the North Anna 3 COL FSAR, Revision 8, including the corresponding sections in the referenced ESBWR DCD. Specifically, the staff verified that the following sections of the ESBWR DCD contain information appropriate for incorporation by reference and that any supplemental information to be provided by the COL applicant is addressed in the COLA:

5.4.1 Reactor Recirculation System
5.4.2 Steam Generators (not applicable to the 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

5.4.2 Summary of Application

Section 5.4 of the North Anna 3 COL FSAR, Revision 8, incorporates by reference Section 5.4 of the certified ESBWR DCD, Revision 10, referenced in 10 CFR Part 52, Appendix E. In addition, in FSAR Section 5.4, the applicant provides the following:

Supplemental Information

- STD SUP 5.4-1

In FSAR Section 5.4.8, the applicant states 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 Section 5.4.12, the applicant states that the human factors analysis of 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 Section 5.4.12, the applicant states that operating procedures for the reactor vent system address considerations regarding when venting is and is not needed, including a variety of initial conditions that may require venting. Section 13.5 of the North Anna 3 COL FSAR addresses the development of operating procedures.

5.4.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is in NUREG–1966. In addition, the relevant requirements of the Commission regulations for the reactor coolant system component and subsystem design, and the associated acceptance criteria, are in Section 5.4 of the SRP.

5.4.4 Technical Evaluation

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

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

Supplemental Information

- STD SUP 5.4-1

In FSAR Section 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 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 Section 5.4.12, the applicant stated that the 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 DCD Chapter 18.

The staff found that this information is entirely incorporated into Chapter 18 of the North Anna 3 COL FSAR. The staff thus concludes that STD SUP 5.4-2 is acceptable.

- STD SUP 5.4-3

In FSAR Section 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.

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

5.4.5 Post Combined License Activities

There are no post COL activities related to this section.

5.4.6 Conclusion

The staff's finding related to information incorporated by reference is in NUREG–1966. The 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 E, 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 COLA to the relevant NRC regulations and the guidance in Section 5.4 of the SRP. The staff's review finds that the applicant has adequately addressed the supplemental information in accordance with NRC regulations. The supplemental information is therefore acceptable.

References

1. 10 CFR 50.2, "Definitions."

2. 10 CFR 50.55a, "Codes and standards."

3. 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light-water nuclear power reactors for normal operation."

4. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

5. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

6. 10 CFR Part 50, Appendix A, GDC 1, "Quality standards and records."

7. 10 CFR Part 50, Appendix A, GDC 14, "Reactor coolant pressure boundary."

8. 10 CFR Part 50, Appendix A, GDC 2, "Design bases for protection against natural phenomena."

9. 10 CFR Part 50, Appendix A, GDC 30, "Quality of reactor coolant pressure boundary."

10. 10 CFR Part 50, Appendix A, GDC 32, "Inspection of reactor coolant pressure boundary."

11. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

12. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

13. 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

14. 10 CFR Part 52, Appendix E, "Design Certification Rule for the ESBWR Design."

15. ASME Boiler and Pressure Code (BPVC).

16. ASME, BPVC, Section III, "Rules for Construction of Nuclear Facility Components," 2001 Edition, 2003 Addenda.

17. ASME, BPVC, Section III, Subsection N, "Division 1," 2001 Edition, 2003 Addenda.

18. ASME, BPVC, Section III, Subsection N, N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels Section XI, Division 1."

19. ASME, BPVC, Section III, Subsection N, N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves Section XI, Division 1."

20. ASME, BPVC, Section XI, "Rules for In-Service Inspection of Nuclear Power Plant Components," 2001 Edition, 2003 Addenda.

21. ASME, BPVC, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure"

22. ASME, BPVC, Section XI, IWA-2300, "Qualifications of Nondestructive Examination Personnel."

23. ASME, BPVC, Section XI, IWA-2314, "Certification and Recertification."

24. ASME, BPVC, Section XI, IWA-4540, "Pressure Testing of Classes 1, 2, and 3 Items."

25. ASME, BPVC, Section XI, IWA-5000, "System Pressure Tests."

26. ASME, BPVC, Section XI, IWB-2500, "Examination and Pressure Test Requirements."

27. ASME, BPVC, Section XI, IWB-5000, "System Pressure Tests."

28. ASME, BPVC, Section XI, Subsection IWA, "General Requirements."

29. ASME, BPVC, Section XI, Subsection IWB, "Requirements for Class 1 Components of Light-Water Cooled Plants."

30. ASME, OM Code 2001 including Addenda through 2003, "Code for Operation and Maintenance of Nuclear Power Plants."

31. ASME, OM Code N13, "Requirements for Extending Snubber In-Service Visual Examination Interval at LWR Power Plants."

32. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," 1982.

33. GEH ESBWR Design Control Document (DCD), Revision 10, April 2014 (ADAMS Accession No. ML14104A929).

34. NEDC-33441P, "GE Hitachi Nuclear Energy Methodology for the Development of Economic Simplified Boiling Water Reactor (ESBWR) Reactor Pressure Vessel Pressure-Temperature Curves," Revision 6, November 29, 2013 (ADAMS Accession No. ML13346A656 [Public Version NEDO-33441]).

35. NEDE-32983-P-A, Revision 2, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations." January 31, 2006 (ADAMS Accession No. ML072480121 [Public Version NEDO-32983-A]).

36. NRC GL 1992-001, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)," February 28, 1992 (ADAMS Accession No. ML031200626).

37. NRC GL 1992-001, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," May 19, 1995 (ADAMS Accession No. ML031070449).

38. NRC GL 1996-003, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996 (ADAMS Accession No. ML031110004).

39. NRC RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001 (ADAMS Accession No. ML010890301).

40. NRC RG 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," June 2003 (ADAMS Accession No. ML030730430).

41. NRC RG 1.47, Revision 16, "In-Service Inspection Code Case Acceptability, ASME Section XI, Division 1," October 2010 (ADAMS Accession No. ML101800536).

42. NRC RG 1.84, Revision 35, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," October 2010 (ADAMS Accession No. ML101800532).

43. NRC RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988 (ADAMS Accession No. ML003740284).

44. NRC RG 1.99, Revision 2, "Results of Periodic Review of Regulatory Guide 1.99," January 2014 (ADAMS Accession No. ML13346A001).

45. NRC SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," October 28, 2005, (ADAMS Accession Nos. ML052770225, ML052770257), and the related SRM, dated February 22, 2006 (ADAMS Accession No. ML060530316).

46. NRC Staff NUREG 0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," March 2007 (ADAMS Accession No. ML070660036).

47. NRC Staff NUREG-1966, "Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design," and its Supplement 1, April 2014 (ADAMS Accession Nos. ML14099A519, ML14099A522, ML14099A532, ML14100A187, ML14100A190, ML14100A194, ML14265A084).

48. NRC TSTF-419, "Analysis of NRC Position Regarding TSTF-363, 408, and 419," September 9, 2001 (ADAMS Accession No. ML012690166).