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Docket Number 50-346

License Number NPF-3

Serial Number 2675

November 9, 2000

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Subject: License Amendment Application to Relocate Technical Specification 3/4.4.9.2, "Reactor Coolant System – Pressurizer," to the Technical Requirements Manual (License Amendment Request No. 98-0017)

Ladies and Gentlemen:

Enclosed is an application for an amendment to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Operating License Number NPF-3, Appendix A, Technical Specifications. The proposed change involves: Technical Specification (TS) 3/4.4.9.2, "Reactor Coolant System – Pressurizer."

The proposed change would relocate TS 3/4.4.9.2 to the DBNPS Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM). A corresponding change to the TS Index is also proposed.

The TRM is a DBNPS-controlled document, which has been incorporated into the USAR. Changes to the USAR are controlled in accordance with Section 50.59 of Title 10 of the Code of Federal Regulations. Relocation of TS 3/4.4.9.2 to the USAR TRM will therefore allow future changes to these requirements to be made in accordance with the provisions of Section 50.59.

The DBNPS requests that this license amendment application be approved by the NRC and the amendment issued by May 15, 2001.

A001

If you have any questions regarding this application, please contact Mr. David H. Lockwood, Manager - Regulatory Affairs, at (419) 321-8450.

Very truly yours,

FWK/laj

Enclosures

cc: J. E. Dyer, Regional Administrator, NRC Region III
S. P. Sands, NRC/NRR Project Manager
D. J. Shipley, Executive Director, Ohio Emergency Management Agency, State of Ohio (NRC Liaison)
K. S. Zellers, NRC Region III, DB-1 Senior Resident Inspector
Utility Radiological Safety Board

APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NUMBER NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

Attached is the requested change to the Davis-Besse Nuclear Power Station, Unit Number 1 Facility Operating License Number NPF-3. Also included is the Safety Assessment and Significant Hazards Consideration.

The proposed change (submitted under cover letter Serial Number 2675) concerns Appendix A, Technical Specification:

Page V	Index
3/4.4.9.2	Reactor Coolant System - Pressurizer

I, Guy G. Campbell, state that (1) I am Vice President - Nuclear of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification on behalf of the Toledo Edison Company and The Cleveland Electric Illuminating Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

By: Guy G. Campbell

Guy G. Campbell, Vice President - Nuclear

Affirmed and subscribed before me this 9th day of November, 2000.

Nora L. Flood

Notary Public, State of Ohio - Nora L. Flood

My Commission expires September 4, 2002.

The following information is provided to support issuance of the requested change to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1, Facility Operating License Number NPF-3, Appendix A, Technical Specifications. The proposed change involves Technical Specification (TS) 3/4.4.9.2, "Reactor Coolant System – Pressurizer." A corresponding change to the TS Index is also proposed.

A. Time Required to Implement: The License Amendment associated with this license amendment application is to be implemented within 120 days after NRC issuance.

B. Reason for Change (License Amendment Request Number 98-0017):

   The proposed change would relocate TS 3/4.4.9.2 to the DBNPS Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM). A corresponding change to the TS Index is also proposed.

   Relocation of TS 3/4.4.9.2 to the USAR TRM will allow future proposed changes to the requirements to be evaluated in accordance with 10 CFR 50.59 and implemented if prior NRC approval is not required.

   The proposed change is in accordance with the requirements of 10 CFR 50.36 and the relocation guidance provided in the NRC's "Final Policy Statement on TS Improvements for Nuclear Reactors," dated July 22, 1993. The proposed change is also in accordance with the guidance provided by the improved "Standard Technical Specifications - Babcock and Wilcox Plants," NUREG-1430, Revision 1.

C. Safety Assessment and Significant Hazards Consideration: See Attachment.

Docket Number 50-346
License Number NPF-3
Serial Number 2675
Attachment 1

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION

FOR

LICENSE AMENDMENT REQUEST NUMBER 98-0017

(17 Pages Follow)

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION FOR LICENSE AMENDMENT REQUEST NUMBER 98-0017

TITLE:

Proposed Modification to the Davis-Besse Nuclear Power Station (DBNPS) Unit Number 1, Facility Operating License NPF-3, Appendix A Technical Specifications (TS) to Re-Locate TS 3/4.4.9.2, "Reactor Coolant System – Pressurizer," to the Technical Requirements Manual.

