

United States Nuclear Regulatory Commission Official Hearing Exhibit	
In the Matter of:	Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3)
	ASLBP #: 07-858-03-LR-BD01
	Docket #: 05000247   05000286
	Exhibit #: NYS000494-00-BD01
	Admitted: 11/5/2015
	Rejected: Other:
Identified: 11/5/2015	
Withdrawn:	
Stricken:	

**NYS000494**  
**Submitted: June 9, 2015**

UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
OFFICE OF NUCLEAR REACTOR REGULATION  
OFFICE OF NEW REACTORS  
WASHINGTON, DC 20555-0001

October 14, 2014

**NRC REGULATORY ISSUE SUMMARY 2014-11**  
**INFORMATION ON LICENSING APPLICATIONS FOR FRACTURE TOUGHNESS**  
**REQUIREMENTS FOR FERRITIC REACTOR COOLANT PRESSURE BOUNDARY**  
**COMPONENTS**

**ADDRESSEES**

All holders of and applicants for an operating license or construction permit for a nuclear power reactor under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

All holders of and applicants for a power reactor combined license, standard design approval, or manufacturing license under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Reactors." All applicants for a standard design certification, including such applicants after initial issuance of a design certification rule.

**INTENT**

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to provide guidance to addressees on the scope and detail of information that should be provided in reactor vessel fracture toughness and associated pressure-temperature (P-T) limits licensing applications to facilitate staff review. This RIS requires no action or written response on the part of addressees.

**BACKGROUND INFORMATION**

The fracture toughness of reactor vessel materials may decrease with time in the presence of sufficient neutron irradiation. Therefore, NRC regulations require monitoring of reactor vessel material fracture toughness during plant operation as stated in 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements." P-T limits are used to restrict reactor operation so that reactor vessel material toughness margins are maintained based on the results of the 10 CFR Part 50, Appendix H reactor vessel material surveillance program. P-T limits define the pressure and temperature operating conditions that must be maintained to ensure adequate margins of safety to material fracture toughness limits.

10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," Section I, "Introduction and Scope," states the following:

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.

In particular, 10 CFR Part 50, Appendix G specifies P-T limits and minimum temperature requirements for the reactor vessel.

Ferritic materials of pressure-retaining components of the reactor coolant pressure boundary (RCPB) include the following:

- (1) reactor vessel forgings (e.g., shells, nozzles, and flanges)
- (2) plates and welds from which the reactor vessel was manufactured
- (3) ferritic materials in reactor coolant system piping, pumps, valves, and other pressure vessels

Of particular interest is the reactor vessel beltline, which is defined in 10 CFR Part 50, Appendix G as the region of the reactor vessel that “directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.” The beltline region experiences increased embrittlement over the operating period of the reactor vessel as a result of accumulated neutron radiation from the core.

Appendix H to 10 CFR Part 50 provides the requirements to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline resulting from exposure to neutron irradiation and the thermal environment. Appendix H to 10 CFR Part 50 states that no material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods that the peak neutron fluence at the end of the design life will not exceed  $1 \times 10^{17}$  neutrons/centimeter-squared ( $n/cm^2$ ) with energy greater than one million electron volts ( $E > 1$  MeV). Appendix G to 10 CFR Part 50 states, “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.” Furthermore, Section 2.2 of NUREG-1511<sup>1</sup>, “Reactor Pressure Vessel Status Report,” dated December 1994, states, “The NRC staff considered materials with a projected neutron fluence of greater than  $1.0E17$  neutrons per square centimeter ( $n/cm^2$ ) at end of license (EOL) to experience sufficient neutron damage to be included in the beltline.” Therefore, the beltline definition in 10 CFR Part 50, Appendix G is applicable to all reactor vessel ferritic materials with projected neutron fluence values greater than  $1 \times 10^{17}$   $n/cm^2$  ( $E > 1$  MeV), and this fluence threshold remains applicable for the design life as well as throughout the licensed operating period.

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<sup>1</sup> Available in Agencywide Documents Access and Management Systems (ADAMS) under Accession No. [ML082030506](#).

Changes in the fracture toughness properties of ferritic reactor vessel materials have been demonstrated to occur at neutron fluence levels as low as  $1 \times 10^{17}$  n/cm<sup>2</sup> (E > 1 MeV). This threshold neutron fluence level is therefore reflected in the surveillance program criteria set forth in 10 CFR Part 50, Appendix H, as described in NUREG-1511. As a result of changes made to the plant license, including power uprates, increased operating periods due to license renewal, modified fuel design, neutron fluence methodology updates, and fuel placement within the core, the physical region of the reactor vessel with fluence that exceeds this level can expand. The result is that changes in fracture toughness properties resulting from neutron embrittlement may occur in materials where the effects of radiation damage may not have been previously considered when developing the P-T limits for the vessel. In particular, this may be true for reactor vessel nozzle materials when the nozzles are positioned above or below the active core height.

Determination of the P-T limits for a plant in accordance with the requirements of 10 CFR Part 50, Appendix G considers several factors, including the initial properties and composition of the vessel materials, the accumulated neutron fluence for each material (and hence the neutron embrittlement of the material), and the stress levels applied to the materials resulting from operating loads and structural discontinuities, such as nozzles. In the development of the P-T limits, it is not sufficient to only consider the limiting reactor vessel material (generally considered to be the vessel shell materials with the highest reference temperature) and not consider the stress levels due to structural discontinuities. This is because the effects of structural discontinuities for a material with a lower reference temperature (such as a nozzle with a lower neutron fluence) may result in more restrictive allowable P-T limits than those for the vessel shell material limiting material with a higher reference temperature. Thus, the development of P-T limits for the reactor vessel must consider not only the vessel shell material with the highest reference temperature but also other vessel materials with structural discontinuities. In addition, all addressees should consider the effects of any replaced ferritic reactor coolant pressure boundary components on the adequacy of the P-T limits.

