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December 18, 2009
L-09-225

10 CFR 50.12
10 CFR 50.90

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

SUBJECT:
Davis-Besse Nuclear Power Station, Unit No. 1
Docket No. 50-346, License No. NPF-3
Supplemental Information Related to a License Amendment Request to Incorporate the Use of Alternate Methodologies for the Development of Reactor Pressure Vessel Pressure-Temperature Limit Curves, and Request for Exemption From Certain Requirements Contained in 10 CFR 50.61 and 10 CFR 50, Appendix G (TAC No. ME1127 – License Amendment Request, ME1128 – Exemption Request)

On April 15, 2009, the FirstEnergy Nuclear Operating Company (FENOC) submitted a license amendment request and an exemption request on behalf of the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS) to the Nuclear Regulatory Commission (NRC) (Agencywide Documents Access and Management System [ADAMS] Accession No. ML091130228) for review and approval. The proposed amendment request and proposed exemption request incorporates alternate methodologies for the development of reactor coolant system (RCS) pressure-temperature limits into Technical Specification 5.6.4, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)." These alternate methodologies would also be used in the low temperature RCS overpressure event analysis. The proposed amendment also requested a revision to the periodicity of reanalysis requirement for the low temperature RCS overpressure events contained in Operating License (OL) Condition 2.C(3)(d).

On July 1, 2009, a teleconference between the NRC and FENOC staffs was held to discuss the alternate methodologies presented in the license amendment request. As a result of that teleconference, FENOC elected to eliminate one of the alternate

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methodologies for the calculation of the adjusted reference temperatures (ARTs), replacing it with the existing NRC-approved methodology, which is described in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

On September 29, 2009, during the development of the submittal which replaced the ART methodologies, FENOC was notified by AREVA NP of an error in the projected fluence of the reactor vessel nozzle belt forging. The updated calculations show that the fluence at 52 Effective Full Power Years (EFPY) is higher than previously calculated. This issue was entered into both the AREVA NP and DBNPS corrective action programs. As a result, the low temperature overpressure protection analysis described in the original amendment request and the values contained in the data table in the original exemption request need to be revised.

As a result, FENOC is providing this supplement to the license amendment request and exemption request to reflect both the methodology change resulting from the July teleconference and modifications due to the resolution of the fluence issue. Enclosure A contains a description and evaluation of the proposed amendment, which incorporates the use of the Regulatory Guide 1.99, Revision 2 methodology for the calculation of the ARTs. Attachments to Enclosure A provide a copy of the existing OL/TS pages marked up to reflect the proposed amendment, re-typed OL/TS pages with the proposed amendment incorporated, and the TS Bases pages marked up to reflect the proposed amendment (for information only). Enclosure B contains the revised request for exemption from certain requirements of 10 CFR 50, Appendix G, "Fracture Toughness Requirements," and 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events." Additionally, other minor changes to both requests were incorporated to improve the clarity of the documents. Therefore, for ease of review, this supplement provides complete replacements for the initial license amendment and exemption requests that were submitted on April 15, 2009.

FENOC is requesting approval of the proposed license amendment and request for exemption by June 30, 2010, to be implemented within 90 days of approval.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager - Fleet Licensing, at (330) 761-6071.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 18, 2009.

Sincerely,

Barry S. Allen

Barry S. Allen

Enclosures:
A. Evaluation of Proposed License Amendment
B. Request for Exemption

cc: NRC Region III Administrator
NRC Project Manager
NRC Resident Inspector
Executive Director, Ohio Emergency Management Agency, State of Ohio (NRC Liaison)
Utility Radiological Safety Board

Subject: License Amendment Request to incorporate into Technical Specification (TS) 5.6.4 the use of alternate methodologies for the determination of the initial reference temperature (RT_{NDT}) and in the development of the reactor pressure vessel (RPV) pressure-temperature (P-T) limit curves and the low temperature reactor coolant system (RCS) overpressure analysis. The license amendment request also revises the reanalysis requirement for the low temperature RCS overpressure events from 21 to 32 Effective Full Power Years (EFPY) contained in Operating License (OL) Condition 2.C(3)(d).

Attachments:

1. Proposed Operating License and Technical Specification Changes (Mark Up)

2. Proposed Operating License and Technical Specification Changes (Re-typed – For Information Only)

3. Proposed Technical Specification Bases Changes (For Information Only)

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating License NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS). The proposed amendment will incorporate into Technical Specification (TS) 5.6.4, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)" the use of alternate methodologies for the determination of the initial reference temperatures (RT_{NDT}) for welds in the reactor pressure vessel (RPV) beltline region and in the development of the RPV pressure-temperature (P-T) limit curves and the low temperature reactor coolant system (RCS) overpressure analysis. The proposed amendment also revises the requirement to provide a reanalysis of the low temperature RCS overpressure events from 21 to 32 Effective Full Power Years (EFPY) contained in Operating License (OL) Condition 2.C(3)(d).

The FirstEnergy Nuclear Operating Company (FENOC) is submitting this amendment request in order to use more current methods to develop the DBNPS P-T limit curves than are presently described within the DBNPS TS.

2.0 DETAILED DESCRIPTION

2.1 <u>Proposed Changes</u>

TS 5.6.4 identifies the approved methodology for performing RCS P-T limits analyses. The currently approved method is the Nuclear Regulatory Commission (NRC)-approved Babcock and Wilcox (B&W) Topical Report BAW-10046A, Revision 2, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G."

The proposed amendment requests, to include into TS 5.6.4, the use of three alternate methodologies in the development of RCS P-T limit curves and low temperature RCS overpressure protection analyses in addition to BAW-10046A, Revision 2. One alternate methodology will be used to determine a material specific initial RT_{NDT} value for the welds in the RPV beltline region. Another alternate methodology will be used to specify the flaw orientation in welds and base metal. Finally, an alternate methodology will be used to select the acceptance criteria for the P-T curve analysis.

The methodology used to determine the material specific initial RT_{NDT} is described in B&W Topical Report BAW-2308, Revisions 1A and 2A, "Initial RT_{NDT} of Linde 80 Weld Materials." This topical report has been reviewed and approved for use by the NRC.

The methodology of the American Society of Mechanical Engineers (ASME) Code Case N-588 provides an alternate procedure for assuming axially oriented reference defects in all axial welds and base metal and circumferentially oriented reference defects in all circumferential welds. This Code Case has been accepted for use by the NRC.

The methodology of the ASME Code Case N-640 allows the use of K_{IC} (the material toughness property measured in terms of stress intensity factor, K_I, which will lead to non-ductile crack propagation) instead of K_{IA} (the critical value of the stress intensity factor, K_I, for crack arrest as a function of temperature) in the development of P-T limit curves. This Code Case has been accepted for use by the NRC.

OL Condition 2.C(3)(d) states that prior to operation beyond 21 EFPY FENOC shall provide to the NRC a reanalysis and proposed modifications, as necessary, to ensure continued means of protection against low temperature RCS overpressure events. The proposed amendment will revise the submittal requirement from 21 to 32 EFPY. The reanalysis required prior to exceeding 21 EFPY is included in this document as Section 3.2.

Attachment 1 contains a copy of TS 5.6.4 and OL Condition 2.C(3)(d) marked up with the aforementioned changes. Attachment 2 contains a copy of TS 5.6.4 and OL Condition 2.C(3)(d) re-typed with the proposed changes incorporated.

