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Mr. John P. Stetz
Vice President - Nuclear, Davis-Besse
Centerior Service Company
c/o Toledo Edison Company
300 Madison Avenue
Toledo, Ohio 43652

SUBJECT: AMENDMENT NO. 192 TO FACILITY OPERATING LICENSE NO. NPF-3 - DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 (TAC NO. M84541)

Dear Mr. Stetz:

The Commission has issued Amendment No. 192 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The amendment revises the Technical Specifications (TS) in response to your application dated September 3, 1992, as supplemented on August 22, 1994.

This amendment revises TS 3/4.4.5 and its corresponding Bases to include the maximum allowable steam generator level as a variable limit based on the plant's mode of operation. The maximum steam generator level is based on initial conditions specified in the Updated Safety Analysis Report. This amendment also revises TS 3.1.1.1 by adding a footnote on additional requirements for SHUTDOWN MARGIN in Mode 3.

A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by Linda L. Gundrum
Linda L. Gundrum, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures: 1. Amendment No. 192 to License No. NPF-3
2. Safety Evaluation

cc w/encl: See next page

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Mr. John P. Stetz
Toledo Edison Company

Davis-Besse Nuclear Power Station
Unit No. 1

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TOLEDO EDISON COMPANY

CENTERIOR SERVICE COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 192
License No. NPF-3

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Toledo Edison Company, Centerior Service Company, and the Cleveland Electric Illuminating Company (the licensees) dated September 3, 1992, as supplemented on August 22, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-3 is hereby amended to read as follows:

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(a) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 192, are hereby incorporated in the license. The Toledo Edison Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented not later than 90 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Cynthia A Carpenter

Cynthia A. Carpenter, Acting Director
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: October 7, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 192

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3/4 1-1	3/4 1-1
3/4 4-6	3/4 4-6
-	3/4 4-6a
-	3/4 4-6b
B 3/4 4-3	B 3/4 4-3

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 1% $\Delta k/k$.

APPLICABILITY: MODES 1, 2*, 3**, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN < 1% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 25 gpm of 7875 ppm boron or its equivalent, until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 1% $\Delta k/k$:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2#, at least once per 12 hours, by verifying that regulating rod groups withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2## within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading by consideration of the factors of e. below, with the regulating rod groups at the maximum insertion limit of Specification 3.1.3.6.

#With $k_{eff} \geq 1.0$
##With $k_{eff} < 1.0$
*See Special Test Exception 3.10.4
**See LCO 3.4.5, Steam Generators, for additional SHUTDOWN MARGIN requirements.

LIMITING CONDITION FOR OPERATION

3.4.5 Each Steam Generator shall be OPERABLE with a minimum water level of 18 inches and the maximum specified below as applicable:

MODES 1 and 2:

a. The acceptable operating region of Figure 3.4-5.

MODE 3*:

b. 50 inches Startup Range with the SFRCS Low Pressure Trip bypassed and one or both Main Feedwater Pump(s) capable of supplying Feedwater to any Steam Generator.

c. 96 percent Operate Range with:

1. The SFRCS Low Pressure Trip active.

   Or

2. The SFRCS Low Pressure Trip bypassed and both Main Feedwater Pumps incapable of supplying Feedwater to the Steam Generators.

MODE 4:

d. 625 inches Full Range Level

APPLICABILITY: MODES 1, 2, 3, and 4, as above.