DESCRIPTION:

The purpose of this License Amendment Request is to relocate the Davis-Besse Nuclear Power Station (DBNPS) Unit Number 1 Operating License NPF-3, Appendix A, Technical Specification (TS) 3/4.4.9.2, "Reactor Coolant System – Pressurizer," from the TS to the Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM). This change is consistent with 10 CFR 50.36, "Technical Specifications," and the guidance provided by the NRC's "Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (58 FR 39132, dated July 22, 1993) as explained later under the "Effects on Safety" Section. The proposed change is also consistent with the NRC's NUREG-1430, Revision 1, "Standard Technical Specifications – Babcock and Wilcox Plants," in that the TS content being relocated from the DBNPS TS is not contained in the Standard Technical Specifications. A corresponding change to the TS Index page V that removes reference to the TS 3/4.4.9.2, "Pressurizer," is an administrative change. This proposed change does not include TS 3/4.4.4, "Reactor Coolant System – Pressurizer," that requires an operable pressurizer with a steam bubble and a water level between 45 and 305 inches during plant Modes 1 (Operation) and 2 (Startup).

The DBNPS TS Bases Section 3/4.4.9 refers to the Pressure/Temperature Limits of the Reactor Coolant System and is intended to apply to LCO 3/4.4.9.1, "Reactor Coolant System – Pressure/Temperature Limits – Reactor Coolant System," and LCO 3/4.4.9.2, "Reactor Coolant System – Pressurizer." However, it contains no information on the Pressurizer and therefore, no TS Bases content will be relocated to the TRM.

Relocating this TS to the USAR TRM will provide for future changes to be made by the DBNPS staff under the regulatory controls of 10 CFR 50.59, "Changes, Tests, and Experiments." The DBNPS staff will be allowed to control and evaluate proposed changes to this TS without the need for the DBNPS and NRC staffs to process a License Amendment Request, when such proposed changes do not fall under the category requiring prior NRC approval. Changes made to the USAR TRM will be submitted to the NRC in accordance with the USAR revision requirements of 10 CFR 50.71(e). The relocation of this TS to the USAR TRM will be completed no later than the implementation of the NRC-approved License Amendment that allows for its removal from the TS.

SYSTEMS, COMPONENTS, AND ACTIVITIES AFFECTED:

The proposed change affects the administrative location of the temperature limits and surveillance requirements for the Reactor Coolant System Pressurizer.

FUNCTIONS OF THE AFFECTED SYSTEMS, COMPONENTS, AND ACTIVITIES:

The overall function of the Operating License, Appendix A TS is to impose those conditions of limitation upon reactor operation necessary to preserve the validity of the DBNPS USAR Design Bases analyses. The purpose of TS 3/4.4.9.2 is to impose limitations on the pressurizer heatup and cooldown, and spray water temperature differential to assure that the pressurizer remains within the design criteria assumed for its fatigue analysis. The pressurizer is described in Section 5.5.10, "Pressurizer," of the DBNPS USAR.

The pressurizer is an American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section III Class A seismic category 1 vessel. The code of construction was Section III-1968 including the summer 1968 Addenda.

The pressurizer is a vertical-cylindrical vessel that is connected to the reactor outlet piping by the surge line piping. The electrically heated pressurizer establishes and maintains reactor coolant pressure within prescribed limits. It provides a surge chamber and a water reserve to accommodate changes in reactor coolant volume during operation.

Replaceable electric heater bundles in the lower section and a water spray nozzle in the upper section maintain the steam and water at the saturation temperature corresponding to the desired Reactor Coolant System pressure. During outsurges, as the pressure in the Reactor Coolant System decreases, some of the water in the pressurizer flashes to steam to maintain pressure. Electric heaters are actuated to restore the normal operating pressure. During insurges, as pressure in the Reactor Coolant System increases, a water spray from an inlet line condenses steam and thus reduces pressure.

The total stresses resulting from thermal expansion, pressure, and mechanical and seismic loadings are considered in the design of the pressurizer. The total stresses expected in the pressurizer are within the maximum allowed by the ASME B&PV Code, Section III. Connections have thermal sleeves when required, to limit stresses from thermal shock to acceptable values.

EFFECTS ON SAFETY:

The purpose of the Operating License, Appendix A, Technical Specifications is to impose those conditions or limitations upon reactor operation necessary to preserve the validity of the results of USAR Design Basis analyses. Section 182a of the Atomic Energy Act of 1954, as amended, requires applicants for nuclear power plant operating licensees to include TS as part of the license. The NRC regulatory requirements related to the content of the TS are set forth in 10 CFR 50.36, "Technical Specifications." This regulation requires that the TS include items in five specific categories, including: (1) Safety Limits, Limiting Safety System Settings, and Limiting Control Settings; (2) Limiting Conditions for Operation; (3) Surveillance Requirements; (4) Design Features; and, (5) Administrative Controls. However, 10 CFR 50.36 does not specify the particular requirements to be included in a plant's TS.

