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

March 31, 2014

Mr. Raymond A. Lieb
Site Vice President
FirstEnergy Nuclear Operating Company
Mail Stop A-DB-3080
5501 North State, Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RELATED TO STEAM GENERATOR INVENTORY CHANGE (TAC NO. MF0536)(L-13-040)

Dear Mr. Lieb:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 287 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 January 18, 2013, as supplemented by letters dated September 27, and December 13, 2013, and January 10, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML13018A350, ML13273A356, ML13350A594, and ML14010A147, respectively). This amendment revises TS 3.4.17, "Steam Generator (SG) Tube Integrity," TS 3.7.18, "Steam Generator Level," TS 5.5.8, "Steam Generator Program," and TS 5.6.6, "Steam Generator Tube (SGTIR) Inspection Report." The revision of these TSs support replacement of the original SGs during a refueling outage in April 2014. The changes to TS 3.4.17, TS 5.5.8, and TS 5.6.6 include adoption of the program improvements in Technical Specifications Task Force (TSTF)-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," which was approved by the NRC on October 27, 2011.

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

Sincerely,

/ RA /

Eva A. Brown, Senior Project Manager
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures:
1. Amendment No. 287 to NPF-3
2. Safety Evaluation

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

FIRSTENERGY NUCLEAR OPERATING COMPANY

AND

FIRSTENERGY NUCLEAR GENERATION CORP.

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

DOCKET NO. 50-346

Amendment No. 287
License No. NPF-3

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

   A. The application for amendment filed by FirstEnergy Nuclear Operating Company et al. (the licensee), dated January 18, 2013 as supplemented by letters dated September 27, and December 13, 2013, and January 10, 2014, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 287, are hereby incorporated in the license. FENOC 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 within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Travis L. Tate

Travis L. Tate, Chief
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Facility Operating License

Date of Issuance: March 31, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 287

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Replace the following pages of the Facility Operating License and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License NPF-3	License NPF-3
Page 4	Page 4
TS pages	TS pages
3.4.17-1	3.4.17-1
3.4.17-2	3.4.17-2
3.7.18-1	3.7.18-1
3.7.18-3	3.7.18-3
5.5-5	5.5-5
5.5-6	5.5-6
5.5-7	5.5-7
5.5-8	5.5-8
5.6-4	5.6-4

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. 287, 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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube plugging criteria shall be plugged in accordance with the Steam Generator Program.

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

ACTIONS

---NOTE---
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection. AND A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	7 days Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

3.7 PLANT SYSTEMS

3.7.18 Steam Generator Level

LCO 3.7.18 Water Level of each steam generator shall be:

a. Less than or equal to the maximum water level shown in Figure 3.7.18-1 when in MODE 1 or 2;

b. ≤ 96% Operate Range with LCO 3.3.11, "Steam and Feedwater Rupture Control System (SFRCS) Instrumentation," Function 1 (Main Steam Line Pressure - Low) not bypassed when in MODE 3;

c. ≤ 74% Operate Range with LCO 3.3.11, Function 1 bypassed and both main feedwater (MFW) pumps not capable of supplying feedwater to the steam generators when in MODE 3; and

d. ≤ 50 inches Startup Range with LCO 3.3.11, Function 1 bypassed and one MFW pump capable of supplying feedwater to the steam generators when in MODE 3.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
Enter applicable Conditions and Required Actions of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," when high steam generator water level results in exceeding the SDM limits.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Water level in one or more steam generators not within limits.	A.1	Restore steam generator level to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 4.	12 hours

