NRC 2018-0041
10 CFR 50.90

August 30, 2018

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301

Subject: Response to Second Round Request for Additional Information Regarding Point Beach Nuclear Plant, Units 1 and 2, License Amendment Request for Risk-Informed Approach to Resolve Construction Truss Design Code Nonconformances, (EPID L-2017-LLA-0209)

References:

1. NextEra Energy Point Beach, LLC letter NRC 2017-0017, "License Amendment Request 278, Risk-Informed Approach to Resolve Construction Truss Design Code Nonconformances, " March 31, 2017 (ML17090A511)
2. NRC Letter, Point Beach Nuclear Plant, Units 1 and 2 – Request for Additional Information for Point Beach Nuclear Plant, Units 1 and 2, Regarding License Amendment Request to Resolve Nonconformances Relating to Containment Dome Truss (EPID L-2017-LLA-0209), January 31, 2018 (ML18025C043)
3. NextEra Energy Point Beach, LLC letter NRC 2018-0014, "Construction Truss License Amendment Request 278, Response to Request for Additional Information," April 12, 2018 (ML18102B164)
4. NextEra Energy Point Beach, LLC letter NRC 2018-0030, "Construction Truss License Amendment Request 278 Draft Updated Final Safety Analysis Report (UFSAR) Revision," May 29, 2018 (ML18149A466)
5. NRC e-mail, "Final - Second Round RAIs - Point Beach Nuclear Plant, Units 1 and 2 - License Amendment Request to Resolve Non-conformances Relating to Containment Dome Truss - EPID L-2017-LLA-0209," August 1, 2018 (ML18214A153)

In Reference 1, supplemented by References 3 and 4, NextEra Energy Point Beach, LLC (NextEra) submitted License Amendment Request (LAR) 278 to resolve legacy design code nonconformances associated with the Point Beach Units 1 and 2 containment dome construction trusses.

In Reference 5, NRC staff requested additional information to support its review of LAR 278. Attachment 1 to this letter provides the requested information.

On August 15, 2018 a teleconference was held between NRC staff and NextEra to discuss administrative aspects of the construction truss LAR commitments, license condition, and proposed Updated Final Safety Analysis Report (UFSAR) change. NextEra has clarified these items as discussed during the teleconference.

Following this discussion with the NRC, NextEra proposes to consolidate several previous commitments in a License Condition. Attachment 2 to this letter contains the proposed License Condition. Attachment 3 provides the comprehensive list of implementation items required to resolve the construction truss design code nonconformance. This list supersedes Enclosure 2 of Reference 1 and the additional commitment in Reference 3.

In Enclosure 2 of Reference 1, NextEra committed to a modification that would install a 24 hour pneumatic backup supply (nitrogen) to the power operated relief valves (PORVs) of Unit 1 and Unit 2, and the routing of the nitrogen supply lines so that all tubing outside the pressurizer cubicles is located below the containment operating floor, to provide protection from a postulated falling object. The Unit 1 modification was completed on December 19, 2017, and the Unit 2 modification was completed on July 6, 2017. Therefore, this previous commitment is not included in the attached list of implementation items.

Attachment 4 to this letter identifies the text to be deleted from the draft UFSAR Revision in Reference 4, and the final draft which supersedes the corresponding page submitted in Reference 4.

This response does not alter the conclusions in Reference 1 that the proposed change does not involve a significant hazards consideration pursuant to 10 CFR 50.92, and there are no significant environmental impacts associated with the change.

This letter contains no new or revised regulatory commitments.

Should you have any questions regarding this submittal, please contact Mr. Eric Schultz, Licensing Manager, at 920-755-7854.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 8/30, 2018

Sincerely,

Robert Craven
Site Director
NextEra Energy Point Beach, LLC

Attachment 1: Response to Request for Additional Information
Attachment 2: Markup of the Point Beach Units 1 and 2 Operating Licenses
Attachment 3: Implementation Items Required to Resolve the Construction Truss Design Code Nonconformances
Attachment 4: Proposed Draft UFSAR Revision

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
Public Service Commission of Wisconsin

Attachment 1

Response to Request for Additional Information

Request for Additional Information (RAI)-1.a-1

In RAI-1.a, the staff requested the licensee to clarify the changes to the licensing basis sought by the submittal. In its response to RAI-1.a (ADAMS Accession No. ML18102B164), the licensee provided a listing of proposed licensing basis changes including a tabular comparison of the current licensing basis and the proposed criteria for each change. Specifically for the containment structure concrete behind the steel liner, the licensee included two proposed criteria. One of those criteria permits localized concrete strain exceedance if "[l]ocalized exceedance of strain limits does not significantly reduce containment structure shell strength." The intent and purpose of that criterion is not evident to the staff based on Section 5.6 of Enclosure 5 of the submittal (ADAMS Accession No. ML17090A511), and the other criterion that requires the liner maintaining leak-tight integrity. Further, the term "significantly reduce" in the proposed criterion is not defined in the response.

