**ENCLOSURE 2**

**Clinch River Nuclear Site**
**Construction Permit Application**
**Enclosure 4, Exemptions and Variances**
**(Public Version)**

TENNESSEE
VALLEY
AUTHORITY

# Clinch River Nuclear Site Exemptions and Variances

Construction Permit Application – Enclosure 4

Revision 0

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## LIST OF ACRONYMS AND ABBREVIATIONS

| Acronym | Explanation |
|---|---|
| ASME | American Society of Mechanical Engineers |
| ASTM | ASTM International |
| BPVC | Boiler and Pressure Vessel Code |
| CFR | Code of Federal Regulations |
| CRN | Clinch River Nuclear |
| ESP | Early Site Permit |
| ESPA | Early Site Permit Application |
| FSAR | Final Safety Analysis Report |
| GEH | GE Hitachi Nuclear Energy |
| USNRC | U.S. Nuclear Regulatory Commission |
| SSAR | Site Safety Analysis Report |
| TVA | Tennessee Valley Authority |

# ENCLOSURE 4: EXEMPTIONS AND VARIANCES

## 1.0 Exemptions

### 1.0.1 Introduction

An exemption is required if information proposed in the construction permit application does not comply with one or more requirements of 10 Code of Federal Regulations (CFR) Part 50. Exemptions are submitted pursuant to 10 CFR 50.12, Specific exemptions. The U.S. Nuclear Regulatory Commission (USNRC) may grant exemptions which are, "authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security," as described in 10 CFR 50.12(a)(1), and show that the special circumstances in 10 CFR 50.12(a)(2) are present.

10 CFR 50.35 establishes the ability for the USNRC to issue a construction permit to an applicant as long as there is reasonable assurance that the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public. Issuance of a construction permit allows an applicant to begin construction activities but does not indicate approval of the safety of any design feature or specification unless the applicant specifically requests such approval, and such approval is incorporated in the permit (10 CFR 50.3(b)).

The BWRX-300 design supporting this construction permit application is preliminary and continues to progress to the final design. The exemptions presented in this enclosure are necessary to support approval of the final BWRX-300 design. If additional technical information is required to make a final determination on granting the exemption, allowances under 10 CFR 50.35(a)(2) are used to support the issuance of a construction permit, and the additional information is provided in a supplement to the construction permit application after approval or in the Final Safety Analysis Report (FSAR).

## 1.1 10 CFR 50 Appendix H, Reactor Vessel Material Surveillance Program Requirements

### 1.1.1 Summary of Requested Exemption

Pursuant to 10 CFR 50.12, TVA requests an exemption for the CRN Unit 1 from the portion of requirements of 10 CFR 50 Appendix H Paragraphs I and III that incorporate by reference ASTM International (ASTM) E185-82 “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels,” (Reference 1.1-1), and that require ASTM E185-82 be applied to the design of the surveillance program and withdrawal schedule for reactor vessels purchased after 1982. The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light-water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment. The surveillance program for the BWRX-300 design is based on ASTM E185-21, “Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels,” (Reference 1.1-2).

### 1.1.2 Regulatory Requirements

10 CFR 50 Appendix G, “Fracture Toughness Requirements,” refers to 10 CFR 50 Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” for test requirements and results of the surveillance program.

10 CFR 50 Appendix G, Paragraph I, “Introduction and Scope,” states in part:

> “This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.”

10 CFR 50 Appendix G Paragraph III, “Fracture Toughness Tests,” and Paragraph IV “Fracture Toughness Requirements” refer to 10 CFR 50 Appendix H for test requirements and results of the surveillance program respectively.

10 CFR 50 Appendix G Paragraph III.A states in part:

> “To demonstrate compliance with the fracture toughness requirements of section IV of this appendix, ferritic materials must be tested in accordance with the ASME Code and, for the beltline materials, the test requirements of appendix H of this part.”

10 CFR 50 Appendix G Paragraph IV.A states in part:

> “For the reactor vessel beltline materials, including welds, plates and forgings, the values of $RT_{NDT}$ and Charpy upper-shelf energy must account for the effects of neutron radiation, including the results of the surveillance program of Appendix H of this part.”

10 CFR 50 Appendix G Paragraph IV.A.1.a states in part:

> "Reactor vessel beltline materials must have Charpy upper-shelf energy[(1)] in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code."

Where footnote (1) states:

> "Defined in ASTME 185-79 (sic) and -82 which are incorporated by reference in Appendix H to Part 50."