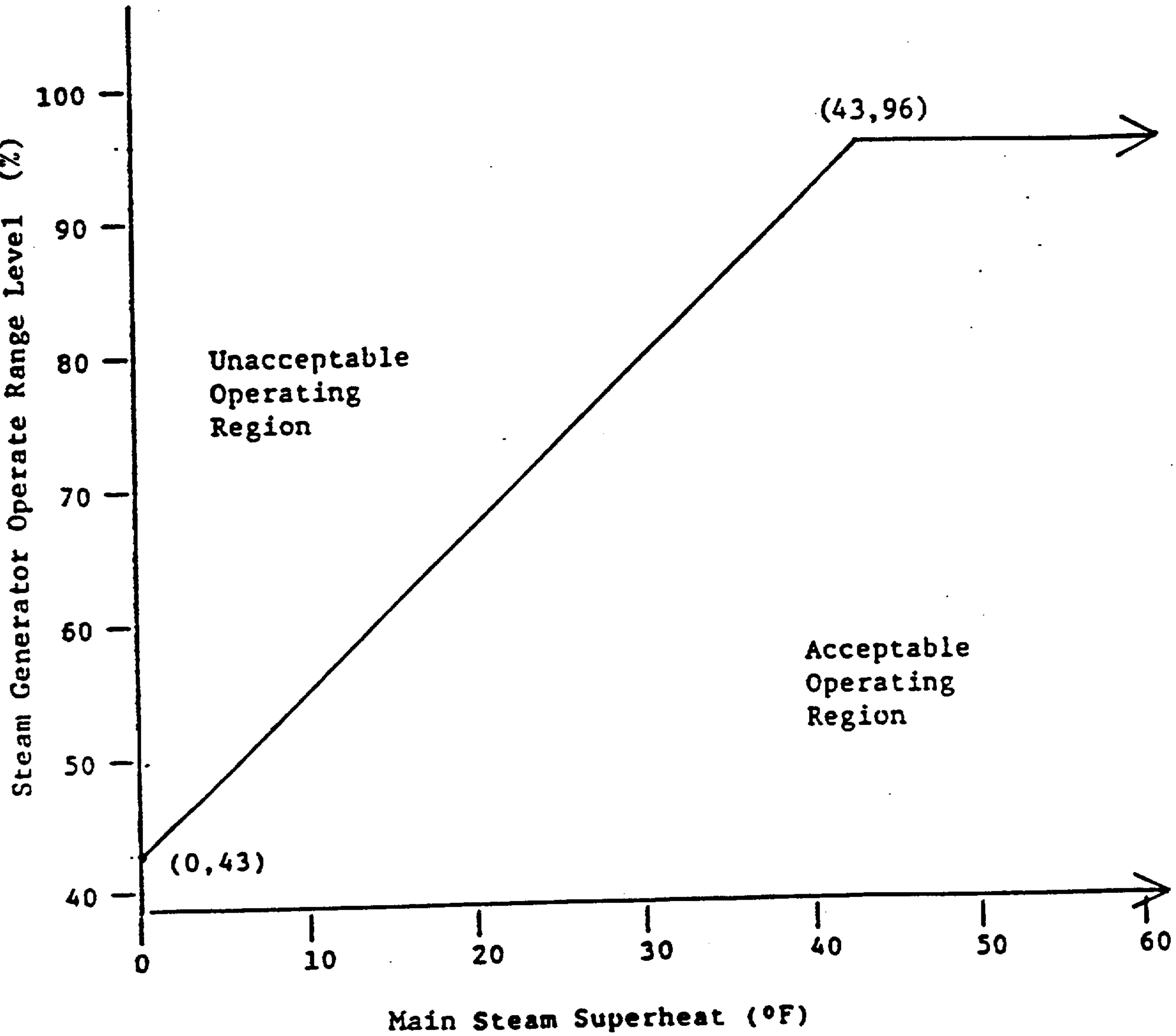
ACTION:

a. With one or more steam generators inoperable due to steam generator tube imperfections, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

b. With one or more steam generators inoperable due to the water level being outside the limits, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

*Establish adequate SHUTDOWN MARGIN to ensure the reactor will stay subcritical during a MODE 3 Main Steam Line Break.

Figure 3.4-5
Maximum Allowable Steam Generator Level
in MODES 1 and 2



SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4-4.1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. The first sample inspection during each inservice inspection (subsequent to the baseline inspection) of each steam generator shall include:

   1. All tubes or tube sleeves that previously had detectable wall penetrations (> 20%) that have not been plugged or repaired by sleeving in the affected area.