Remove the cited criterion or clarify the term "significantly reduce" as well as justify the purpose of the cited criterion.

Response

The criteria listed in response to RAI-1.a are taken from the acceptance criteria noted within calculation 11Q0060-RPT-002, Methodology and Criteria to Determine the Strength Capacity of the Point Beach Nuclear Plant Containment Dome Trusses and Attached/Adjacent Components in Support of a Risk-Informed License Amendment Request . The acceptance criteria were established to support the evaluation of the containment liner and the containment structure that was performed in calculation 11Q0060-C-021, Analysis of Containment Liner for Contact Load from Containment Dome Trusses.

The evaluation performed in calculation 11Q0060-C-021 modeled the interaction between the truss structures and the containment liner using finite element analyses. The containment concrete structure was modeled integral with the containment liner (i.e., contact surface modeled as a composite structure). The two criteria developed, as listed in RAI-1.a (ADAMS Accession No. ML18102B164), were used in calculation 11Q0060-C-021 to assess the interaction between the truss structures and the containment structure and liner, and determine the maximum permissible contact load from the truss structures.

The first criterion listed (i.e., "The liner maintains leak-tight integrity as noted above.") was redundant to the acceptance criteria for the containment liner listed in the Table under the response to RAI-1.a. Specific guidance for permissible containment liner stress, with the intent of maintaining leak-tight integrity, was invoked from the ASME Boiler & Pressure Vessel Code, Section III, Division 1, 1983, Appendix F. This criterion was used to demonstrate acceptability of the steel liner.

The second criterion (i.e., "Localized exceedance of strain limits does not significantly reduce containment structure shell strength.") was established to justify the localized strain exceedance identified within the concrete structure under the maximum permissible contact load (52.8 kips per calculation 11Q0060-C-021). The acceptance criteria for the containment structure concrete noted that the proposed code of record would remain ACI 318-63, Building Code Requirements for

Reinforced Concrete. Calculation 11Q0060-C-021 identified that the resulting localized strain developed at the point of contact in the concrete would exceed code of record permissible values (i.e., 0.003 in/in). The second criterion for the containment structure concrete establishes acceptability for these localized points exceeding permissible strain, which is integral to the analysis demonstrating that leak tight integrity is maintained. The criteria is intended to limit the portion of concrete within the wall that exceeds the permissible strain to a negligible localized volume at the surface of the containment wall in contact with the liner, relative to the remaining concrete wall.

The extent of the concrete strain exceedance is noted in Section 4.2 of calculation 11Q0060-C-021, which identifies the volume of concrete as similar in shape to a paraboloid. The estimated concrete volume in which the permissible strain limit is exceeded is approximated to be a 3.5 in. diameter circular area in plan view (i.e., when looking towards the liner) extending into the concrete to a depth of 2.25 in. The identified volume remains confined in all directions by the remaining containment wall concrete and the containment liner. The thickness of the containment structure concrete wall at the point of contact is approximately 90 in . Considering a 1 ft. wide cross-section of the wall at the point of contact (at maximum area of localized strain exceedance), the portion of the local wall area exceeding permissible strain is approximately 0.73% of the total area (conservatively considered as 2.25 in x 3.5 in / 12 in x 90 in). The area exceeding permissible strain is considered to be negligible by comparison (i.e., less than 1% of local area), which reflects the intended criteria of demonstrating that the containment structure shell strength is not significantly reduced.