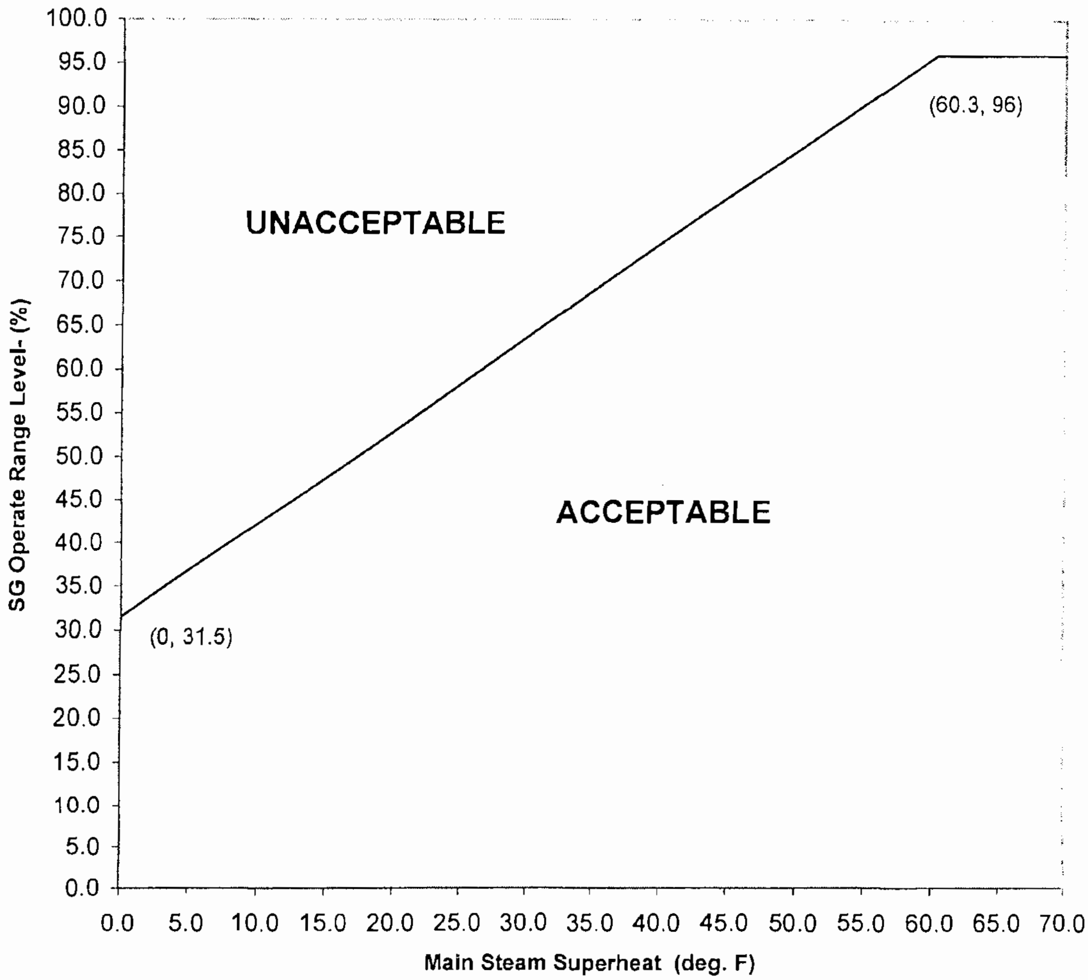

Figure 3.7.18-1 (page 1 of 1)
Maximum Allowable Steam Generator Level

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2 and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

   a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

   b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

   c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

   d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

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

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of safety related filter ventilation systems in accordance with Regulatory Guide 1.52, Revision 2, ANSI/ASME N510-1980, and ASTM D 3803-1989.

a. Demonstrate for each of the safety related systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI/ASME N510-1980 at the system flowrate specified below.

Safety Related Ventilation System	Flowrate (cfm)
Station Emergency Ventilation System (EVS)	≥ 7200 and ≤ 8800
Control Room Emergency Ventilation System (CREVS)	≥ 2970 and ≤ 3630

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. 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 during the inspection outage for each degradation mechanism;

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG;

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

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.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 287 TO FACILITY OPERATING LICENSE NO. NPF-3

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

1.0 INTRODUCTION

By application to the U.S. Nuclear Regulatory Commission (NRC, the Commission) dated January 18, 2013, as supplemented by letters dated September 27, and December 13, 2013, and January 10, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML13018A350, ML13273A356, ML13350A594, and ML14010A147, respectively), FirstEnergy Nuclear Operating Company (the licensee) requested changes to the technical specifications (TSs) for the Davis-Besse Nuclear Power Station (DBNPS), Unit No. 1 The proposed changes would affect TS 3.4.17, "Steam Generator (SG) Tube Integrity," TS 3.7.18, "Steam Generator Level," TS 5.5.8, "Steam Generator Program," and TS 5.6.6, "Steam Generator Tube (SGTIR) Inspection Report." The revision of these TSs was requested to support replacement of the original SGs during a refueling outage in April 2014.