   2. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems.

operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 1 GPM can be detected by monitoring the secondary coolant. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired by sleeving in the affected areas.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. As described in Topical Report BAW-2120P, degradation as small as 20% through wall can be detected in all areas of a tube sleeve except for the roll expanded areas and the sleeve end, where the limit of detectability is 40% through wall. Tubes with imperfections exceeding the repair limit of 40% of the nominal wall thickness will be plugged or repaired by sleeving the affected areas. Davis-Besse will evaluate, and as appropriate implement, better testing methods which are developed and validated for commercial use so as to enable detection of degradation as small as 20% through wall without exception. Until such time as 20% penetration can be detected in the roll expanded areas and the sleeve end, inspection results will be compared to those obtained during the baseline sleeved tube inspection.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results shall be reported to the Commission prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

The steam generator water level limits are consistent with the initial assumptions in the USAR. While in MODE 3, examples of Main Feedwater Pumps that are incapable of supplying feedwater to the Steam Generators are tripped pumps or a manual valve closed in the discharge flowpath. The reactivity requirements to ensure adequate SHUTDOWN MARGIN are provided in plant operating procedures.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 192 TO FACILITY OPERATING LICENSE NO. NPF-3

TOLEDO EDISON COMPANY

CENTERIOR SERVICE COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

1.0 INTRODUCTION

By letter dated September 3, 1992, as supplemented on August 22, 1994, Toledo Edison Company (the licensee), filed an application for an amendment requesting a modification to the Davis-Besse Nuclear Power Station (DBNPS), Unit 1, Technical Specifications (TS). The submittal addressed changes to TS 3.1.1.1 (Shutdown Margin Limiting Condition for Operation), TS 3.4.5 (Steam Generators Limiting Condition for Operation) and TS Bases 3/4.4.5 (Steam Generators). The August 22, 1994, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The proposed change to the TS 3.4.5 Limiting Condition for Operation (LCO) would modify the maximum allowable steam generator water level in Modes 1, 2, 3, and 4. The current LCO specifies a maximum water level of 348 inches for these modes of operation. The proposed changes would establish the following limitations for steam generator water level.

> During MODES 1 and 2, both the acceptable and unacceptable operating regions for steam generator level are shown on Figure 3.4-5 (new). This figure presents the maximum Steam Generator level on the Operate Range level instrumentation expressed in percent as a function of Main Steam Superheat in °F.
>
> During Mode 3, with the Steam and Feedwater Rupture Control System (SFRCS) Low Pressure Trip bypassed and one or both Main Feedwater Pump(s) capable of supplying feedwater to any Steam Generator, the maximum Steam Generator water level shall be 50 inches on the Startup Range level instrumentation.
>
> During Mode 3, with either the SFRCS Low Pressure Trip active or the SFRCS Low Pressure Trip bypassed and both Main Feedwater Pumps incapable of supplying feedwater to the Steam Generators, the maximum Steam

Generator water level shall be 96 percent on the Operate Range level instrumentation.

During MODE 4 the maximum Steam Generator water level shall be 625 inches on the Full Range level instrumentation.

The proposed change to TS 3.1.1.1 will add a footnote to Mode 3 in APPLICABILITY. The footnote will reference TS 3.4.5 for additional SHUTDOWN MARGIN requirements.

The proposed changes will permit the maximum steam generator level to be a variable limit based on the plant's operating MODE. The changes will allow the plant to produce full power as steam generator fouling occurs, without compromising the margins to safety limits, while ensuring the plant response to accident conditions remain acceptable.

The proposed change to the Bases Section 3/4.4.5 revises the text of the Bases to show that the design basis of the level requirements is in the Updated Safety Analysis Report (USAR). The original assumptions of the Final Safety Analysis Report are included in the USAR, so there is no loss of information regarding the permissible steam generator water levels. Examples of incapable main feedwater pumps are also proposed in the revised Bases text. This change has no adverse effects on safety.