While the calculated area exceeding permissible strain for a local cross-section of the containment concrete structure was identified to be 0.73%, the peak liner contact interaction ratio (applied liner contact force divided by maximum permissible contact load) is calculated for Unit 2 in calculation 11Q0060-C-023, Thermal Evaluation of Unit 2 Containment Dome Truss in Support of Risk Informed LAR, as 0.78 (at one location). Therefore, the volume of concrete where the localized strain is exceeded is expected to be further reduced. The total number of truss locations which contact the liner and exceed permissible strain is limited, as identified within calculations 11Q0060-C-022 (7 trusses), Thermal Evaluation of Unit 1 Containment Dome Truss in Support of Risk Informed LAR, 11Q0060-C-023 (9 trusses), 11Q0060-C-024 (2 trusses), Seismic Evaluation of Unit 1 Containment Dome Truss in Support of Risk Informed LAR, and 11Q0060-C-025 (2 trusses) , Seismic Evaluation of Unit 2 Containment Dome Truss in Support of Risk Informed LAR. The calculated contact loads for Unit 1 thermal loads and both Unit 1 and Unit 2 seismic loads are bounded by the calculated peak contact load for Unit 2 thermal loading.

RAI-10.a-1

The response to RAI-10.a stated that the logic used to develop the event trees in Sections 5.2.1 through 5.2.3 of Enclosure 4 of the original submittal was completely revised to address staff's RAIs. Section F.5 of Enclosure 2 to the supplement dated April 12, 2018 (ADAMS Accession No. ML18102B164), provides discussion of the top events in the revised event tree for the demonstrably conservative analysis for the seismic initiator. The top event description for containment truss induced very small loss-of-coolant accident (LOCA) ("LOCA – CT induced very small LOCA") indicates that a very small LOCA would occur due to the in-core instrumentation seal table failure from a postulated falling truss member. It is expected that a very small LOCA would require Power Operated Relief Valves (PORVs) as a mitigation system but in the event tree in Section F.4 of Enclosure 2 to the supplement dated April 12, 2018, only High Head Injection (HHI) appears to be required for mitigation of "LOCA – CT induced very small LOCA." Such a modeling appears to be indicative of treatment of the event as a small LOCA, not a very small LOCA. Therefore, the logic model for the "LOCA – CT induced very small LOCA" appears to be different from the description.

Justify the modeling of the "LOCA – CT induced very small LOCA" event as a small LOCA.

Response

MAAP runs show that with AFW available a very small LOCA does not require PORVs for mitigation of core damage. If AFW is available for at least one steam generator, AFW provides sufficient RCS heat removal during a very small LOCA to preclude the need to open the PORVs for feed and bleed cooling.

The Internal events' small LOCA event tree does not question the availability of Feed and Bleed if AFW is available. MAAP runs documented in the PRA Success Criteria Notebook support the event tree logic.

This justifies the modeling of the "LOCA – CT induced very small LOCA" event as a small LOCA.

RAI-12.c-1

In RAI-12, the staff noted that the initiating event frequencies in the submittal (ADAMS Accession No. ML17090A511) for steam and feedwater line breaks inside containment were based on Electric Power Research Institute (EPRI) report 302000079, Revision 3, "Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments." The staff further noted that the initiating frequencies for the steam and feedwater lines in the EPRI report were for breaks outside containment and considered inapplicable to the risk assessment in the submittal.

The response to RAI-12.c referred to Section 2.3.2 of Enclosure 2 to the April 12, 2018, supplement (ADAMS Accession No. ML18102B164). According to the information in that section, the licensee derived the steam line break inside containment initiating event frequency from "SPAR [Standardized Plant Analysis Risk] Event Data and Results 2015 Parameter Estimation Update" whereas the licensee continued to use information from the aforementioned EPRI report for the feedwater line break inside containment (FLBIC) initiating event frequency. The staff notes that the "SPAR Event Data and Results 2015 Parameter Estimation Update" contains the initiating event frequency for FLBIC. The licensee has not provided a basis for the continued use of the information from the EPRI report for FLBIC even though the information does not appear to be applicable to the risk assessment in the submittal.

Quantitatively demonstrate the impact of using the initiating event frequency for FLBIC from "SPAR Event Data and Results 2015 Parameter Estimation Update" on the reported results in the context of this application.