The September 27 and December 13, 2013; and January 10, 2014, supplements, contained clarifying information within the scope of the proposed action noticed and did not change the NRC staff's initial proposed finding of no significant hazards consideration published by the staff in the *Federal Register* on March 19, 2013 (78 FR 16876, and 16883).

2.0 REGULATORY EVALUATION

2.1 Regulatory Requirements

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. In Section 50.36, "Technical specifications," of Title 10 of the *Code of Federal Regulations* (10 CFR), the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting condition for

Enclosure

operations (LCOs), (3) surveillance requirements (SRs), (4) design features, and (5) administrative controls.

The licensee provided TS BASES revisions, which identified the design basis accident (DBA) that is significant with respect to the proposed revisions to TS 3.7.18. The BASES state that the "most limiting Design Basis Accident that would be affected by SG operating level is a main steam line failure... This LCO is required to preserve the initial condition assumptions of the accident analyses." The NRC staff determined, by reviewing the BASES, that regulatory guidance related to TS LCOs is applicable, as is guidance for reviewing main steam line failures.

The regulations in 10 CFR 50.36(c)(2)(i), state, in part, that the "[l]imiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility." Additionally, 10 CFR 50.36(c)(2)(ii) provides four criteria and requires LCOs to be established for each item meeting one of those criteria. Criterion 2 is a "process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The LCOs for TS 3.7.18 satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

The fundamental regulatory requirements with respect to the integrity of the SG tubing are found in the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50. The GDC states that the reactor coolant pressure boundary (RCPB) shall have "an extremely low probability of abnormal leakage ... and gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDCs 15 and 31), shall be of "the highest quality standards possible" (GDC 30), and shall be designed to permit "periodic inspection and testing ... to assess ... structural and leak tight integrity" (GDC 32). To this end, 10 CFR 50.55a(c) specifies that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). Section 50.55a(g) further requires that components and supports that are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice examination of these components and must meet the pre-service examination requirements set forth in the editions and addenda of Section XI, "Rules for Inservice Inspection [ISI] of Nuclear Power Plant Components," of the ASME Code incorporated by reference in 10 CFR 50.55a(b) that were applied to the construction of the particular component. Section XI requirements pertaining to the ISI of SG tubing are augmented by additional SG tube SRs in the TSs.

As part of the plant licensing basis, applicants for pressurized-water reactor (PWR) licenses are required to analyze the consequences of postulated DBAs such as an SG tube rupture (SGTR) and main steam line break (MSLB). These analyses consider the primary-to-secondary leakage that may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of the 10 CFR Part 100.11 guidelines for offsite doses (or 10 CFR 50.67, as appropriate), GDC-19 criteria for control room operator doses, or some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis.

It is required in TS 5.5.8 "Steam Generator (SG) Program" for DBNPS, Unit 1, that an SG program be established and implemented to ensure that SG tube integrity is maintained. Tube integrity is maintained by meeting specified performance criteria for structural and leakage

integrity consistent with the plant design and licensing bases. To confirm that the performance criteria are being met, TS 5.5.8.a requires a condition monitoring assessment be performed during each outage in which the SG tubes are inspected or plugged. TS 5.5.8.d includes provisions regarding the scope, frequency, and methods of SG tube inspections.

Appendix A to 10 CFR Part 50 requires in (1) GDC 27, that reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (2) GDC 28, that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; (3) GDC 31, that the RCPB be designed with sufficient margin to assure that, under specified conditions, it will behave in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and (4) GDC 35, that the reactor cooling system and associated auxiliaries be designed to provide abundant emergency core cooling. Specific review criteria are contained in Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

The proposed changes to TSs 3.4.17, 5.5.8, and 5.6.6 include adoption of the program improvements in Technical Specifications Task Force (TSTF)-510, Rev. 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection." This TSTF was approved by the NRC on October 27, 2011. A notice related to the availability of TSTF-510 Rev. 2, for use as part of the consolidated line item improvement process (CLIIP) was published in the *Federal Register* on October 27, 2011 (76 FR 66763). Because the submittal includes TS revisions for both SG replacement and implementation of TSTF-510, this amendment request is not being processed as a CLIIP amendment.