Changes to the Bases of TS 3.4.12, "Low Temperature Overpressure Protection (LTOP)," are required to reflect the change in OL Condition 2.C(3)(d) from 21 to 32 EFPY. Attachment 3 contains a copy of the TS Bases pages marked up with changes that support the proposed TS/OL changes. The Bases pages are provided for information only. These changes will be implemented using the DBNPS Technical Specifications Bases Control Program (TS 5.5.13) and evaluated under the DBNPS 10 CFR 50.59 program.

The alternate methodology used to calculate the initial RT_{NDT} values for the welds in the RPV beltline region described in the NRC-approved Topical Report BAW-2308, Revision 1-A and Revision 2-A, requires an exemption request to be submitted to the NRC to use the alternate methodology. The exemption request is submitted concurrently with this request.

2.2 Background

10 CFR 50, Appendix G, "Fracture Toughness Requirements," specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, which the pressure boundary may be subjected to over its service lifetime. Appendix G also states that the ASME Boiler and Pressure Vessel Code (Code) forms the basis for the requirements of this regulation.

10 CFR 50, Appendix G requires P-T limits and minimum temperature requirements be established for the RPV. These limits are defined by the various operating conditions that the plant can be in, whether or not fuel is in the vessel, whether or not the core is critical, and the reactor vessel pressure. The operating conditions include, but are not limited to, hydrostatic pressure and leak testing, and normal operation including anticipated operational occurrences. The regulation also requires the P-T limits be at least as conservative as the limits obtained by following the methods of analysis and margins contained in the ASME Code, Section XI, Appendix G.

The P-T limits are used to prevent non-ductile failure of the reactor coolant pressure boundary during the aforementioned plant conditions. The P-T limits also prevent exceeding stress limits during cyclic operations. The three RPV regions that have been analyzed are the RPV head, the RPV nozzles, and the RPV beltline. The maximum allowable pressure for each of the three vessel regions is computed across the allowed temperature range using NRC-approved methods. A composite P-T limit curve is then constructed using the most restrictive pressure of the three vessel regions across the entire range of expected plant temperatures.

The low temperature overpressure protection (LTOP) system limits RCS pressure at low temperatures so that the integrity of the RCS pressure boundary is not challenged by violating the P-T limits. Each time the P-T limit curves are revised, the LTOP system must be re-evaluated to ensure that its functional requirements can still be met.

The current DBNPS P-T limit curves and LTOP limits were developed using the NRC-approved topical report BAW-10046A, Revision 2 and Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The curves are valid until the DBNPS reactor vessel reaches 21 EFPY of neutron irradiation.

Since the DBNPS vessel is approaching the 21 EFPY point, revised P-T limit curves and LTOP limits must be developed in order to support operation beyond 21 EFPY. The revised P-T limit curves and LTOP limits will be based, in part, upon the alternate methodologies that have been developed since the 21 EFPY P-T limit curves and LTOP limits were implemented. Since these methodologies are not included in TS 5.6.4, the three methodologies will not be implemented until NRC approval of this license amendment and an exemption request is received. Further, the DBNPS will not be operated beyond the point where the reactor vessel has accumulated 21 EFPY of neutron irradiation until NRC approval of this license amendment and the submitted exemption request is received and is implemented at the DBNPS.

The alternate methodologies to be used in developing the new DBNPS P-T limit curves and LTOP limits are NRC-approved methodologies. The additional methodologies include a different method of determining the material specific initial RT_{NDT} value for the welds in the RPV beltline region, a different method of analyzing stresses based on the flaw orientation assumed in the analysis, and the use of different acceptance criteria, K_{IC}. The methods are described in the NRC-approved Topical Report BAW-2308, Revision 1-A and 2-A; the NRC-accepted ASME Code Case N-588; and the NRC-accepted Code Case N-640, respectively.

The alternate methodology for the calculation of the initial RT_{NDT} values for the welds in the reactor vessel beltline region are described in the NRC-approved BAW-2308, Revisions 1-A and 2-A. As stated in the NRC Safety Evaluations for the topical report revisions, one of the conditions for licensees to reference the topical report in licensing activities is to submit an exemption request to the NRC for review. The exemption request is being submitted concurrently with this request.

3.0 TECHNICAL EVALUATION

3.1 Evaluation of the Analytical Methods Used to Develop P-T Limit Curves

The analytical methods to be used to determine the revised P-T limit curves and LTOP limits for the DBNPS are contained in the NRC-approved BAW-10046A, Revision 2; in the NRC-approved BAW-2308, Revisions 1-A and 2-A; in the NRC-accepted ASME Code Cases N-588 and N-640; and in RG 1.99, Revision 2.

BAW-10046A

This topical report describes the Babcock and Wilcox practices, methods, and criteria for compliance with the requirements of 10 CFR 50, Appendix G. The topical report includes, but is not limited to, the description of the methodology for the development of the initial RT_{NDT} values for ferritic materials of the beltline region and the procedure used to determine the P-T limits of the RCS. The topical report was approved by the NRC as documented in a Safety Evaluation dated April 30, 1986.

Since the closure head and outlet nozzle regions of the reactor vessel do not receive significant irradiation during reactor operation, the material properties do not experience a significant change over plant life. Consequently, there are no requirements to monitor their reference transition temperatures. The methods which determine the maximum acceptable pressure for these regions are described within the topical report, as modified by the use of ASME Code Cases N-588 and N-640.

Since the reactor vessel beltline region does receive significant irradiation during reactor operation, the region experiences a significant change in RT_{NDT} over the life of the plant. The methodologies described in RG 1.99, Revision 2, are used in the analyses to account for the effects of neutron embrittlement on the materials used in the reactor vessel beltline region. The RG describes the methodologies used to calculate a material's adjusted reference temperature (ART). When credible surveillance data from the reactor is not available, then the following equation is used.

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

The ΔRT_{NDT} term is an adjustment in the reference temperature caused by irradiation, and the Margin term is added to provide conservatism in the values of the ART. RG 1.99, Revision 2 contains the methodologies to calculate these two terms.

The proposed amendment uses the BAW-2308, Revisions 1-A and 2-A method to determine the initial RT_{NDT} values for the RPV beltline welds rather than the methodology described within Topical Report BAW-10046A, Revision 2, which is used to evaluate the other beltline components.

The P-T limit curves and LTOP limits will use, in part, the methodologies described in BAW-10046A, Revision 2, except where noted within this License Amendment. Since this topical report is currently approved for use in DBNPS P-T limit curves, no further review of the topical report is required.

BAW-2308

This topical report was developed to provide an alternate method for determining the initial RT_{NDT} for Linde 80 beltline welds in Babcock and Wilcox fabricated reactor vessels. The alternative methodology was based on brittle-to-ductile transition range fracture toughness test data of the weld materials in accordance with ASTM Standard E 1921 and using ASME Code Case N-629. This methodology would be used in lieu of the nil-ductility reference temperature parameter specified in ASME Section III Paragraph NB-2331. The use of ASME Section III Paragraph NB-2331 is specified in 10 CFR 50, Appendix G(II)(D)(i), and its use is also described in RG 1.99. Revision 1-A of the topical report was approved by the NRC as documented in a Safety Evaluation dated August 4, 2005. Revision 2-A, which supplements Revision 1-A of the topical report, was approved by the NRC as documented in a Safety Evaluation dated March 24, 2008.

The Safety Evaluation for Revision 1-A of the topical report contained six conditions. Licensees had to satisfy four of these conditions in order to use the alternate methodology. The first three conditions describe performance requirements for use of the methodology. These conditions have been satisfied.