2.0 BACKGROUND

The Babcock and Wilcox Nuclear Steam Supply System at DBNPS employs a Once-Through-Steam-Generator. This is a counterflow shell and tube heat exchanger with the secondary boiling mixture on the shell-side and the reactor coolant on the tube side. Indications for the steam generator shell-side water level are derived from differential pressure measurements taken from different regions of the steam generator. Since this level is determined through pressure measurements, the reading inherently includes the effects from frictional losses between the level taps. At higher reactor power levels, the steam generator water level is a combination of the shell side-mass and the frictional and momentum effects of fluid flow through the steam generator. With increased steam generator fouling, the present maximum specified steam generator level could be the factor limiting reactor output to less-than-full power. The change to TS 3.4.5 is proposed to allow the plant to continue to produce full power with increased steam generator fouling.

When not at full power, the fouling effects are diminished. Changes for maximum water level in Modes 3 and 4 are proposed to improve the process of obtaining desired steam generator water chemistry (Mode 3), to minimize the exposure of the steam generator internals to oxygen, thus limiting corrosion (Mode 4), and to ensure the plant response to accident conditions is acceptable and the margins to safety limits are maintained (all Modes). The proposed change to TS 3.1.1.1 includes a reference to additional SHUTDOWN MARGIN requirements in TS 3.4.5 when operating in Mode 3.

3.0 EVALUATION

The primary concern associated with the proposed modification to steam generator water level is the additional inventory released during an accident condition. The safety evaluation provided by the licensee addressed the pertinent scenarios and determined the impact which the proposed changes have on safety. The licensee evaluated consequences of altering the maximum steam generator water level in the following situations:

1) The effects on containment integrity for accidents inside containment,

2) The consequences of accidents for Auxiliary Building environments,

3) Reactivity and core cooling effects for all accidents, and

4) Radiological consequences for all accidents.

The evaluation was performed assuming the plant was in Modes 1, 2, 3, and 4. Since Mode 4, operation is such a low-energy state, it was unnecessary to evaluate this condition for several situations.

3.1 Accidents Within Containment

The proposed changes have no effect on containment integrity while in Modes 1, 2, 3, and 4. The analysis of a Main Steam Line Break (MSLB) inside containment is given in USAR Chapter 6.2, "Containment Systems" and Chapter 15.4.4, "Steam Line Break." Calculations by the licensee indicate that the proposed changes to steam generator maximum level will limit mass and energy releases below those determined in the USAR for MODES 1, 2, and 4, as a result of an MSLB in containment.

In MODE 3, the SG inventory is limited to 50 inches on the Startup Range level instrumentation, if a Main Feedwater Pump is capable of supplying water to the SG and the SFRCS Low Pressure Trip is bypassed. This limits the amount of energy available for release to the containment to less than that released during an MSLB at 100 percent Full Power.

When the SFRCS Low Pressure Trip is protecting the plant, or once the possible feedwater flow to the SG is limited to that available from the Motor Driven Feedwater Pump (MDFP), the mass permitted in the SG may be increased until the Operate Range SG level indicates 96 percent.

The analyses performed for MODE 3 assumed the MDFP (the worst single failure) continues for 10 minutes after the MSLB. An MSLB inside containment, while the plant is operating in Mode 3, results in higher mass releases and lower energy input to containment than that determined in the USAR for a Mode 1 MSLB. The net effect from the mass and energy addition is less than the mass and energy released by the MSLB analyzed in the USAR.

For operation in MODES 1, 2, 3, and 4, the containment pressure and temperature response will be bounded by the USAR results. Therefore, containment integrity will not be more severely challenged.

3.2 Effects of Accidents on Auxiliary Building Environments

Several pipe breaks outside of containment have been evaluated to determine their potential impact on the environmental qualification profiles of important safety equipment. The significant line breaks which were evaluated included an MSLB, main feedwater line break, main steam to auxiliary feed pump turbines line break, and Steam Generator Blowdown System line break.