Response:

Basis for the Continued Use of EPRI Information

EPRI data used to calculate feedwater pipe break frequency inside containment provides a more realistic but still conservative result that is consistent with the data sources used to calculate the SPAR frequency. The bases for this conclusion are summarized as follows:

1. **SAME DATA POINTS.** The EPRI data incorporates the two data points used in the "SPAR Event Data and Results 2015 Parameter Estimation Update." This data represents breaks outside containment that occurred in 1990 and 1991 at Millstone Unit 3 (moisture separator drain line) and Unit 2 (reheater drain line) respectively [reference NUREG/CR-5750, appendix I, section 1.5]. Also the SPAR data for "FWLB PWR FI", "Feedwater Line Break (PWR)", does not distinguish itself as inside or outside containment as do other initiating events in the grouping of High Energy Line Breaks.
2. **BROADER DATA SET.** The SPAR data scope is limited to only US Nuclear Plants. The EPRI data scope includes data for plants outside the US.
3. **REALISTIC PLANT SPECIFIC FREQUENCY**. EPRI provides pipe break frequency on a per foot basis, which allows calculating a plant specific event frequency based on the actual pipe lengths inside containment.
4. **CONSERVATIVE.** Although the EPRI data is limited to outside containment events, application of this data to characterize failure rates inside containment is conservative since the feedwater piping inside containment is more robust (Seismic Category I) than piping outside containment, and is subject to more rigorous inspections, and therefore expected to be more reliable.

Impact of Using the Initiating Event Frequency for FLBIC from "SPAR"

The PWR Feedwater Line Break Pipe frequency, INIT-FBIC, was changed to the INL 2015 IE update value of 1.50E-03, versus 2.50E-05 used in the PRA evaluation. The thermal bounding ΔCDF increases from 4.8E-08 to 8.6E-08. If 50% of piping is assumed to be inside containment and the failure rate is adjusted by 50%, the thermal result increases from 4.8E-08 to 6.7E-08. In both cases, the results remain within the RG 1.174 guidance.

RAI-15.a-1

In its response to RAI-15.a, the licensee provided the target assessment document (ADAMS Accession No. ML18102B173), supporting the supplement dated April 12, 2018, Section 3.5.4 of the target assessment document, discusses the likelihood of the main steam vent line failure due to postulated falling truss member. However, Attachment G of Enclosure 2 to the supplement dated April 12, 2018 (ADAMS Accession No. ML18102B164), evaluates the failure probability of the main steam lines for use in the demonstrably conservative analysis for the seismic initiator. It is unclear how the main steam vent line failure due to postulated falling truss member is reflected in the demonstrably conservative analysis.

Describe and justify how the main steam vent line failure due to postulated falling truss member was considered in the demonstrably conservative analysis for the seismic initiator.

Response

The demonstrably conservative seismic analysis considers two consequences related to puncturing or shearing of the main steam vent line due to being struck by a postulated falling construction truss member; its impact on containment integrity, and the impact of a potential steam vent line break on accident mitigation.

Containment Integrity

PBN-BFJR-17-019, Revision 1, POINT BEACH UNITS 1& 2 CONSTRUCTION TRUSS PRA EVALUATION Section 2.1.3, evaluates the integrity of the vent line penetration due to a postulated falling truss member in regards to its impact on containment integrity. The evaluation concluded that valves located outside containment would isolate a perforation of the steam vent lines inside containment.

Accident Mitigation

Plant EOPs direct operators not to isolate a faulted steam generator if it is needed for RCS cooldown. MAAP analyses confirm that in the event of a steam line vent rupture the faulted steam generator remains capable of cooling the RCS for 24 hours. The AFW system can provide sufficient water to offset leakage from the break and maintain steam generator water level with sufficient margin to cool the RCS. This conclusion also applies to a rupture of both steam vent lines.

The CT seismic model used in the PRA evaluation assumed the faulted steam generator was failed and not recoverable. This conservative characterization of the main steam line break in the PRA seismic model, along with other conservative assumptions, are bounding and therefore precludes the need to consider the probability and consequences of a steam line vent rupture. In addition, sensitivity analyses in the PRA evaluation, section 9.1.3, Damage Probabilities, show that ΔCDF and ΔLERF values remain within RG 1.174 Region III when main steam line damage probabilities are significantly increased.

Attachment 2

Markup of the Point Beach Units 1 and 2 Operating Licenses

(3 pages follow)

INSERT

Containment Building Construction Truss

NextEra Energy Point Beach shall complete the items listed in Attachment 3 of NextEra Energy Point Beach letter NRC 2018-0041, "Response to Second Round Request for Additional Information Regarding Point Beach Nuclear Plant, Units 1 and 2, License Amendment Request for Risk-Informed Approach to Resolve Construction Truss Design Code Nonconformances, (EPID L 2017-LLA-0209)," dated August 30, 2018, in accordance with the schedule included in that Attachment.

