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Docket Number 50-346

10CFR50.90

License Number NPF-3

Serial Number 2740

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

Subject: Davis-Besse Nuclear Power Station
License Amendment Application to Revise Technical Specification 3/4.4.4, "Reactor Coolant System - Pressurizer," to Adopt New Pressurizer Level Requirements (License Amendment Request No. 01-0012)

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, the following amendment is requested for the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS). The proposed amendment would revise Technical Specification (TS) 3/4.4.4, "Reactor Coolant System - Pressurizer " to adopt a new pressurizer high level limit and to revise the required action when the pressurizer is inoperable. Enclosure 1 to this letter contains the technical justification for these proposed changes and the proposed no significant hazards consideration determination.

Approval of the proposed amendment is requested by June 28, 2002, in order to align the TS pressurizer high level limit value with plant operating limits. The plant has implemented administrative controls to limit operation to the more conservative pressurizer high level limit. Once approved, the amendment shall be implemented within 120 days.

This proposed amendment has been prepared using the Nuclear Energy Institute guideline "Standard Format for Operating License Amendment Requests from Commercial Reactors."

A001

The proposed changes have been reviewed by the DBNPS Station Review Board and Company Nuclear Review Board.

Should you have any questions or require additional information, please contact Mr. David H. Lockwood, Manager - Regulatory Affairs, at (419) 321-8450.

Very truly yours,

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)
D. S. Simpkins, NRC Region III, DB-1 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 are the requested changes to the Davis-Besse Nuclear Power Station, Unit Number 1 Facility Operating License Number NPF-3.

The proposed changes concern:

Appendix A, Technical Specifications (TS):

TS 3/4.4.4, "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: [signature]
Guy G. Campbell, Vice President - Nuclear

Affirmed and subscribed before me this 30th day of November, 2001.

Nora L. Flood

Notary Public, State of Ohio - Nora L. Flood
My commission expires September 4, 2002.

Docket Number 50-346
License Number NPF-3
Serial Number 2740
Enclosure 1

DAVIS-BESSE NUCLEAR POWER STATION
EVALUATION
FOR
LICENSE AMENDMENT REQUEST NUMBER 01-0012

(14 pages follow)

DAVIS-BESSE NUCLEAR POWER STATION EVALUATION FOR LICENSE AMENDMENT REQUEST NUMBER 01-0012

Subject: License Amendment Request to Revise Technical Specification 3/4.4.4, "Reactor Coolant System - Pressurizer," to Adopt New Pressurizer Level Requirements

1.0 DESCRIPTION

This letter is a request to amend the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Facility Operating License Number NPF-3.

The proposed changes would revise the Operating License Technical Specification (TS) 3/4.4.4, "Reactor Coolant System - Pressurizer," to revise the pressurizer high level limit. Additionally, the Action required when the pressurizer is inoperable would be revised to make it more consistent with NUREG-1430, "Standard Technical Specifications - Babcock and Wilcox Plants," Revision 2 (NUREG-1430).

2.0 PROPOSED CHANGE

The proposed changes affect TS 3/4.4.4 and are shown on the marked-up TS page in Attachment 1.

The DBNPS staff has identified during a review of the design basis that the pressurizer level limits in TS Limiting Condition for Operation (LCO) 3.4.4 are not in agreement with the basis described in TS Bases Section 3/4.4.4, "Pressurizer." Specifically, TS Bases Section 3/4.4.4 currently states, in part:

> The high level limit is based on providing enough steam volume to prevent a pressurizer high level as a result of any transient.

Contrary to this statement, in the event of a loss of feedwater transient with the initial pressurizer level at the current TS high level limit, 305 inches, the pressurizer would go water solid. To address this non-conservative limit, the proposed changes would revise TS LCO 3.4.4 to reference a reduced high level limit of 228 inches. Additionally, the proposed changes would revise the TS LCO 3.4.4 Action statement to allow up to one hour to restore the pressurizer to operable status prior to taking action to place the plant in the Hot Standby operational mode. This change is consistent with NUREG-1430. The proposed revised TS LCO 3.4.4 would read as follows:

> The pressurizer shall be OPERABLE with:
>
> a. A steam bubble,
>
> b. A water level between 45 and 228 inches.
>
> <u>APPLICABILITY</u>: MODES 1 and 2.
>
> <u>ACTION</u>:
>
> With the pressurizer inoperable, restore the pressurizer to OPERABLE status within 1 hour or be in at least HOT STANDBY with the control rod drive trip breakers open within the next 6 hours.

In summary, the proposed changes would revise Technical Specification 3/4.4.4 to correct a non-conservative pressurizer high level limit and to adopt an allowed outage time for the pressurizer consistent with NUREG-1430, "Standard Technical Specification - Babcock and Wilcox Plants," Revision 2.

Related to this amendment application, TS Bases Section 3/4.4.4 is being revised to reflect the correct bases for the pressurizer level limits and the other proposed changes. The marked up TS Bases pages are provided in Attachment 3. Since the TS Bases are not a formal part of the Technical Specifications, these pages are being provided for information only. TS Bases changes are processed under the DBNPS Technical Specifications Bases Control Program.