The fourth condition requires the submittal of an exemption request from the requirements of 10 CFR 50, Appendix G and 10 CFR 50.61 in order to use the methodology. The exemption request is submitted concurrently with this request.

The topical report was revised to incorporate additional information from the pressurized water reactor owner's group to satisfy the remaining two conditions contained in the NRC Safety Evaluation for Revision 1-A. The Safety Evaluation for Revision 2-A required licensees to follow the four conditions contained in the Safety Evaluation for Revision 1-A. The Safety Evaluation for BAW-2308, Revision 2-A also stated that the initial RT_{NDT} values for certain materials contained in the Revision 1-A version are more conservative. These values apply to the DBNPS materials, hence, the revised DBNPS P-T limit curves and LTOP limits are based upon the values contained in Revision 1-A.

Regulatory Guide 1.99, Revision 2

Regulatory Guide (RG) 1.99, Revision 2 contains a methodology for determining a material's ART to account for neutron embrittlement. The use of the RG provides an acceptable means of ensuring the requirements of 10 CFR 50, Appendix G are satisfied.

Since credible surveillance data for the DBNPS is not available, the ART was calculated by use of the following RG equation.

$$\text{ART} = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

Since the RG provides the current methodology for the determination of a material's ART, no further evaluation of the RG is required.

ASME Code Cases

ASME Code Case N-588 provides an alternative procedure for assuming axially oriented reference defects in all axial welds and base metal and circumferentially oriented reference defects in all circumferential welds.. ASME Code Case N-640 allows the use of K_{IC} (the material toughness property measured in terms of stress intensity factor, K_I, which will lead to non-ductile crack propagation) instead of K_{IA} (the critical value of the stress intensity factor, K_I, for crack arrest as a function of temperature) in the development of P-T limit curves.

NRC Regulatory Issue Summary 2004-04, "Use of Code Cases N-588, N-640, and N-641 in Developing Pressure-Temperature Operating Limits," states that these ASME Code Cases have been accepted as appropriate alternatives to the ASME Code, Section XI, Appendix G methodology. This acceptance has been documented in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1."

Though Regulatory Issue Summary 2004-04 accepts the use of ASME Code Cases N-588, N-640, and N-641; the use of Code Case N-641 is not part of this proposed amendment.

3.2 Evaluation of Low Temperature Overpressure Protection (LTOP)

As part of developing a revision to the DBNPS P-T limit curves, the LTOP limits are also reanalyzed. The LTOP analysis uses the same methodologies that will be used to develop the P-T limit curves, the methodologies described in BAW-10046A, Revision 2; RG 1.99, Revision 2; BAW-2308 Revisions 1-A and 2-A; ASME Code Case N-588; and ASME Code Case N-640. Hence, this License Amendment and the exemption request is also applicable to the LTOP reanalysis. Since not all of the methodologies are currently authorized for use at the DBNPS, the results of the revised LTOP analysis will not be implemented until this License Amendment and exemption request are reviewed and approved by the NRC. Further, the DBNPS will not be operated beyond the point where the reactor vessel has accumulated 21 EFPY of neutron irradiation until NRC approval of this license amendment and exemption request is received and is implemented at the DBNPS.

OL Condition 2.C(3)(d) requires FENOC to provide the NRC a reanalysis and proposed modifications, as necessary, to ensure continued means of protection against low temperature RCS overpressure events. The following information satisfies this requirement. No plant modifications are necessary to extend the protection against LTOP events to 32 EFPY.

The P-T limits for LTOP are obtained by taking 100 percent of the controlling normal operation heatup/cooldown limit. To support the development of an LTOP system effective temperature, the temperature difference is calculated between the 1/4t location of the reactor vessel wall and the RCS fluid for the condition when the metal temperature is at 156.23 °F + 50 °F, or 206.23 °F, as required by Article G-2215 of Section XI of the ASME Code, where 156.23 °F is the highest adjusted reference temperature (ART) and the 50 °F is a margin term. Temperature differences between the reactor coolant and the reactor vessel wall are evaluated for both the ramp and step heatup conditions when the metal temperature is lower and significantly lags the coolant temperature. During cooldown, the coolant temperature is always lower than the metal temperature, and therefore, it is not limiting in the development of the LTOP system enable temperature. The temperature at which LTOP must be provided is then calculated by summing the ART, the margin term, and the reactor coolant to vessel temperature difference (156.23 °F + 50 °F+ 28.77 °F = 235 °F). Therefore, LTOP must be enabled prior to the average RCS temperature going below 235 °F. Since TS 3.4.12 requires the LTOP to be enabled any time the plant is operating below 280 °F, the need to have the LTOP system active before going below 235 °F is met.

ASME Code Case N-640 allows the use of the K_{IC} fracture toughness curve rather than the K_{IA} curve. However, if this alternative is used, Case N-640 requires that LTOP systems limit the maximum pressure to 100 percent of the pressure allowed by the P-T limit curves. The minimum allowable pressure below 235 °F is controlled by the reactor pressure vessel head nil-ductility temperature limit. Therefore, 625 psig is selected to be the uncorrected LTOP limit. This pressure value is corrected for sensor location for both the A Hot Leg and B Hot Leg pressure taps.

The LTOP pressure limit as measured from the A Hot Leg pressure tap is 625 – 85 = 540 psig. The LTOP pressure limit as measured from the B Hot Leg pressure tap is 625 – 60 = 565 psig. These LTOP limiting pressures are applicable from 70 °F up to 235 °F. Above 235 °F, the standard NDT limits for the limiting component (for example, the reactor pressure vessel head, beltline region, and outlet nozzles, for a given temperature) are applicable.

The requirements of TS 3.4.12, "Low Temperature Overpressure Protection (LTOP)," are more conservative than the values calculated using BAW-2308, Revisions 1-A and 2-A, and Code Cases N-588 and N-640; therefore, no changes are proposed to TS 3.4.12.

OL Condition 2.C(3)(d) requires the submittal of a LTOP reanalysis prior to exceeding 21 EFPY. Since the LTOP reanalysis was performed, in part, using the alternate methodologies contained in the NRC-approved BAW-2308, Revisions 1-A and 2-A and the NRC-accepted ASME Code Cases N-588 and N-640, which are not yet approved by the NRC for use at the DBNPS, the reanalysis will not be made effective until the amendment request and the exemption request are approved and implemented at the DBNPS. The DBNPS will not be operated beyond the point where the reactor vessel has accumulated 21 EFPY of neutron irradiation until NRC approval of this license amendment and the exemption request is received and is implemented at the DBNPS. Since this is expected to occur prior to the DBNPS achieving 21 EFPY, compliance with the OL Condition will be satisfied.

The LTOP reanalysis is valid through 52 EFPY. Since the end of service for the DBNPS is 32 EFPY, the OL Condition will be revised to that point.

3.3 <u>Conclusion</u>

The revised DBNPS P-T limit curves will be developed using the NRC-approved topical report, BAW-10046A, Revision 2, RG 1.99, Revision 2, and three alternate methodologies. The alternate methodologies are the NRC-approved topical report BAW-2308, Revisions 1-A and 2-A; and the NRC-accepted ASME Code Cases N-588 and N-640. The LTOP reanalysis was developed using the aforementioned documents. Since the alternate methodologies are either approved or accepted for use by the NRC, the alternate methodologies provide an acceptable means of satisfying 10 CFR 50, Appendix G, which governs the development of P-T limit curves and LTOP limits.