The NRC has provided guidance for the content of TS in its "Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," dated July 22, 1993, (reference: 58 FR 39132) in which the NRC indicated that compliance with the Final Policy Statement satisfies Section 182a of the Atomic Energy Act of 1954, as amended. In particular, the NRC indicated that certain items may be relocated from the TS to licensee-controlled documents since TS are to be reserved for those matters to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

The NRC's Final Policy Statement recognized that implementation of the policy would result in the relocation of some existing TS requirements to licensee-controlled documents such as the USAR. Those items relocated to the USAR would, in turn, be controlled in accordance with the requirements of 10 CFR 50.59. This regulation provides criteria to determine when facility or operating changes planned by a licensee require prior NRC approval in the form of a license amendment.

Four criteria were published by the NRC in the Final Policy Statement. The policy established that any TS requirements which did not meet any of the four criteria could be proposed for relocation to licensee-controlled documents such as the USAR. These criteria were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36, dated July 19, 1995, (reference: 60 FR 36953). The TS proposed for relocation to the USAR TRM does not meet these criteria for inclusion in the DBNPS TS and is, therefore, proposed for relocation. The removal of this TS is also consistent with its absence in the NRC's guidance document NUREG-1430, Revision 1, dated April 1995, "Standard Technical Specifications - Babcock and Wilcox Plants."

This TS is proposed for relocation to the DBNPS USAR TRM. This TS will be incorporated into the USAR TRM with the same content that it possessed as part of the Operating License, modified to conform to the TRM format and style.

The TS proposed for relocation is evaluated below with respect to the four criteria of 10 CFR 50.36(c)(2)(ii) used in determining whether a particular item is required to be retained in the TS, or may be relocated to other licensee-controlled documents. The four criteria that require retaining a particular item in TS are as follows:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

As described in the Federal Register notice (Reference: 58 FR 39132) of the NRC's Final Policy Statement, the purpose of this criterion is to ensure that the TS control those instruments specifically installed to detect excessive Reactor Coolant System leakage. The Federal Register notice states that this criterion should not be interpreted to include instrumentation to detect precursors to reactor coolant pressure boundary leakage or instrumentation to identify the source of actual leakage (e.g., the loose parts monitor).

2. A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

As described in the Final Policy Statement's Federal Register notice, the purpose of this criterion is to capture those process variables that have initial values assumed in the Design Basis Accident and Transient Analyses, and which are monitored and controlled during power operation. As long as these variables are maintained within the established values, risk to the public safety is presumed to be acceptably low. This criterion also includes active design features (e.g., high pressure/low pressure system valves and interlocks) and operating restrictions needed to preclude unanalyzed accidents and transients.

3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

As described in the Final Policy Statement's Federal Register notice, the purpose of this criterion is to capture only structures, systems, and components that are part of the primary success path of a safety sequence analysis. Also captured by this criterion, are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path for a particular mode of operation does not include backup and diverse equipment (e.g., safety valves which are a backup to low temperature overpressure relief valves during cold shutdown.).

4. A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

As described in the Final Policy Statement's Federal Register notice, the purpose of this criterion is to retain in TS those requirements that the probabilistic safety assessment or operating experience shows as significant to public health and safety.

The Pressurizer temperature limits are used to assure the Pressurizer remains within the design criteria assumed for its fatigue analysis. An evaluation of the Pressurizer temperature limits with respect to the four criteria of 10 CFR 50.36(c)(2)(ii) follows:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Pressurizer temperature limits are not installed instrumentation used to detect degradation of the RCS pressure boundary. Therefore, these limits do not meet Criterion 1 for inclusion in the TS.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

This Technical Specification specifies limits on process variables consistent with boundaries used in the structural analyses. These limits, however, do not reflect initial condition assumptions in the Design Bases Accident or Transient Analysis. Therefore, these limits do not meet Criterion 2 for inclusion in the TS.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Pressurizer temperature limits are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient. TS 3/4.4.4, "Reactor Coolant System – Pressurizer," which requires the pressurizer to be operable, will be retained in TS and is unaffected by this change. Therefore, these limits do not meet Criterion 3 for inclusion in the TS.

Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The DBNPS Probabilistic Risk Assessment (PRA) does not address Pressurizer temperature limits, and does not evaluate normal heatup and cooldown cycles. Significant overcooling events were analyzed for Babcock & Wilcox plants to determine the probability of reactor vessel failure due to pressurized thermal shock (PTS) (Topical Report BAW – 1791, "BWOG Probabilistic Evaluation of Pressurized Thermal Shock"). BAW-1791 determined that PTS is not a dominant contributor to risk. Furthermore, TS 3/4.4.4, "Reactor Coolant System -Pressurizer," will be retained in the TS and is unaffected by this proposed change. Therefore, the Pressurizer temperature limits do not meet Criterion 4 for inclusion in the TS.