The calculated mass and energy releases for the line breaks evaluated were less than those stated in the USAR, except for one case. In Mode 3, with an initial level of 96 percent Operating Range level instrumentation and with the MDFP supplying the SGs or with the SFRCS Low Pressure Trip active, the mass of water released is larger than was assumed in the analysis referenced by USAR Section 3.6.2.7.1.5. However, the energy content of the steam exiting the break is always lower at any given time in the transient, because the transient begins in MODE 3. As a result, the environmental effects are judged to be no more severe than the cases currently presented in the USAR.

3.3 Reactivity and Core Cooling Concerns

A MSLB with increased inventory in the steam generators would result in rapid overcooling of the RCS, thereby adding positive reactivity to the reactor. Administrative controls are included in the proposed TS to ensure adequate SHUTDOWN MARGIN to prevent the reactor from attaining criticality during any postulated MSLB. The licensee addressed reactivity concerns for accidents affected by the proposed changes in steam generator maximum water level, as discussed below.

The worst case situation is an MSLB during MODE 3 with the steam generators at the maximum level, the SFRCS bypassed, and the MDFP supplying feedwater to the SGs. All other situations are bounded by this scenario, except the case of the Main Feedwater Pumps supplying the SG's with the SFRCS Low Pressure Trip bypassed, which has also been evaluated for cooldown effects. Although both main feedwater pumps will be capable, neither will be operating in MODE 3. Based on the inadvertent operation of one pump, the SG level allowed will be limited to 50 inches as indicated on the Startup Range instrumentation and with sufficient SHUTDOWN MARGIN provided, with one or both feedwater pumps capable of supplying feedwater.

Administrative control requirements, as well as a specific footnote in TS 3.4.5, will ensure that there is adequate SHUTDOWN MARGIN to prevent the reactor from becoming critical during any MODE 3 MSLB. The footnote in TS 3.1.1.1 serves to reference the additional SHUTDOWN MARGIN requirements in TS 3.4.5. The administrative controls include determining the boron concentration required to compensate for the calculated cooldown and procedural requirements to establish the necessary boron concentration in the RCS prior to raising the SG level, above the low level limits. During MODES 1 and 2, with maximum water levels in the steam generators, the plant response is bounded as described in the USAR. When the plant is in MODE 4, the SGs can only induce a very limited cooldown of the RCS following any secondary line breaks. Therefore, no additional reactivity requirements are needed.

The assumptions in the USAR Section 15.4.4.2.3 related to Departure from Nucleate Boiling Ratio (DNBR) are consistent with the mass in the steam generators, under the proposed change in Modes 1 and 2. In Modes 3 and 4, the departure from nucleate boiling cannot occur due to the low heat flux of the reactor. Consequently, the proposed change has no effect on keeping the reactor fuel adequately cooled.

3.4 Radiological Consequences for All Accidents

Of the accidents evaluated in Chapter 15 of the USAR, only two were potentially affected by steam generator inventory levels. These are the Steam Generator Tube Rupture (SGTR) and the Steam Line Break accidents.

A review of the assumptions of the USAR Chapter 15.4.2 analysis involving SGTR indicates that the total mass and radioactive inventory released are independent of the initial steam generator levels. Consequently, the proposed changes do not result in releases, above those described in the USAR, for this accident.

The radiological consequences of an MSLB bound all other steam line breaks. The licensee has calculated the radiological releases at the higher proposed steam generator water levels. Results determined with an extensive degree of conservatism indicate that the thyroid dose resulting from an MSLB increases from 0.79 REM, currently in the USAR to 0.95 REM, under the proposed changes. This result is well below the limits specified in 10 CFR Part 100.

3.5 Evaluation Summary

The licensee has demonstrated that the proposed changes to TS 3.1.1.1 to footnote additional SHUTDOWN MARGIN requirements for Mode 3, to TS 3.4.5 to include a variable maximum water level based on the MODE of operation, and the changes to the TS Bases 3/4.4.5, do not significantly impact safety. Therefore, the NRC staff finds the proposed changes to be acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (57 FR 48830). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: P. Rush
L. Gundrum

Date: October 7, 1994