As a result, TS 5.6.4 needs to be revised to reflect the use of the aforementioned alternative methodologies. A Pressure-Temperature Limits Report will be submitted to the NRC in accordance with TS 5.6.4 after this proposed amendment request and the exemption request are approved by the NRC, and implemented by the DBNPS staff, but before the DBNPS operates beyond 21 EFPY.

The LTOP reanalysis has been performed as described in Section 3.2 of this submittal. Since the LTOP reanalysis was performed, in part, using the alternate methodologies contained in the NRC-approved BAW-2308, Revisions 1-A and 2-A and the NRC-accepted ASME Code Cases N-588 and N-640, which are not yet approved by the NRC for use at the DBNPS, the reanalysis will not be made effective until the amendment request and an exemption request are approved and implemented at the DBNPS. Since the reanalysis is required to be submitted to the NRC prior to 21 EFPY and given that the proposed amendment and exemption request would be approved and implemented prior to the DBNPS achieving 21 EFPY, compliance with the OL Condition would be satisfied.

The effective period for License Condition 2.C(3)(d) is being extended to 32 EFPY to ensure that the DBNPS re-examines the efficacy of its LTOP system prior to operating beyond that point in time.

4.0 REGULATORY EVALUATION

The FirstEnergy Nuclear Operating Company (FENOC) requests Nuclear Regulatory Commission (NRC) review and approval of an amendment for the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS) Technical Specifications (TS). The proposed amendment would incorporate into TS 5.6.4, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)" the use of alternate methodologies for the calculation of the reactor vessel beltline weld initial reference transition temperatures (RT_{NDT}), the use of axially oriented reference defects in axial welds and base metal, the use of circumferentially oriented reference defects in circumferential welds, and the use of K_{IC} in the development of the P-T limit curves and the low temperature RCS overpressure analysis, in addition to the use of the NRC-approved BAW-10046A, Revision 2 topical report. Additionally, the proposed amendment would revise the 21 Effective Full Power Years (EFPY) requirement in Operating License (OL) Condition 2.C(3)(d) to provide a reanalysis of the low temperature RCS overpressure events to 32 EFPY.

4.1 Significant Hazards Consideration

FENOC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The amendment request proposes two changes to the TS/OL. The first change incorporates the use of alternate methodologies to develop the DBNPS P-T limit curves and LTOP limits into TS 5.6.4 to augment the existing listed methodology of BAW-10046A, Revision 2. The second change revises OL Condition 2.C(3)(d) to reflect the revised LTOP analysis is valid to 32 EFPY.

The first change incorporates the use of Topical Report BAW-2308, Revisions 1-A and 2-A and ASME Code Cases N-588 and N-640. The topical report and ASME code cases have been approved or accepted for use by the NRC (provided that any conditions/limitations are satisfied). The proposed additions to the methodologies for the reactor vessel P-T curve and LTOP limit development provide an acceptable means of satisfying the requirements of 10 CFR 50, Appendix G. The proposed additions do not alter the design, function, or operation of any plant equipment. Therefore, the proposed additions do not affect the probability or consequences of any previously evaluated accidents, including reactor coolant pressure boundary failures.

The second change is considered administrative in nature and reflects the revised methodologies. It does not alter the design, function, or operation of any plant equipment. Therefore, the proposed change does not affect the probability or consequences of any previously evaluated accidents.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The amendment request proposes two changes to the TS/OL. The first change incorporates the use of alternate methodologies to develop the DBNPS P-T limit curves and LTOP limits into TS 5.6.4 to augment the existing listed methodology of BAW-10046A, Revision 2. The second change revises OL Condition 2.C(3)(d) to reflect that the revised analysis is valid to 32 EFPY.

The first change incorporates methodologies that either have been approved or accepted for use by the NRC (provided that any conditions/limitations are satisfied). The changes do not alter the design, function, or operation of any plant equipment. The P-T limit curves and LTOP limits will provide the same level of protection to the reactor coolant boundary as was previously evaluated. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The second change is considered administrative in nature and reflects the revised methodologies. It does not alter the design, function, or operation of any plant equipment. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The amendment request proposes two changes to the TS/OL. The first change incorporates the use of alternate methodologies to develop the DBNPS P-T limit curves and LTOP limits into TS 5.6.4 to augment the existing listed methodology of BAW-10046A, Revision 2. The second change revises OL Condition 2.C(3)(d) to reflect that the revised analysis is valid to 32 EFPY. The first change incorporates methodologies that either have been approved or accepted for use by the NRC (provided that any conditions/limitations are satisfied). The second change is considered administrative in nature and reflects the revised methodologies. The changes do not alter the design, function, or operation of any plant equipment. The P-T limit curves and LTOP limits will provide the same level of protection to the reactor coolant boundary as was previously evaluated. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, FENOC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.2 <u>Applicable Regulatory Requirements/Criteria</u>

The following paragraphs describe relevant regulatory requirements associated with the proposed TS/OL change.

10 CFR 50, Appendix G, "Fracture Toughness Requirements"

10 CFR 50, Appendix G requires P-T limits and minimum temperature requirements be established for the RPV. These limits are defined by the various operating conditions that the plant can be in, whether or not fuel is in the vessel, whether or not the core is critical, and the reactor vessel pressure. The operating conditions include, but are not limited to, hydrostatic pressure and leak testing, and normal operation including anticipated operational occurrences. The

regulation also requires the P-T limits be at least as conservative as the limits obtained by following the methods of analysis and margins contained in the ASME Code, Section XI, Appendix G.

The revised DBNPS P-T limit curves will be developed using the NRC-approved topical report, BAW-10046A, Revision 2; RG 1.99, Revision 2; and the alternate methods described in the NRC-approved topical report BAW-2308, Revisions 1-A and 2-A, and the NRC-accepted ASME Code Cases N-588 and N-640. The LTOP reanalysis was developed using the aforementioned documents. The analysis used to support the change to License Condition 2.C(3)(d) also uses the aforementioned documents. Since the alternative methodologies are either approved or accepted for use by the NRC, the alternate methodologies provide an acceptable means of satisfying 10 CFR 50, Appendix G, which governs the development of P-T limit curves and LTOP limits.

Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials"

Regulatory Guide (RG) 1.99, Revision 2 contains methodologies for determining the change in RT_{NDT} resulting from neutron radiation. The use of the RG provides an acceptable means of ensuring the requirements of 10 CFR 50, Appendix G are satisfied. The RG also indicates that other methodologies could be used for complying with the specified portions of the regulation.

When credible surveillance data from the reactor is not available, then the RG states that the ART is calculated using the following equation:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

The factors are defined within the RG. This equation is used to develop the ART for various materials. However, the use of Topical Report BAW-2308, Revisions 1-A and 2-A provided the initial RT_{NDT} values for the DBNPS reactor vessel beltline welds. The topical report has been approved by the NRC with a condition that an exemption request needs to be submitted and approved by the NRC in order to use the topical report. The exemption request has been submitted concurrent with this request.

Since the DBNPS P-T curves will be developed using RG 1.99, Revision 2, as modified by the use of NRC-approved topical report BAW-2308, Revisions 1-A and 2-A, the requirements of 10 CFR 50, Appendix G will be satisfied upon NRC approval of the use of BAW-2308 for the DBNPS.

American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure"

As indicated in the description for 10 CFR 50, Appendix G, above, the proposed changes to the TS and OL meet the requirements of the ASME Code, Section XI, Appendix G as modified by the NRC-accepted ASME Code Cases N-588 and N-640.