G. Secondary Water Chemistry Monitoring Program

NextEra Energy Point Beach shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;
2. Identification of the procedures used to quantify parameters that are critical to control points;
3 Identification of process sampling points;
4. Procedure for the recording and management of data;
5. Procedures defining corrective actions for off control point chemistry condition; and
6. A procedure for identifying the authority responsible for the interpretation of the data, and the sequence and timing of administrative events required to initiate corrective action.

H. The licensee is authorized to repair Unit 1 steam generators by replacement of major components. Repairs shall be conducted in accordance with the licensee's commitments identified in the Commission approved Point Beach Nuclear Plant Unit No. 1 Steam Generator Repair Report dated August 9, 1982 and revised March 1, 1983 and additional commitments identified in the staff's related safety evaluation.

I. ~~Deleted~~ INSERT

J. Deleted

K. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

L. Mitigation Strategy

Strategies shall be developed and maintained for addressing large fires and explosions that include the following key areas:

1. Fire fighting response strategy with the following elements:

   a. Pre-defined coordinated fire response strategy and guidance
   b. Assessment of mutual aid fire fighting assets
   c. Designated staging areas for equipment and materials
   d. Command and control
   e. Training of response personnel

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6. A procedure for identifying the authority responsible for the interpretation of the data, and the sequence and timing of administrative events required to initiate corrective action.

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    d. Command and control
    e. Training of response personnel

Attachment 3

Implementation Items Required to Resolve the Construction Truss Design Code Nonconformances

(3 pages follow)

IMPLEMENTATION ITEMS

The following table identifies the regulatory items that must be completed to support implementation of the license amendment to resolve the construction truss design code nonconformance.

IMPLEMENTATION ITEMS	COMPLETION DATE
1. <u>Unit 1 Spatial Clearance</u> Upon approval of LAR 278, a modification will be implemented to the Unit 1 construction truss to improve clearance between the truss and the containment liner. The modification includes trimming the top chord at the first panel point of six trusses, specifically trusses 1, 2, 3, 7, 8, and 15, as identified in NextEra Energy Point Beach, LLC letter NRC 2018-0014, "Construction Truss License Amendment Request 278, Response to Request for Additional Information," April 12, 2018 (ML18102B164). The spatial clearance modification requires approval of LAR 278 prior to implementation as the small amount of material removal modifies the sectional properties of the construction truss, which impacts the available seismic margin. No clearance modifications are required for Unit 2.	Contingent upon receipt of the NRC Safety Evaluation for LAR 278 by March 19, 2019, the modification will be implemented during the Unit 1 refueling outage in the fall of 2020.
2. <u>Unit 1 Containment Spray Pipe Support</u> Unit 1 containment spray pipe support SI-301R-1-H202 will be modified to achieve additional seismic capacity. The modification will increase the size of the support's U-bolt diameter. No containment spray pipe support modifications are required for Unit 2.	Contingent upon receipt of the NRC Safety Evaluation for LAR 278 by March 19, 2019, the modification will be implemented during the Unit 1 refueling outage in the fall of 2020.

IMPLEMENTATION ITEMS	COMPLETION DATE
3. Unit 1 and Unit 2 New Seismic Operating Limits NextEra will implement new seismic operating limits applicable to Unit 1 and Unit 2 to maintain stresses in the construction trusses within elastic stress limits. The applicable bounding limits are: Horizontal: 0.05g Peak Ground Acceleration Vertical: 0.04g Peak Ground Acceleration The existing seismic monitors will be used to detect the new operating seismic limits. Site procedures will be revised to initiate actions to commence a controlled dual Unit backdown to hot shutdown upon reaching the specified peak ground acceleration limits. The procedure will specify that the Units may be reduced in power individually in series, rather than concurrently, to ensure station staff have the proper resources and focus on each Unit for maximum safety and operational excellence during this off-normal operation. The procedure will also initiate inspection and/or evaluation actions for the construction trusses, equipment supported by the trusses, and the containment liners.	Contingent upon receipt of the NRC Safety Evaluation for LAR 278 by March 19, 2019, procedure revisions will be performed during the first quarter of 2021, following implementation of the modifications noted in Items 1, 2, and 4.
4. Unit 2 PORV Control Cable Protection NextEra will implement a modification to protect the control cables for Unit 2 pressurizer power operated relief valve (PORV) 2RC-431C, and associated block valve, 2RC-515, from a postulated falling object.	Contingent upon receipt of the NRC Safety Evaluation for LAR 278 by March 19, 2019, the modification will be implemented during the Unit 2 refueling outage in the spring of 2020.