3.0 BACKGROUND

The proposed changes affect the requirements for the pressurizer, which is a part of the reactor coolant system (RCS). The RCS is described in DBNPS Updated Safety Analysis Report (USAR) Section 5.0, "Reactor Coolant System." The RCS primarily consists of the reactor vessel, two vertical once-through steam generators, four shaft-sealed reactor coolant pumps, the electrically heated pressurizer, and interconnecting piping.

The pressurizer is described in DBNPS USAR Section 5.5.10, "Pressurizer." The pressurizer is a vertical-cylindrical vessel that is connected to the reactor outlet piping by the surge piping. The electrically heated pressurizer establishes and maintains reactor coolant pressure within prescribed limits and provides a surge chamber and a water reserve to accommodate changes in reactor coolant volume during operation. Overpressure protection for the pressurizer and the RCS is provided by two American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code safety valves. An additional pilot-operated relief valve (PORV) is provided to limit the lifting frequency of the code safety valves.

There are three redundant differential pressure type level transmitters to monitor the pressurizer level during pressurizer operation. Two of these three level transmitters are safety grade (Class 1E) transmitters. Each of these two essential transmitters is supplied by redundant essential 1E power supplies. These two essential transmitters feed two level indicators each (one per channel on the main control board located in the control room and one per channel on the Auxiliary Shutdown Panel). In addition to the above essential equipment, another level transmitter is fed by a non-essential uninterruptible power supply through a non-essential inverter. The pressurizer level indication to the operator is further supported by three redundant computer points fed by the above three transmitters.

4.0 TECHNICAL ANALYSIS

The proposed changes would revise TS 3/4.4.4, "Reactor Coolant System - Pressurizer," to replace the existing high level limit of 305 inches with a reduced high level limit of 228 inches. TS Bases Section 3/4.4.4, "Pressurizer," states the basis for the pressurizer high level limit is to prevent a pressurizer high level as a result of any transient. As determined from a review of the

pressurizer design basis, the existing high level limit of 305 inches will not prevent the pressurizer from going water solid during a loss of feedwater (LOFW) event, which is the most severe anticipated transient with respect to pressurizer insurges. It is desirable to prevent the pressurizer from going water solid to prevent the code safety valves and PORV from controlling RCS pressure by water relief rather than steam relief since water relief could potentially challenge valve reliability. Although the code safety valves may be capable of adequately controlling RCS pressure under water relief conditions, no credit is taken for code safety valve water relief.

The new high level limit of 228 inches will reduce the likelihood of the pressurizer going water solid during the most severe anticipated transient and enhance the reliability of RCS pressure control by the code safety valves and the PORV. The DBNPS LOFW transient was analyzed in Babcock and Wilcox document 32-1171148-00, *Davis-Besse Loss of Feedwater with 220 Inch Pressurizer Setpoint*. In this analysis, a RELAP5 model of the LOFW event was analyzed with an initial pressurizer level of 220 inches, which is the nominal controller setpoint for power operations. One auxiliary feedwater train was assumed to fail due to the single failure criterion. The analysis showed that when the pressurizer reached its peak level during the transient, 26 cubic feet of steam volume existed in the pressurizer. This 26 cubic feet of available steam volume corresponds to 8 inches of initial pressurizer level. Hence, a new high level limit of 228 (220 + 8) inches will ensure the pressurizer will not go water solid during a LOFW event initiated when operating below the new pressurizer high level limit. Consistent with the statements in NUREG-1430 Bases Section B 3.4.9, "Pressurizer," the new limit for pressurizer level is nominal and is not adjusted for instrument error since prevention of water relief is a goal for abnormal transient operation, rather than a safety limit.

Compliance with the revised TS 3/4.4.4 ensures that appropriate action will be taken when pressurizer level exceeds the limit at which water relief through the safety valves may occur during a LOFW event. The proposed new limit is more restrictive than the existing limit but is consistent with normal operating level. Therefore, the proposed change to the pressurizer high level limit will have no adverse effect on nuclear safety.

Additionally, the proposed changes would revise the TS 3/4.4.4 Action statement to provide a one-hour time period to restore the pressurizer to operable status prior to requiring the plant to be placed in the Hot Standby operational mode. The proposed one-hour allowed outage time will provide a reasonable amount of time to restore pressurizer level to within limits (and hence ensure a steam bubble) prior to requiring that a reactor shutdown commence. The one-hour allowed outage time is consistent with the required actions for NUREG-1430 Specification 3.4.9, "Pressurizer."

Compliance with the revised TS 3/4.4.4 Action statement will continue to ensure that appropriate actions are taken in timely manner to put the plant in a safe condition when the pressurizer becomes inoperable. Therefore, the proposed change to the TS 3/4.4.4 Action statement will have no adverse effect on nuclear safety.

The proposed changes to the TS 3/4.4.4 Action statement to allow a one-hour time period to restore the pressurizer level is consistent with NRC guidance in NUREG-1430. In accordance

with the guidance of Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," proposed TS changes that are consistent with approved NRC staff positions do not require probabilistic risk information to be submitted in support of the proposed changes.