10 CFR 50 Appendix H incorporates by reference ASTM E185-82 (Reference 1.1-1) in Paragraph I "Introduction," and requires ASTM E185-82 (Reference 1.1-1) be applied to the design of the surveillance program and withdrawal schedule for reactor vessels purchased after 1982 in Paragraph III, "Surveillance Program Criteria." 10 CFR 50 Appendix H also imposes report of test results in Paragraph IV, "Report of Test Results."

10 CFR 50 Appendix H Paragraph I states in part:

> "The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data will be used as described in Section IV of Appendix G to Part 50."

10 CFR 50 Appendix H Paragraph III.B states in part:

> "Reactor vessels that do not meet the conditions of paragraph III.A of this appendix must have their beltline materials monitored by a surveillance program complying with ASTM E185, as modified by this appendix."

10 CFR 50 Appendix H Paragraph III.B.1 states in part:

> "The design of the surveillance program and the withdrawal schedule must meet the requirements of the edition of the ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased; for reactor vessels purchased after 1982, the design of the surveillance program and the withdrawal schedule must meet the requirements of ASTM E 185-82."

10 CFR 50 Appendix H Paragraph III.B.4 states:

> "Optional provisions. As used in this section, references to ASTM E 185 include the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased through the 1982 edition."

10 CFR 50 Appendix H Paragraph IV.B states:

> "The report must include the data required by ASTM E 185, as specified in paragraph III.B.1 of this appendix, and the results of all fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions."

### 1.1.3 Technical Basis

A comparison of the two ASTM E185 editions is presented in Table 1.1-1 based on the comparative data presented in ASTM E185-21 (Reference 1.1-2). The most salient difference between the two editions is the withdrawal schedule. ASTM E185-82 (Reference 1.1-1) prescribes withdrawing capsules by fluence, temperature shifts or effective full power years. Given the relatively long life of BWRX-300 and small anticipated temperature shifts, the effective full power years intervals would likely be the most limiting condition. Under this withdrawal scheme, all but one capsule are withdrawn at 15 effective full power years or less. Given that the minimum operational lifetime of BWRX-300 is 60 years and industry trends aspire to extend plant operations to 80 years, using the surveillance capsule withdrawal schedule in ASTM E185-82 (Reference 1.1-1) is not practical.

Over the past 60 years, the embrittlement mechanisms in the reactor vessel and underlying causes have been extensively studied and the value of measuring reactor vessel temperature shifts at low fluences has been diminished as operating experiences have evolved. Controlling certain deleterious elements, namely copper and nickel, in the vessel forgings and in particular weld fillers metals have alleviated many problems associated with reactor vessel embrittlement experienced in some earlier nuclear power plants. Therefore, withdrawing surveillance capsules at evenly spaced intervals based on  the design life of the reactor, is a practical method to measuring and managing reactor vessel aging as compared to the approach prescribed in ASTM E185-82 (Reference 1.1-1) which is biased toward measuring low fluence temperatures shifts in a reactor vessel with an assumed operational lifetime of 40 years.

In addition to a more practical surveillance withdrawal schedule, the 2021 Edition of ASTM E185 (Reference 1.1-2) requires a standby capsule, Charpy V-Notch specimens, and compact tension fracture toughness specimens. ASTM E185-21 (Reference 1.1-2) also does not require heat affected zone specimens which have been proven to be of limited value. The addition of more, pertinent specimens provide more useful, direct data with lower uncertainties. The inclusion of compact tension specimens facilitate the use of the Master Curve method which could be a powerful tool to reduce measurement uncertainties and provide data to justify extended plant operational lifetimes.

The 2021 Edition of ASTM E185 (Reference 1.1-2) does not mandate actions which are contrary to the requirements of 10 CFR 50 Appendix H; therefore, ASTM E185-21 (Reference 1.1-2) and other requirements in 10 CFR 50 Appendix H are compatible and can be concurrently followed.

### 1.1.4 Regulatory Basis

Exemptions are submitted pursuant to 10 CFR 50.12 and must be authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security as described in 10 CFR 50.12(a)(1), and shown that the special circumstances in 10 CFR 50.12(a)(2) are present.