IMPLEMENTATION ITEMS	COMPLETION DATE
5. <u>Unit 1 and Unit 2 New Thermal Operating Limits</u> NextEra will implement new thermal operating limits applicable to Unit 1 and Unit 2 construction trusses and attached components for any thermal excursion or occurrence resulting in exceeding elastic stress limits. The proposed new operational containment temperature limits are: Unit 1: 227°F containment atmospheric temperature Unit 2: 236°F containment atmospheric temperature The station implementation of these new limits will include taking the affected Unit offline and performing an inspection and/or evaluation, as necessary, to confirm that the construction truss' future stability during a postulated design basis accident was not compromised by the thermal occurrence. Detection of the thermal occurrence and initiation of actions to address the new limiting values will be addressed in site procedures. The site procedures will be revised to initiate inspection and/or evaluation actions prior to Unit startup for the affected Unit's construction truss, equipment supported by the truss (as necessary), and the containment liner. No new instrumentation is required to support this change. Existing station instrumentation will be utilized for containment temperature identification.	Contingent upon receipt of the NRC Safety Evaluation for LAR 278 by March 19, 2019, procedure revisions will be performed during the first quarter of 2021, following implementation of the modifications noted in Items 1, 2, and 4.

Attachment 4

Proposed Draft UFSAR Revision

(2 pages follow)

- In-structure seismic response spectra is determined through soil-structure interaction (SSI) analysis. Ground motion time histories shall meet Section 2.4 of ASCE/SEI 43-05 with the limitations identified in NUREG/CR-6926.
- Damping for the truss structures is 7%.Damping for the containment spray piping attached to the truss structures is 4%.AISC N690-1994(R2004), American National Standard Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities, is used as the code for evaluating the truss structural components, using the increased allowable stresses for dead load and seismic load combinations and dead load and thermal load combinations.
  - For truss members of the upper and/or lower chord that do not meet the limits of AISC N690-1994(R2004), the maximum permissible strain is limited to 1.5% for combined axial and flexure or flexure only.
- The allowable contact load on the containment liner is based on guidance in ASME B&PV Code, Section III, Division 1, 1983, Appendix F:
  - The allowable load under seismic or design basis accident loads is the minimum of the load that develops a maximum primary stress intensity of 0.9Su (ultimate strength) and 2/3 of the maximum sustainable load.
  - Liner integrity for applied cyclic loading is assessed by comparing the accumulation in strains and the change in strains between cycles, in combination with the fatigue curve from Figure 1-9.1 of ASME Boiler and Pressure Vessel Code, Section III, 1983.
  - Localized exceedance of permissible concrete strain truss contact points, with an allowable limit of 0.003 in/in per ACI 318-63.
  - The concrete compressive strength is based on the compressive strength from test data as permitted in ACI 318-63.

Note: The above criteria are limited in application to the truss structures and adjacent or supported equipment near the truss structures which was used to resolve the nonconformances addressed in Reference 21.

All of the equipment supported by the truss structures, such as the containment spray piping, PACV piping, associated pipe supports, VNCC ductwork, lighting, and associated conduits shall use the design code of record for evaluation.

The above criteria was used to calculate the seismic fragility and a thermal probability of failure for the trusses and attached components for use in the probabilistic risk assessment (Reference 33), based on which the trusses were accepted. The analyses evaluated Unit 1 assuming completion of a modification to trim the truss structures at six designated locations to increase clearance between the trusses and the containment liner, and evaluated Unit 2 in the as-found/post-construction condition with no modifications pending. Moving forward, future evaluations/modifications for the trusses and/or attached components shall follow the above criteria. ~~As long as the above criteria are met, the probabilistic risk assessment remains valid. Changes to the above criteria that reduce the seismic fragility and thermal probability of failure shall require a probabilistic risk assessment.~~

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  - The concrete compressive strength is based on the compressive strength from test data as permitted in ACI 318-63.

Note: The above criteria are limited in application to the truss structures and adjacent or supported equipment near the truss structures which was used to resolve the nonconformances addressed in Reference 21.

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