4.3 Precedent

An exemption request for use of BAW-2308, Revision 1-A at the Surry Power Station Unit Nos. 1 and 2 was approved by NRC Safety Evaluation dated June 27, 2007 (ADAMS Accession No. ML071160287).

The use of ASME Code Cases N-588, N-640, and N-641 were incorporated into WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," and approved by the NRC in a Safety Evaluation dated February 27, 2004. WCAP-14040, Revision 4 removed the requirement for exemption requests to use the code cases. The WCAP-14040, Revision 4 was approved for use at the Vogtle Electric Generating Plant, Units 1 and 2, and the Comanche Peak Steam Electric Station, Units 1 and 2 by NRC Safety Evaluations dated March 28, 2005 and February 22, 2007, respectively (ADAMS Accession No. ML050690216 and ML070320823, respectively).

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Nuclear Regulatory Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Babcock and Wilcox Owner's Group Topical Report BAW-2308, Revision 1-A, "Initial RT_{NDT} of Linde 80 Weld Materials," August 2005
2. Babcock and Wilcox Owner's Group Topical Report BAW-2308, Revision 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials," March 2008
3. NRC Regulatory Issue Summary 2004-04, "Use of Code Cases N-588, N-640, and N-641 in Developing Pressure-Temperature Operating Limits," April 5, 2004
4. Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 13
5. AREVA Document ANP-2718, "Appendix G Pressure-Temperature Limits For 52 EFPY, Using ASME Code Cases for Davis-Besse Nuclear Power Station," Revision 002, October 2008 as accepted for use by FENOC on November 2, 2009.
6. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988
7. 10 CFR 50, Appendix G, "Fracture Toughness Requirements," December 19, 1995 and January 31, 2008
8. Babcock and Wilcox Topical Report BAW-10046A, Revision 2, "Methods of Compliance With Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," June 1986

Attachment 1

Proposed Operating License and Technical Specification Changes (Mark Up)

(Eight Pages Enclosed)

NO CHANGE PROPOSED.
INCLUDED FOR CONTEXT.

2.C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

FENOC is authorized to operate the facility at steady state reactor core power levels not in excess of 2817 megawatts (thermal). Prior to attaining the power level, Toledo Edison Company shall comply with the conditions identified in Paragraph (3) (o) below and complete the preoperational tests, startup tests and other items identified in Attachment 2 to this license in the sequence specified. Attachment 2 is an integral part of this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 280 , are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission:

(a) FENOC shall not operate the reactor in operational Modes 1 and 2 with less than three reactor coolant pumps in operation.

(b) Deleted per Amendment 6

(c) Deleted per Amendment 5

2.C(3)(d) Prior to operation beyond ~~21~~ 32 Effective Full Power Years, FENOC shall provide to the NRC a reanalysis and proposed modifications, as necessary, to ensure continued means of protection against low temperature reactor coolant system overpressure events.

(e) Deleted per Amendment 33

(f) Deleted per Amendment 33

(g) Deleted per Amendment 33

(h) Deleted per Amendment 24

(i) Deleted per Amendment 11

(j) Revised per Amendment 3
Deleted per Amendment 28

(k) Within 60 days of startup following the first (1st) regularly scheduled refueling outage, Toledo Edison Company shall complete tests and obtain test results as required by the Commission to verify that faults on non-Class IE circuits would not propagate to the Class IE circuits in the Reactor Protection System and the Engineered Safety Features Actuation System.

(l) Revised per Amendment 7
Deleted per Amendment 15

(m) Deleted per Amendment 7

(n) Deleted per Amendment 10

(o) Deleted per Amendment 2

(p) Deleted per Amendment 29

(q) Deleted per Amendment 7

(r) Deleted per Amendment 30

(s) Toledo Edison Company shall be exempted from the requirements of Technical Specification 3/4.7.8.1 for the two (2) Americium-Beryllium-Copper startup sources to be installed or already installed for use during the first refueling cycle until such time as the sources are replaced.

(t) Added per Amendment 83
Deleted per Amendment 122

Amendment No. ~~2,3,7,10,11,15,17,24,28,29,30,33,83,122,199,~~228

NO CHANGES PROPOSED.
INCLUDED FOR CONTEXT.

2.C(4) Fire Protection

FENOC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Safety Analysis Report and as approved in the SERs dated July 26, 1979, and May 30, 1991, subject to the following provision:

FENOC may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(5) Deleted per Amendment No. 279.

(6) Antitrust Conditions

FENOC and FirstEnergy Nuclear Generation Corp. shall comply with the antitrust conditions delineated in Condition 2.E of this license as if named therein. FENOC shall not market or broker power or energy from the Davis-Besse Nuclear Power Station, Unit No. 1. FirstEnergy Nuclear Generation Corp. is responsible and accountable for the actions of FENOC to the extent that said actions affect the marketing or brokering of power or energy from the Davis-Besse Nuclear Power Station, Unit No. 1, and in any way, contravene the antitrust license conditions contained in the license.

NO CHANGES PROPOSED.
INCLUDED FOR CONTEXT.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.2 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.3 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

   1. SL 2.1.1.1, "Reactor Core Safety Limits";

   2. LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";

   3. LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";

   4. LCO 3.1.7, "Position Indicator Channels," (SR 3.1.7.1 limits);

NO CHANGE PROPOSED.
INCLUDED FOR CONTEXT.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

5. LCO 3.1.8, "PHYSICS TEST Exceptions - MODE 1";
6. LCO 3.1.9, "PHYSICS TEST Exceptions - MODE 2";
7. LCO 3.2.1, "Regulating Rod Insertion Limits";
8. LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits";
9. LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits";
10. LCO 3.2.4, "QUADRANT POWER TILT (QPT)";
11. LCO 3.2.5, "Power Peaking Factors";
12. LCO 3.3.1,"Reactor Protection System (RPS) Instrumentation," Function 8 (Flux - ΔFlux – Flow) Allowable Value; and
13. LCO 3.9.1, "Boron Concentration."

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, as described in BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," or any other new NRC approved analytical methods used to determine core operating limits that are not yet referenced in the applicable approved revision of BAW-10179P-A. The applicable approved revision number for BAW-10179P-A at the time of the reload analyses are performed shall be identified in the CORE OPERATING LIMITS REPORT (COLR). The COLR shall also list any new NRC approved analytical methods used to determine core operating limits that are not yet referenced in the applicable approved revision of BAW-10179P-A.

c. As described in reference documents listed in accordance with the instructions given above, when an initial assumed power level of 102% of RTP is specified in a previously approved method, an actual value of 100.37% of RTP may be used when the input for reactor thermal power measurement of feedwater mass flow and temperature is from the Ultrasonic Flow Meter. The following NRC approved documents are applicable to the use of the Ultrasonic Flow Meter with a 0.37% measurement uncertainty:

1. Caldon Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, dated March, 1997.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

2. Caldon Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM√™ or LEFM CheckPlus™ System," Revision 5, dated October, 2001.

d. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

e. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. BAW-10046A, Rev. 2, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50 Appendix G," June 1986.

INSERT →

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.5 Post Accident Monitoring Report

When a report is required by Condition B or F of LCO 3.3.17, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

INSERT

2. ASME Code Section XI, Appendix G, 1995 Edition with Addenda through 1996, as modified by the alternative procedures provided in ASME Code Case N-640 and ASME Code Case N-588; and

3. BAW-2308, Revision 1-A and Revision 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials," August 2005 and March 2008, respectively.

NO CHANGE PROPOSED.
INCLUDED FOR CONTEXT.