In summary, the Pressurizer temperature limits can be relocated from the TS to the USAR TRM with no adverse effect on nuclear safety. Future changes to these limits will be controlled under the 10 CFR 50.59 process.

SIGNIFICANT HAZARDS CONSIDERATION:

The Nuclear Regulatory Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazard exists due to a proposed amendment to an Operating License for a facility. A proposed amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed changes would: (1) Not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) Not create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Not involve a significant reduction in a margin of safety. The Davis-Besse Nuclear Power Station has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, (DBNPS) Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no change is being made to any accident initiator. No previously analyzed accident scenario is changed, and initiating conditions and assumptions remain as previously analyzed.

    The proposed change would relocate TS 3/4.4.9.2 "Reactor Coolant System – Pressurizer," to the DBNPS Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM). TS 3/4.4.9.2 provides temperature limits for the Pressurizer based on its fatigue analysis design criteria. The proposed change to remove this TS is in accordance with 10 CFR 50.36 and the NRC's "Final Policy Statement on TS Improvements for Nuclear Power Reactors," dated July 22, 1993. The proposed change is also consistent with the improved "Standard Technical Specifications – Babcock and Wilcox Plants, "NUREG-1430, Revision 1. A corresponding change to the TS Index page V that removes reference to the Pressurizer Pressure/Temperature Limits is an administrative change.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed change does not affect accident conditions or assumptions used in evaluating the radiological consequences of an accident. The proposed change does not alter the source term, containment isolation or allowable radiological releases.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new failure mode is introduced since the proposed relocation does not involve a modification or change in operation of any plant systems, structures, or components. No new or different types of failures or accident initiators are introduced by the proposed change.

3. Not involve a significant reduction in a margin of safety because the proposed change is administrative in nature, consisting of the relocation of certain TS requirements into a licensee-controlled document, and has no bearing on the margin of safety which exists in the present TS or Updated Safety Analysis Report (USAR).

CONCLUSION:

Based on the above, the Davis-Besse Nuclear Power Station has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

ATTACHMENT:

Attached is the proposed marked-up change to the Operating License.

REFERENCES:

1. DBNPS *Updated Safety Analysis Report* (USAR) through Revision 21.

2. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 243.

3. DBNPS System Description for *Reactor Coolant System*, SD-039A.

4. *Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors*, 58 FR 39 132, dated July 22, 1993.

5. NUREG-1430, Revision 1, *Standard Technical Specifications – Babcock and Wilcox Plants*, dated April 1995.

6. 10 CFR 50.36, "Technical Specifications."

7. 10 CFR 50.59, "Changes, Tests, and Experiments."

8. 10 CFR 50.71, "Maintenance of Records, Making of Reports."

9. *American Society of Mechanical Engineers, Boiler and Pressure Vessel Code* Section III – 1968 including summer 1968 Addenda.

10. Topical Report BAW-1791, *BWOG Probabilistic Evaluation of Pressurized Thermal Shock*, June, 1983.

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INFORMATION ONLY

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with:

a. A steam bubble,

b. A water level between 45 and 305 inches.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the pressurizer inoperable, be in at least HOT STANDBY with the control rod drive trip breakers open within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4 The pressurizer shall be demonstrated OPERABLE by verifying pressurizer level to be within limits at least once per 12 hours.

~~REACTOR COOLANT SYSTEM~~

~~PRESSURIZER~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.4.9.2 The pressurizer temperature shall be limited to:~~

~~a. A maximum heatup and cooldown of 100°F in any one hour period,~~

~~b. A maximum spray water temperature differential of 410°F, and~~

~~c. A minimum temperature of 120°F when the pressurizer pressure is ≥ 625 psig.~~

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 500 psig, within the following 30 hours.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.4.9.2 The pressurizer temperature shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit once per 12 hours during spray operation with pressurizer temperature ≥ 440°F.~~

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INFORMATION ONLY

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation during one continuous time interval with specific activity levels exceeding 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 48 hours since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to < 530°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves.

The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The pressure-temperature limits of the reactor coolant pressure boundary are established in accordance with the requirements of Appendix G to 10 CFR 50 and with the thermal and loading cycles used for design purposes.

The limitations prevent non-ductile failure during normal operation, including anticipated operational occurrences and system hydrostatic tests. The limits also prevent exceeding stress limits during cyclic operation. The loading conditions of interest include:

1. Normal operations, including heatup and cooldown,
2. Inservice leak and hydrostatic tests, and
3. Reactor core operation.