As required by 10 CFR 50.12(a)(1):

- The requested exemption is "*authorized by law*." This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The USNRC has authority under 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the proposed exemption is authorized by law.
- The requested exemption "*will not present an undue risk to the public health and safety.*" The differences between ASTM E185-21 (Reference 1.1-2) and ASTM E185-82 (Reference 1.1-1) are laid out in Table 1.1-1. This exemption does not affect the design, function, or operation of structures or plant equipment that are necessary for the safe operation of the plant. Therefore, this exemption does not present an undue risk to the public health and safety.
- The requested exemption is *"consistent with the common defense and security."* This exemption does not affect the design, function, or operation of structures or plant equipment that are necessary to maintain the secure status of the plant. This exemption has no effect on plant security or safeguards procedures. Therefore, this exemption is consistent with the common defense and security.

As required by 10 CFR 50.12(a)(2)(ii):

- *"Special circumstances are present whenever application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule."* The underlying purpose of 10 CFR 50 Appendix H is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light-water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment. As ASTM E185-21 (Reference 1.1-2) includes a sufficient monitoring program but is different from the program in ASTM E185-82 (Reference 1.1-1), special circumstances are present in accordance with 10 CFR 50.12(a)(2)(ii).

### 1.1.5 Conclusion

Based on the requirements in ASTM E185-21 (Reference 1.1-2), 10 CFR Appendix H, the exemption is sought based on 10 CFR 50.12 including the requirements of 10 CFR 50.12(a)(1) and 10 CFR 50.12(a)(2)(ii).

### 1.1.6 References

1.1-1 ASTM E185-82, “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels,” ASTM International, 1982.

1.1-2 ASTM E185-21, “Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels,” ASTM International, 2021.

**Table 1.1-1 Comparison of ASTM E185 1982 vs. 2021 Editions**

| ASTM E185 Criteria | 1982 Edition Requirement | 2021 Edition Requirement |
|---|---|---|
| Materials Monitored | Beltline base, weld, and HAZ metal with highest initial transition temperature, largest shift in transition temperature, decrease in USE, or most limiting for setting temperature/pressure limits. | Limiting base and weld materials.<br>If the limiting materials are outside of the beltline, limiting beltline base and weld shall also be included. |
| Number of Capsules | 3, 4, or 5<br>based on predicted temperature shift | 4 + 1 standby |
| Number of Unirradiated Specimens | Charpy – at least 18<br>Tension – at least 3 for base and weld only | Charpy – at least 15<br>Tension – at least 6 for base and weld only<br>Fracture toughness – at least 8 |
| Number of Irradiated Specimens per Exposure Set | Charpy – 12<br>Tension – 3 base and weld only | Charpy – 15<br>Tension – 3 base and weld only<br>Fracture toughness – 8 (limiting material) |
| Charpy Specimen Orientation | Normal to working direction<br>Notch perpendicular to surface | Normal to working direction<br>Notch perpendicular to surface |
| Withdrawal Schedule | At specified effective full-power years based on shifts in transition temp with last capsule as a standby | 1/4 Maximum Design Fluence<br>1/2 Maximum Design Fluence<br>3/4 Maximum Design Fluence<br>Maximum Design Fluence<br>(< 2 Maximum Design Fluence, Standby, Testing Not Required) |

## 2.0 Variances

### 2.0.1 Introduction

A *variance* is a plant-specific deviation from one or more of the site characteristics, design parameters, or terms and conditions of an Early Site Permit (ESP) or from the Site Safety Analysis Report (SSAR).

The following sections provide requests for variances from the site characteristics for the CRN ESP (Reference 2.0-1) and from the CRN SSAR (Reference 2.0-2). To support a decision whether to grant a variance, each variance request provides the technical justification for meeting the technically relevant regulatory acceptance criteria.

This CPA requests a variance where the CRN Preliminary Safety Analysis Report (PSAR) references the CRN ESP or the SSAR and: a) the CRN PSAR does not demonstrate that the design of the BWRX-300 falls within the ESP site characteristics; or b) the CRN PSAR does not demonstrate that the design of the BWRX-300 falls within the ESP (design) controlling parameters. Accordingly, this CPA includes the following requests for variances:

## 2.1 Variance: CRN ESP VAR 2.0-1 – Site Grade Level

***Request***

TVA requests to use a finished elevation for the remediated nuclear island excavation area of approximately 814.5 ft. In the CRN ESPA, the finished elevation was set at approximately 821 ft.