5.6 Reporting Requirements

5.6.6 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG;

b. Active degradation mechanisms found;

c. Nondestructive examination techniques utilized for each degradation mechanism;

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications;

e. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism;

f. Total number and percentage of tubes plugged or repaired to date;

g. The results of condition monitoring, including the results of tube pulls and in-situ testing;

h. The effective plugging percentage for all plugging and tube repairs in each SG; and

i. Repair method utilized and the number of tubes repaired by each repair method.

5.6.7 Remote Shutdown System Report

When a report is required by Condition C of LCO 3.3.18, "Remote Shutdown System," a report shall be submitted within the following 30 days. The report shall outline the action taken, the cause of the inoperability, and the plans and schedule for restoring the control circuit or transfer switch of the Function to OPERABLE status.

Attachment 2

Proposed Operating License and Technical Specification Changes
(Re-typed – For Information Only)

(Three Pages Enclosed)

2.C(3)(d) Prior to operation beyond 32 Effective Full Power Years, FENOC shall provide to the NRC a reanalysis and proposed modifications, as necessary, to ensure continued means of protection against low temperature reactor coolant system overpressure events.

(e) Deleted per Amendment 33

(f) Deleted per Amendment 33

(g) Deleted per Amendment 33

(h) Deleted per Amendment 24

(i) Deleted per Amendment 11

(j) Revised per Amendment 3
Deleted per Amendment 28

(k) Within 60 days of startup following the first (1st) regularly scheduled refueling outage, Toledo Edison Company shall complete tests and obtain test results as required by the Commission to verify that faults on non-Class IE circuits would not propagate to the Class IE circuits in the Reactor Protection System and the Engineered Safety Features Actuation System.

(l) Revised per Amendment 7
Deleted per Amendment 15

(m) Deleted per Amendment 7

(n) Deleted per Amendment 10

(o) Deleted per Amendment 2

(p) Deleted per Amendment 29

(q) Deieted per Amendment 7

(r) Deleted per Amendment 30

(s) Toledo Edison Company shall be exempted from the requirements of Technical Specification 3/4.7.8.1 for the two (2) Americium-Beryllium-Copper startup sources to be installed or already installed for use during the first refueling cycle until such time as the sources are replaced.

(t) Added per Amendment 83
Deleted per Amendment 122

Amendment No. ~~2,3,7,10,11,15,17,24,28,29,30,33,83,122,199,228~~, 5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

2. Caldon Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM√™ or LEFM CheckPlus™ System," Revision 5, dated October, 2001.

d. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

e. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. BAW-10046A, Rev. 2, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50 Appendix G," June 1986;

2. ASME Code Section XI, Appendix G, 1995 Edition with Addenda through 1996, as modified by the alternative procedures provided in ASME Code Case N-640 and ASME Code Case N-588; and

3. BAW-2308, Revision 1-A and Revision 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials," August 2005 and March 2008, respectively.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6 Reporting Requirements

5.6.5 Post Accident Monitoring Report

When a report is required by Condition B or F of LCO 3.3.17, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

Attachment 3

Proposed Technical Specification Bases Changes (For Information Only)

(Five Pages Enclosed)

NO CHANGES THIS PAGE

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP)

BASES

BACKGROUND

LTOP controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) requirements of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for providing such protection. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," provides the allowable combinations for operational pressure and temperature during cooldown, shutdown, and heatup to keep from violating the Reference 1 limits.

The reactor vessel material is less tough at reduced temperatures than at normal operating temperature. Also, as vessel neutron irradiation accumulates, the material becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure must be maintained low when temperature is low and must be increased only as temperature is increased.

Operational maneuvering during cooldown, heatup, or any anticipated operational occurrence must be controlled to not violate LCO 3.4.3. Exceeding these limits could lead to brittle fracture of the reactor vessel. LCO 3.4.3 presents requirements for administrative control of RCS pressure and temperature to prevent exceeding the P/T limits.

This LCO provides RCS overpressure protection in the applicable MODES by ensuring an adequate pressure relief capacity through the Decay Heat Removal (DHR) System relief valve.

The DHR System relief valve provides overpressure protection for the RCS during low temperature operations. RCS and DHR Systems are monitored for temperature and pressure. Maintaining the relief setpoint within the limits of the LCO ensures the Reference 1 limits will be met in any event in the LTOP analysis.

If system pressure exceeds the lift setpoint of the DHR System relief valve, it will open. As the relief valve opens, coolant is released and pressure decreases. When the relief valve reset is reached, below the LTOP pressure limit, the relief valve closes.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 3) demonstrate that the reactor vessel can be adequately protected against overpressurization transients during shutdown. In MODES 1 and 2, and portions of MODE 3, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. For the remaining portions of MODE 3, overpressure protection is

NO CHANGES THIS PAGE

APPLICABLE SAFETY ANALYSES (continued)

provided by operating procedures. At nominally 280°F to 140°F, overpressure prevention falls to the OPERABLE DHR System relief valve. Below 140°F, credible overpressurization sources are secured.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as vessel material toughness decreases due to neutron embrittlement. Each time the P/T limit curves are revised, the LTOP will be re-evaluated to ensure that its functional requirements can still be met with the DHR System relief valve and operating procedures.

Transients that are capable of overpressurizing the RCS have been identified and evaluated. These transients relate to either mass input or heat input: actuating the High Pressure Injection (HPI) System, discharging the Core Flooding Tanks (CFTs), energizing the pressurizer heaters, failing the makeup control valve open, losing decay heat removal, and starting a reactor coolant pump (RCP).

The DHR System relief valve (DH-4849), which is in the suction line to the decay heat pumps, has been sized to pass 1800 gpm at the nominal set pressure of 320 psig. The flow rate is based on the maximum developed runout flow (900 gpm per pump) with both HPI pumps running simultaneously. This flow rate is considered to cause the worst credible pressure transient. The opening of a CFT isolation valve was not considered because power is removed from the valve once it is closed upon plant cooldown and depressurization. Other postulated occurrences, makeup control valve failing to open, loss of DHR System cooling, all pressurizer heaters energizing, do not produce a pressure excursion as severe as that produced by the two HPI pumps. Although the pressurizer, by procedure, cannot be solid, for the purpose of analysis it was considered to go solid during the transient. The DHR System relief valve is a Seismic Class I Nuclear Class 2 bellows type of safety-relief valve. It should be noted that the postulation of both HPI pumps starting during DHR System operation is made only for the purpose of sizing the DHR System relief valve. The possibility of this event occurring due to either a single operator error or a single spurious signal is precluded by the design of the Safety Features Actuation System.

The Reference 3 analyses demonstrate the DHR System relief valve can maintain RCS pressure below limits.

The DHR System relief valve is placed in service before RCS temperature is reduced below 280°F. Above this temperature, the pressurizer safety valves and operating procedures provide the reactor

BASES

APPLICABLE SAFETY ANALYSES (continued)

vessel pressure protection. The vessel materials were ~~assumed to~~ have a neutron irradiation accumulation ~~equal to 21~~ GREATER THAN 32 effective full power years (EFPYs) of operation.

Reference 3 contains the acceptance limits that satisfy the LTOP requirements. Any change to the RCS must be evaluated against these analyses to determine the impact of the change on the LTOP acceptance limits.

As required by License Condition 2.C(3)(d), prior to operation beyond ~~21~~ 32 Effective Full Power Years, a reanalysis and proposed modifications, as necessary, to ensure continued means of protection for LTOP events will be provided to the NRC. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to embrittlement induced by neutron irradiation. Revised P/T limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens.