During recent license amendment submittals by licensees pertaining to P-T limits, the NRC staff requested additional information from the applicants due to a lack of sufficient demonstration that all ferritic materials of the reactor vessel were addressed. These requests for additional information are consistent with NRC's guidelines specified in NUREG-0800<sup>2</sup>, Revision 2, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock." To date, licensee responses to NRC requests for additional information have indicated that revised P-T limits have bounded all ferritic materials of the reactor vessel for the approved P-T limits periods<sup>3</sup>. Such responses are consistent with licensees' understanding that has been communicated to NRC in the past (e.g., page 5-8 of EPRI Report No. NP-5792-SRR1<sup>4</sup>, Revision 1, "Primer: Fracture Mechanics in the Nuclear Power Industry: Revision 1," dated May 1991).

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<sup>2</sup> Publicly available for download at <http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/>.

<sup>3</sup> An example request, as summarized in a safety evaluation for P-T limits for the Seabrook Station, Unit No. 1, may be found in ADAMS at Accession No. [ML120820510](https://www.adams.nrc.gov/).

<sup>4</sup> Publicly available for download at <https://www.epri.com>.

Finally, NRC Generic Letter 96-03<sup>5</sup>, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,” dated January 31, 1996, advises licensees that they may request a license amendment to relocate the P-T limit curves from their plant technical specifications (TS) to a pressure temperature limits report (PTLR) or a similar document, and states the following:

The methodology used to determine the P/T and LTOP [low temperature overpressure protection] system limit parameters must comply with the specific requirements of Appendices G and H to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR), be documented in an NRC-approved topical report or in a plant-specific submittal, and be incorporated by reference into the TS. Subsequent changes in the methodology must be approved by a license amendment; 10 CFR 50.59 does not apply.

Therefore, because PTLRs also document 10 CFR Part 50, Appendix G P-T limits, the clarifications made in this RIS also apply to those documents.

## **SUMMARY OF ISSUE**

Some recent licensee submittals pertaining to reactor vessel P-T limits have lacked sufficient demonstration that all ferritic materials of the reactor vessel were addressed in accordance with the requirements of 10 CFR Part 50, Appendix G. Specifically, some submittals regarding P-T limits provided technical bases analyzing only the reactor vessel material with the highest reference temperature without supporting details to demonstrate that the resulting P-T limits were bounding for all ferritic components of the reactor vessel. In determining P-T limits, reactor vessel materials with the highest reference temperature may not always produce the most limiting P-T limits because the consideration of stress levels from structural discontinuities (such as nozzles) may produce a lower allowable pressure. All addressees should ensure that P-T limits (including those implementing NRC-approved PTLR methodologies) sufficiently address all ferritic materials of the reactor vessel, including the impact of structural discontinuities, and address the impact of neutron fluence accumulation in accordance with the requirements of 10 CFR Part 50, Appendix G. Specifically, all ferritic components within the entire reactor vessel must be considered in the development of P-T limits, and the effects of neutron radiation must be considered for any locations that are predicted to experience a neutron fluence exposure greater than  $1 \times 10^{17}$  n/cm<sup>2</sup> (E > 1 MeV) at the end of the licensed operating period.

As specified in Sections I and IV.A of 10 CFR Part 50, Appendix G, ferritic RCPB components outside of the reactor vessel must meet the applicable requirements of ASME Code, Section III, “Rules for Construction of Nuclear Facility Components<sup>6</sup>”. Further guidance on these requirements may be found in NUREG-0800, Revision 2, Section 5.3.2, Paragraph II, “Acceptance Criteria.” Specifically, Items 1 through 4 of the Technical Rationale portion of that Paragraph discuss requirements for RCPB components.

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<sup>5</sup> Available in the ADAMS at Accession No. [ML031110004](#).

<sup>6</sup> Copies may be purchased from the American Society of Mechanical Engineers, Three Park Avenue, New York, NY 10016-5990; phone (212) 591-8500; fax (212) 591-8501; [www.asme.org](http://www.asme.org).

## **BACKFITTING AND ISSUE FINALITY**

This RIS clarifies what information addressees should submit in applications concerning reactor vessel fracture toughness and associated pressure-temperature limits. This RIS requires no action or written response from the licensees to whom the RIS is addressed. The RIS does not contain a new or changed NRC staff position or an interpretation of the applicable regulations that would constitute backfitting as defined in 10 CFR 50.109, or represent an inconsistency with applicable issue finality provisions in 10 CFR Part 52. Therefore, the NRC did not prepare a backfit analysis or further address the issue finality criteria in 10 CFR Part 52 with respect to this RIS.

## **FEDERAL REGISTER NOTIFICATION**

The NRC published a notice of opportunity for public comment on this RIS in the *Federal Register* (79 FR 21812) on April 17, 2014. The NRC staff received comments from the Electric Power Research Institute and Pressurized Water Reactor Owners Group. The final RIS reflects the NRC staff's consideration of these comments. The staff's resolution of these comments is publicly available under ADAMS Accession No. ML14192B402.

## **CONGRESSIONAL REVIEW ACT**

The NRC has determined that this RIS is not a rule as defined in the Congressional Review Act (5 U.S.C. §§ 801-808).

## **PAPERWORK REDUCTION ACT STATEMENT**

This RIS does not contain new or amended information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing requirements were approved by the Office of Management and Budget (OMB), approval number 3150-0011.

### Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

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Please direct any questions about this matter to the technical contacts listed below.

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**ADAMS Accession No.: ML14149A165**

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