In summary, the pressurizer's reduced high level limit of 228 inches has been analyzed and found to be acceptable in ensuring the pressurizer will not go water solid during a LOFW event. The one-hour time period to restore the pressurizer level to within limits provides a reasonable restoration time consistent with NUREG-1430 prior to requiring commencement of a reactor shutdown. Accordingly, these proposed changes do not have an adverse effect on nuclear safety.

5.0 REGULATORY SAFETY ANALYSIS

5.1 <u>No Significant Hazards Consideration</u>

The proposed amendment would revise Technical Specification (TS) requirements for the reactor coolant system pressurizer. The proposed change would revise the pressurizer high level limit to adopt a more restrictive limit. Additionally, the proposed change would provide one hour to restore the pressurizer to operable status prior to commencing reactor shutdown, when it has been declared inoperable.

An evaluation has been performed to determine 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 new pressurizer high level limit is more restrictive than the existing limit, and accident initial conditions, probability, and assumptions remain as previously analyzed. The proposed change to the pressurizer allowed outage time will have no significant effect on accident initiation frequency. The proposed changes do not invalidate the assumptions used in evaluating the radiological consequences of any accident. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes do not introduce any new or different accident initiators. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed change to the pressurizer high level limit will ensure an adequate margin of safety is maintained. The proposed change to the pressurizer allowed outage time is minimal and will not have a significant effect on any margin of safety. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, it is concluded that the proposed amendment presents no 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.

5.2 Applicable Regulatory Requirements/Criteria

The regulatory basis for TS 3/4.4.4 is to ensure the pressurizer is capable of maintaining the RCS such that it is: not water solid, capable of accomodating pressure surges during operations, and protecting the code safety valves and PORV against water relief.

The applicable design criterion for the pressurizer is described in the DBNPS Updated Safety Analysis Report (USAR) Appendix 3D.1.11, "Criterion 15 - Reactor Coolant System Design." DBNPS USAR Appendix 3D.1.11 states, in part:

> The Reactor Coolant System (RCS) and associated auxiliary, control, and protection systems are designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

The TS 3/4.4.4 pressurizer high level limit requirement ensures that adequate steam volume is available to prevent water relief through the pressurizer code safety valves and PORV following the most limiting transient, which is loss of feedwater. By ensuring only steam relief through these valves, which are part of the reactor coolant pressure boundary (RCPB), the reliability of the valves is

maintained. The proposed pressurizer requirements will ensure that design requirements of the valves and the RCPB are not exceeded.

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 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.

6.0 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 effluent 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.

7.0 REFERENCES

1. DBNPS Operating License NPF-3, Appendix A Technical Specifications through Amendment 247.

2. DBNPS Updated Safety Analysis Report through Revision 22.

3. NUREG-1430, *Standard Technical Specifications - Babcock and Wilcox Plants*, Revision 2.

4. B&W Document 32-1171148-00, *Davis-Besse Loss of Feedwater with 220 Inch Pressurizer Setpoint,* dated March 14, 1988.

5. Regulatory Guide 1.177, *An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications*, Revision 0, dated August 1998.

8.0 ATTACHMENTS

1. Proposed Mark-Up of Technical Specification Pages

2. Proposed Retyped Technical Specification Pages

3. Technical Specification Bases Pages

**PROPOSED MARK-UP
OF
TECHNICAL SPECIFICATION PAGES**

(1 page follows)

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~~ 228 inches.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the pressurizer inoperable, restore the pressurizer to OPERABLE status within 1 hour or be in at least HOT STANDBY with the control rod drive trip breakers open within the next 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.

**PROPOSED RETYPED
TECHNICAL SPECIFICATION PAGES**

(1 page follows)

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 228 inches.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the pressurizer inoperable, restore the pressurizer to OPERABLE status within 1 hour or be in at least HOT STANDBY with the control rod drive trip breakers open within the next 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.

TECHNICAL SPECIFICATION BASES PAGES

(1 page follows)

Note: The Bases page is provided for information only.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and pilot operated relief valve against water relief.

The low level limit is based on providing enough water volume to prevent the low level interlock from de-energizing the pressurizer heaters during steady state operations.~~a reactor coolant system low pressure condition that would actuate the Reactor Protection System or the Safety Feature Actuation System.~~ The high level limit is based on providing enough steam volume to prevent water relief through the pressurizer relief valves during the most challenging anticipated pressurizer insurge transient, which is a loss of feedwater.~~a pressurizer high level as a result of any transient.~~ Since prevention of water relief is a goal for abnormal transient operation, rather than a Safety Limit, the value for pressurizer level is nominal and is not adjusted for instrument error.

The ACTION statement provides 1 hour to restore pressurizer level prior to requiring shutdown. The 1-hour completion time is considered to be a reasonable time for restoring pressurizer level to within limits.

The pilot operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the pilot operated relief valve minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken. A process equivalent to the inspection method described in Topical Report BAW-2120P will be used for inservice inspection of steam generator tube sleeves. This inspection will provide ensurance of RCS integrity.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these chemistry limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 150 GPD through any one steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal

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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 FOR 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
None	N/A