LCO

For low temperature overpressure protection, Davis-Besse relies on the four-inch DHR System relief valve (DH-4849) with a lift setpoint ≤ 330 psig.. This relief valve is located on the DHR System suction line from the RCS. The RCS to DHR System isolation valves (DH-11 and DH-12) must be open and control power removed from the valve operators for the DHR System relief valve to be OPERABLE. Control power can be removed either in the control room or at the motor control center (by removing fuses, opening breakers, or racking breakers out).

APPLICABILITY

This LCO is applicable in MODES 4 and 5, and in MODE 6 when the reactor vessel head is on. The Applicability is established by fracture mechanics analyses. The pressurizer safety valves provide overpressure protection to meet LCO 3.4.3 P/T limits in MODES 1, 2, and 3. With the vessel head off, overpressurization is not possible.

LCO 3.4.3 provides the operational P/T limits for all MODES.
LCO 3.4.10, "Pressurizer Safety Valves," requires the pressurizer safety valves OPERABLE to provide overpressure protection during MODES 1, 2, and 3.

ACTIONS

A.1 and A.2

With the DHR System relief valve inoperable due to one or both RCS to DHR System isolation valves closed, the overpressure protection flow path is isolated. The flow path must be restored by opening the RCS to DHR System isolation bypass valves (DH-21 and DH-23), within 1 hour.

NO CHANGES THIS PAGE

ACTIONS

A.1 and A.2 (continued)

After opening, the RCS to DHR System isolation bypass valves must be verified open every 24 hours.

The 1 hour Completion Time reflects the importance of the action and provides time for a timely opening of the RCS to DHR System isolation bypass valves. To ensure they remain in the open position, the positions of the RCS to DHR System isolation bypass valves are required to be verified every 24 hours. RCS to DHR System isolation bypass valves are manual valves and do not have remote position indication.

B.1

With control power available to one or both of the RCS to DHR System isolation valves, the overpressure protection flow path could be inadvertently isolated. The control power must be removed from the valves within 1 hour to ensure the valves will remain open during system operation.

The 1 hour Completion Time reflects the importance of the action and provides time for a timely removal of control power.

C.1

If the DHR System relief valve is inoperable for reasons other than the relief flow path (Condition A or B), the DHR System relief valve must be restored to OPERABLE status within 8 hours. The 8 hour Completion Time is acceptable due to the low probability of an overpressure event.

D.1, D.2, D.3, and D.4

If any Required Action and Completion Time of Condition A, B, or C is not met, other compensatory actions must be taken to minimize the probability and consequences of an LTOP event. Without an OPERABLE relief path for overpressure protection, the RCS water addition capabilities must be limited. Within 1 hour both HPI pumps must be disabled (e.g., by opening motor supply breakers), and within 8 hours the makeup pump suction automatic transfer to the borated water storage tank on low makeup tank level must be disabled. Makeup tank level must be verified to be ≤ 73 inches within 8 hours to minimize volume. Furthermore, without an overpressure relief path, RCS pressure and pressurizer level

NO CHANGES THIS PAGE

BASES

ACTIONS

D.1, D.2, D.3, and D.4 (continued)

must be verified to be in the Acceptable Region of Figure 3.4.12-1 or 3.4.12-2 (depending on the MODE) within 8 hours to ensure an overpressure condition cannot occur. These Figures do not include instrument error uncertainties.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1

Verification of the flow path from the RCS to the DHR System relief valve is required every 24 hours. This verification is performed by checking RCS to DHR System isolation valves in the open position with control power removed from the valve operator. This Surveillance ensures the overpressure relief flow path is aligned and remains aligned. Removal of control power ensures the flow path is not inadvertently closed.

The Frequency is adequate based on operating experience. Manual operation is required to close the isolation valves or energize control power. Valve operations are administratively controlled by procedure. In this configuration the isolation valves will not inadvertently close.

SR 3.4.12.2

Verification of the DHR System relief valve lift setpoint must be performed to ensure LTOP requirements can be met. Overpressure protection of the RCS is ensured by the DHR System relief valve, which relieves pressure and prevents the RCS from exceeding the Pressure/Temperature Limits.

The DHR System relief valve setpoint is verified in accordance with the Inservice Testing (IST) Program for proper operation and correct lift setting of ≤ 330 psig. This lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. The IST Program specifies the testing and frequency, as directed by ASME Code.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. UFSAR, Section 9.3.5.

Enclosure B

REQUEST FOR EXEMPTION

(Six Pages Follow)

Subject: Request for exemption from certain requirements of 10 CFR 50, Appendix G, "Fracture Toughness Requirements," and 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events." The requested exemption would allow the use of an alternate method, described in Topical Report BAW-2308, Revisions 1-A and 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials," for determining the initial, unirradiated material reference temperatures of Linde 80 weld materials present in the beltline region of the Davis-Besse Nuclear Power Station, Unit No. 1 reactor pressure vessel.

1.0 INTRODUCTION

In accordance with the provisions of 10 CFR 50.60(b) and 10 CFR 50.12, the FirstEnergy Nuclear Operating Company (FENOC) is submitting a request for exemption from certain requirements of 10 CFR 50, Appendix G, "Fracture Toughness Requirements," and 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events." The requested exemption would allow use of an alternate method, as described in Topical Report BAW-2308, Revisions 1-A and 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials," for determining the initial, unirradiated material reference temperatures of the Linde 80 weld materials present in the beltline region of the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS) reactor pressure vessel (RPV).

2.0 BACKGROUND

10 CFR 50, Appendix G(II)(D)(i) and 10 CFR 50.61(a)(5) require that the pre-service or unirradiated condition RT_{NDT} be evaluated according to the procedures in the American Society for Mechanical Engineers (ASME) Code, Section III, Paragraph NB-2331, which requires Charpy V-notch impact tests and drop weight tests.

Topical Report BAW-2308, Revisions 1-A and 2-A provide an NRC-approved alternate method for determining the initial, unirradiated material reference temperatures of the Linde 80 weld materials present in the beltline region of the DBNPS RPV. BAW-2308, Revision 2-A is a supplement to Revision 1-A, and incorporated additional test data and a re-evaluation of the reference temperature, T_0, determination, as requested by the NRC in the Safety Evaluation for Revision 1-A of the topical report. As stated in the NRC Safety Evaluation for Revision 2-A of the topical report dated March 24, 2008, the Conditions and Limitations, Items (1) through (4), contained in the Safety Evaluation for Revision 1-A of the topical report must be satisfied in order for licensees to reference the topical report in specific licensing applications.

Condition and Limitation (4) as stated in the NRC Safety Evaluation for BAW-2308, Revision 1-A:

> *Any licensee who wants to utilize the methodology of TR BAW-2308, Revision 1 as outlined in items (1) through (3) above, must request an exemption, per 10 CFR 50.12, from the requirements of Appendix G to 10 CFR Part 50 and 10 CFR 50.61 to do so. As part of a licensee's exemption request, the NRC staff expects that the licensee will also submit information which demonstrates what values the licensee proposes to use for ΔRT_{NDT} and the margin term for each Linde 80 weld in its RPV through the end of its facility's current operating license.*

In the above quotation, Condition and Limitation (1) provides criteria associated with the use of the NRC-accepted values of initial (unirradiated) reference temperature, IRT_{T0}, and the corresponding uncertainty term, σ_I, to define the initial heat-specific or generic properties of its facility's Linde 80 welds.