***Justification***

The CRN Early Site Permit Application (ESPA) was prepared using the Plant Parameter Envelope (PPE) conceptual design. The CRN-1 site layout for BWRX-300 design was optimized at a lower elevation. The new elevation of 814.5 ft. has been used in evaluations for the Enclosure 2, CRN PSAR.

## 2.2 Variance: CRN ESP VAR 2.0-2 – Ground Water Level

***Request***

TVA requests to use a maximum ground water level under foundation structures in power block area of 814.5 ft. In the CRN ESPA, the ground water level was 816.1 ft.

***Justification***

The CRN ESPA groundwater modeling software was Groundwater Vistas, Version 6.07, Build 10 (see CRN ESP VAR 2.4.12C-1). The more recent versions of the software, together with the site layout and grading, predicts the maximum groundwater to be at grade level of 814.5 ft.

## 2.3 Variance: CRN ESP VAR 2.0-3 – Single Unit Thermal Megawatts

***Request***

TVA requests to use the BWRX-300 SMR Technology with a thermal output of 870 MWt for CRN-1 at the CRN Site. A single unit thermal megawatts PPE value of 800 MWt was assumed for the CRN ESP and SSAR.

***Justification***

The BWRX-300 SMR design has an 870 MWt nominal output. The 70 MWt exceedance for a single unit is acceptable for the CRN Site because, as shown in the PPE comparison tables provided in Enclosure 2, CRN PSAR Section 2.0, CRN-1 meets all other Clinch River ESP-006 PPE performance requirements (radiological release, structural design, etc.) for both the site and a single unit.

## 2.4 Variance: CRN ESP VAR 2.1-1 – 2020 Census Data

***Request***

TVA requests to use 2020 census data to replace the 2010 census data.

***Justification***

TVA has updated the 2010 census data in the CRN SSAR with data collected in 2020. The refreshed data shows a more accurate picture of the demographics near the CRN Site. While the numbers have changed slightly, the trends remain the same.

## 2.5 Variance: CRN ESP VAR 2.2-1 – Nearby Industrial, Transportation, and Military Facilities.

***Request***

TVA requests to use refreshed data to include updates to nearby industrial, transportation, and military facilities.

***Justification***

TVA has refreshed the data in CRN SSAR Section 2.2 to include updates to nearby industrial, transportation, and military facilities. As provided in RG 1.78, chemicals stored or situated at distances greater than 5 mi from CRN Site do not need to be considered because if a release occurs at such a distance, atmospheric dispersion will dilute and disperse the incoming plume to such a degree that either toxic limits will never be reached or there would be sufficient time for the control room operators to take appropriate action. Although several of the facilities are located greater than 5 mi from the CRN Site, some of these were identified for further analysis because of their use of anhydrous ammonia, chlorine or sulfur dioxide. The update identified several industrial facilities, one major highway, four major roads, and two natural gas pipelines that are significant enough to be considered for further review (CRN PSAR Figure 2.2-1R).

CRN PSAR Table 2.2-6R shows the result of evaluations performed on potential hazards nearby and on the CRN Site. Except where noted, these evaluations concluded that potential accidents involving explosions, flammable vapor clouds, collisions with intake structures, and liquid spills do not pose a threat to the CRN Site.

## 2.6 Variance: CRN ESP VAR 2.4.12-1 – Groundwater Levels Model

***Request***

TVA requests to use the proposed BWRX-300 design to perform pre- and post-construction modeling to estimate the response of groundwater levels to the inclusion of the adjacent non-Reactor Building power block structures.

***Justification***

Pre- and post-construction modeling in the CRN ESPA was performed based on the PPE conceptual design. Using the BWRX-300 site layout results in a more representative groundwater model.

## 2.7 Variance: CRN ESP VAR 2.4.12C-1 – Groundwater Vistas, Version 8.19, Build 4.

***Request***

TVA requests to use an updated version of the pre- and post-processor groundwater modeling software, Groundwater Vistas, Version 8.19, Build 4.

***Justification***

The CRN ESPA groundwater modeling software was Groundwater Vistas, Version 6.07, Build 10. The more recent versions of the software result in a more representative model.

## 2.8 References

2.0-1 “Clinch River Nuclear Site Early Site Permit No. ESP-006,” U.S. Nuclear Regulatory Commission, December 2019 (ML19352D868) and Final Safety Evaluation Report (ML19162A157).

2.0-2 "Clinch River Nuclear Site Early Site Permit Application Part 2, Site Safety Analysis Report,” Tennessee Valley Authority, Revision 2, January 2019.