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10 CFR 50. The closure head region, reactor vessel outlet nozzles and the beltline region have been identified to be the only regions of the reactor vessel, and consequently of the reactor coolant pressure boundary, that determine the pressure-temperature limitations concerning non-ductile failure.

Bases Figure 4-1

Fast Neutron Fluence (E>1 Mev) as a
Function of Full Power Service Life

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Bases Figure 4-2

Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessel Steels Exposed to 550°F Temperature

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Bases Table 4-1

Reactor Vessel Toughness

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The closure head region is significantly stressed at relatively low temperatures (due to mechanical loads resulting from bolt pre-load). This region largely controls the pressure-temperature limitations of the first several service periods. The outlet nozzles of the reactor vessel also affect the pressure-temperature limit curves of the first several service periods. This is due to the high local stresses at the inside corner of the nozzle which can be two to three times the membrane stresses of the shell. After the first several years of neutron radiation exposure, the RT_{NDT} temperature of the beltline region materials will be high enough so that the beltline region of the reactor vessel will start to control the pressure-temperature limitations of the reactor coolant pressure boundary. For the service period for which the limit curves, are established, the maximum allowable pressure as a function of fluid temperature is obtained through a point-by-point comparison of the limits imposed by the closure head region, outlet nozzles, and beltline region. The maximum allowable pressure is taken to be the lower pressure of the three calculated pressures. The pressure limit is adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all operating reactor coolant pump combinations. The limit curves were prepared based upon the most limiting adjusted reference temperature of all the beltline region materials at the end of twenty-one effective full power years.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside the radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The limit curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The unirradiated transverse impact properties of the beltline region materials, required by Appendices G and H to 10 CFR 50, were determined for those materials for which sufficient amounts of material were available. The adjusted reference temperatures are calculated by adding the predicted radiation-induced ΔRT_{NDT} and the unirradiated RT_{NDT}. The procedures described in Regulatory Guide 1.99, Rev. 2, were used for predicting the radiation induced ΔRT_{NDT} as a function of the material's copper and nickel content and neutron fluence.

Figure 3.4-2 presents the pressure-temperature limit curve for normal heatup. This figure also presents the core criticality limits as required by Appendix G to 10 CFR 50. Figure 3.4-3 presents the pressure-temperature limit curve for normal cooldown. Figure 3.4-4 presents the pressure-temperature limit curves for heatup and cooldown for inservice leak and hydrostatic testing.

All pressure-temperature limit curve are applicable up to twenty-one effective full power years. The protection against non-ductile failure is assured by maintaining the coolant pressure below the upper limits of Figures 3.4-2, 3.4-3 and 3.4-4.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in BAW 1543A. The withdrawal schedule is based on four considerations: (a) uncover possible technical anomalies as early in life as they can be detected (end of first fuel cycle), (b) define the material properties needed to perform the analysis required by Appendix G to 10 CFR 50, (c) reserve two capsules for evaluation of the effectiveness of thermal annealing in the event inplace annealing becomes necessary, (d) provide material property data corresponding to the reactor vessel beltline conditions at the end of service. This withdrawal schedule is specified to assure compliance with the requirements of Appendix H to 10 CFR 50. Appendix H references the requirements of ASTM E185 for surveillance program criteria.

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Attachment 2

TYPED TECHNICAL SPECIFICATION

REPLACEMENT PAGES

(2 Pages Follow)

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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Docket Number 50-346
License Number NPF-3
Serial Number 2675
Enclosure 2

COMMITMENT LIST

THE FOLLOWING LIST IDENTIFIES THOSE ACTIONS COMMITTED TO BY THE DAVIS-BESSE NUCLEAR POWER STATION (DBNPS) IN THIS DOCUMENT. ANY OTHER ACTIONS DISCUSSED IN THE SUBMITTAL REPRESENT INTENDED OR PLANNED ACTIONS BY THE DBNPS. THEY ARE DESCRIBED ONLY AS INFORMATION AND ARE NOT REGULATORY COMMITMENTS. PLEASE NOTIFY THE MANAGER – REGULATORY AFFAIRS (419-321-8450) AT THE DBNPS OF ANY QUESTIONS REGARDING THIS DOCUMENT OR ANY ASSOCIATED REGULATORY COMMITMENTS.

COMMITMENTS	DUE DATE
Relocate the requirements of TS 3/4.4.9.2 to the TRM.	No later than the implementation of the NRC-approved License Amendment that allows for their removal from the TS.