Condition and Limitation (2) requires that a minimum chemistry factor of 167.0°F be applied when the methodology of Regulatory Guide 1.99, Revision 2, is used to assess the shift in nil-ductility transition temperature due to irradiation.

Condition and Limitation (3) requires that a value of σ_Δ = 28.0°F be used to determine the margin term, as defined in BAW-2308, Revision 1-A, and Regulatory Guide 1.99, Revision 2.

DBNPS is planning to develop and implement revised pressure-temperature limit curves and low pressure overpressure protection (LTOP) system limits for operation beyond 21 Effective Full Power Years and desires to use the appropriate BAW-2308 initial RT_{NDT} values in the revision. Hence, exemption from certain requirements of 10 CFR 50, Appendix G is required. However, the DBNPS is not currently planning to revise the pressurized thermal shock (PTS) reference temperature (RT_{PTS}) values. Though not needed at this point, exemption from certain requirements, as described below, of 10 CFR 50.61 would be beneficial should the DBNPS decide to revise the RT_{PTS} in the future.

3.0 PROPOSED EXEMPTION

The exemption requested by FENOC addresses portions of the following regulations:

(1) 10 CFR Part 50, Appendix G which sets forth 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 system may be subjected over its service lifetime; and

(2) 10 CFR 50.61, which sets forth fracture toughness requirements for protection against pressurized thermal shock (PTS).

The exemption from Appendix G to 10 CFR 50 is to replace the required use of the existing Charpy V-notch and drop-weight-based methodology with the use of an alternate methodology that incorporates the use of fracture toughness test data for evaluating the integrity of the Linde 80 weld materials present in the DBNPS RPV beltline regions. The alternate methodology employs direct fracture toughness testing per the Master Curve methodology based on use of ASTM Standard Method E 1921 (1997 and 2002 editions), and ASME Code Case N-629. The exemption is required since 10 CFR 50, Appendix G requires that for the pre-service or unirradiated condition, RT_{NDT} be evaluated by Charpy V-notch impact tests and drop weight tests according to the procedures in the ASME Code, Paragraph NB-2331.

The exemption from 10 CFR 50.61 is to use an alternate methodology to allow the use of fracture toughness test data for evaluating the integrity of the Linde 80 weld materials present in the DBNPS RPV beltline regions, based on the use of the ASTM E 1921 (1997 and 2002 editions) and ASME Code Case N-629. The exemption is required because the methodology for evaluating RPV material fracture toughness in 10 CFR 50.61 requires that the pre-service or unirradiated condition be evaluated using Charpy V-notch impact tests and drop weight tests according to the procedures in the ASME Code, Paragraph NB-2331.

Additionally, the NRC Safety Evaluation for BAW-2308, Revision 1-A, concludes that an exemption is required to address issues related to 10 CFR 50.61 inasmuch as the BAW-2308, Revision 1-A, methodology, as modified and approved by the NRC staff, represents a significant change to the 10 CFR 50.61 methodology for determining the RT_{PTS} value for Linde 80 weld material. The changes in the BAW-2308, Revision 1-A, methodology with respect to the 10 CFR 50.61 methodology, include the requirements for use of a minimum chemistry factor of 167°F and a value of σ_{Δ} = 28.0°F for Linde 80 weld materials.

10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that: 1) the exemption is authorized by law, 2) the exemption will not result in an undue risk to public health and safety, 3) the exemption is consistent with the common defense and security, and 4) special circumstances, as defined in 10 CFR 50.12(a)(2) are present. The requested exemption to allow the use of BAW-2308, Revisions 1-A and 2-A, as the basis for the Linde 80 weld material initial properties at the DBNPS satisfy these requirements as described below.

1. The requested exemption is authorized by law.

   No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendix G when an exemption is granted by the Commission under 10 CFR 50.12.

   In addition, 10 CFR 50.61 permits other methods for use in determining the initial material properties provided such methods are approved by the Director, Office of Nuclear Reactor Regulation.

2. The requested exemption does not present an undue risk to the public health and safety.

   The proposed material initial properties basis described in BAW-2308, Revisions 1-A and 2-A represent an NRC-approved methodology for establishing IRT_{T0} values for Linde 80 welds. BAW-2308, Revisions 1-A and 2-A, includes conservatisms to ensure that use of the proposed initial material properties basis does not increase the probability of occurrence or the consequences of an accident at the DBNPS and will not create the possibility for a new or different type of accident that could pose a risk to public health and safety.

   The use of this proposed approach ensures that the intent of the requirements specified in 10 CFR 50, Appendix G and 10 CFR 50.61 are satisfied.

   The requested exemption is consistent with the NRC staff requirements specified in the Safety Evaluation for the approved BAW-2308, Revision 1-A and Revision 2-A; consequently, the exemption does not present an undue risk to the public health and safety.

3. The requested exemption will not endanger the common defense and security.

   The requested exemption is specifically concerned with RPV material properties and is consistent with NRC staff requirements specified in the Safety Evaluation Report for approved BAW-2308, Revision 1-A and Revision 2-A. Consequently, the requested exemption will not endanger the common defense and security.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50, Appendix G and 10 CFR 50.61.

   Pursuant to 10 CFR 50.12(a)(2), the NRC will not consider granting an exemption to the regulations unless special circumstances are present.

   The requested exemption meets the special circumstances of paragraph 10 CFR 50.12(a)(2)(ii) since application of these regulations in this particular circumstance is not necessary to achieve the underlying purpose of the regulations.

   The underlying purpose of 10 CFR 50, Appendix G and 10 CFR 50.61 is to protect the integrity of the reactor coolant pressure boundary by ensuring that each reactor vessel material has adequate fracture toughness. Application of ASME Section III, paragraph NB-2331 in the determination of initial material properties was developed based on the level of knowledge existing in the early 1970s concerning RPV materials.

Since the early 1970s, the level of knowledge concerning these topics has greatly expanded. This increased knowledge level permits relaxation of the ASME III NB-2331 requirements via application of BAW-2308, Revision 1-A and Revision 2-A, while maintaining the underlying purpose of the ASME Code and NRC regulations to ensure an acceptable margin of safety is maintained.

The proposed change in reactor vessel material initial properties will continue to satisfy the intent of 10 CFR 50, Appendix G and 10 CFR 50.61, thus justifying the exemption request. Issuance of an exemption from the criteria of these regulations to permit the use of BAW-2308, Revision 1-A and Revision 2-A for the DBNPS will not compromise the safe operation of the reactor, and will ensure that RPV integrity is maintained.

4.0 LINDE 80 WELD VALUES

As described in the NRC Safety Evaluation for BAW-2308, Revision 1-A, Condition and Limitation (4), the licensee is required to provide the values proposed for ΔRT_{NDT} and the margin term for each Linde 80 weld in its RPV through the end of its facility's current operating license. Table 1 provides these values. Though the values are valid through 52 EFPY, the end of service for the DBNPS is 32 EFPY.

Table 1

Beltline Materials	Material Ident.	Initial Reference Temperature °F	ΔRT_{NDT}, °F		Margin, °F		Adjusted Reference Temperature, °F	
			¼T Location	¾T Location	¼T Location	¾T Location	¼T Location	¾T Location
Nozzle Belt to Upper Shell Circ. Weld (OD 91%)	WF-233	-47.6	82.3	49.7	65.7	65.7	100.4	67.8
Upper Shell to Lower Shell Circ. Weld	WF-182-1	-80.2	177.4	127.6	59.0	59.0	156.2	106.4