The current Standard Technical Specification (STS) requirements that are a part of TSTF-510, Rev. 2, were established in May 2005, with the NRC staff's approval of TSTF-449, Rev. 4, "Steam Generator Tube Integrity" (70 FR 24126). The TSTF-449 changes to the STS incorporated a new, largely performance-based approach for ensuring the integrity of the SG tubes is maintained. The performance-based requirements were supplemented by prescriptive requirements relating to tube inspections and tube repair limits to ensure that conditions adverse to quality are detected and corrected on a timely basis. As of September 2007, the TSTF-449, Rev. 4, changes were adopted in the plant TS for all PWRs.

The proposed changes in TSTF-510, Rev. 2, reflect licensees' early implementation experience with respect to the TSTF-449, Rev. 4. TSTF-510 characterizes the changes as editorial corrections, changes, and clarifications intended to improve internal consistency, consistency with implementing industry documents, and usability without changing the intent of the requirements. Changes consistent with this TSTF, are an improvement to the existing SG inspection requirements and continue to provide assurance that the plant licensing basis will be maintained between SG inspections.

2.2 System Descriptions

The SGs are described in Section 5.5.2.2 of the DBNPS Updated Final Safety Analysis Report (UFSAR). Section 5.5.2 of the UFSAR states:

> The steam generator is a vertical, straight-tube-and-shell heat exchanger which produces superheated steam at approximately a constant pressure over the power range. Reactor coolant flows downward through the tubes, and steam is generated on the shell side. The high-pressure parts of the unit are the hemispherical heads, the tubesheets, and the straight ... tubes between the tubesheets. Tube supports hold the tubes in a uniform pattern along their length.
>
> The shell, outside of the tubes, and the tubesheets form the boundaries of the steam-producing section of the vessel. Within the shell, the tube bundle is surrounded by a baffle, which is divided into two sections. The upper part of the annulus between the shell and baffle is the superheater outlet, and the lower part is the feedwater inlet-heating zone.

The UFSAR excerpt indicates that the SG serves two functions that have safety importance are: (1) SG internals comprise a part of the RCPB, and (2) outside the pressure boundary, superheated steam is produced in the generator. The RCPB is significant in that it comprises a fission product barrier, and the superheated steam is significant in that it is a vehicle for energy transfer during a postulated MSLB. Although there are materials and secondary side dimensional and thermal performance differences, the general design of the replacement SGs will be similar to the existing ones.

3.0 TECHNICAL EVALUATION

3.1 Methodology and Analysis for the TS LCO 3.7.18

The enclosure to the licensee's January 18, 2013, application describes the requested changes to TS 3.7.18 as follows:

> TS 3.7.18, "Steam Generator Level," establishes limit on SG level based on operating MODE and plant operating conditions. The four restrictions currently established in LCO 3.7.18 are based on the specific physical design characteristics and dimensions of the original SGs. Because of differences in design characteristics and dimensions of the replacement SGs, TS 3.7.18 would be revised to ensure appropriate secondary-side inventory limits are imposed. Specifically, LCO 3.7.18.a provides the maximum SG level allowed in MODE 1 or 2, by reference to TS Figure 3.7.18-1, "Maximum Allowable Steam Generator Level." Figure 3.7.18-1 identified acceptable Steam Generator Operate Range level indication (in percent) as a function of steam superheat [in degrees Fahrenheit (°F)]. Because of secondary-side dimensional and thermal performance differences between the original and replacement SGs, current TS Figure 3.7.18-1 is not appropriate for use with the replacement SGs and is to be replaced. Replacement Figure 3.7.18-1 was developed using the same

methodology used for the original, with the supporting analyses based on the dimensions and thermal performance of the replacement SGs.

TS 3.7.18.b, c, and d provide the SG inventory limitations for MODE 3, based on plant operating conditions. No changes are proposed to LCO 3.7.18.b, which establishes the maximum SG water level with Steam and Feedwater Rupture Control System (SFRCS) Instrumentation, Main Steam Line Pressure – Low (LCO 3.3.11, Function 1), not bypassed. LCO 3.7.18.c would be revised to reduce the maximum SG water level with LCO 3.3.11 Function 1 bypassed and both main feedwater pumps not capable of supplying feedwater to the SGs. LCO 3.7.18.d would be revised to reduce the maximum SG water level with LCO 3.3.11 Function 1 bypassed and one main feedwater pump capable of supplying feedwater to the SGs.

The licensee proposes to modify TS LCO 3.7.18 to ensure that appropriate inventory restrictions are imposed in order to reflect the design characteristics of the replacement SGs. The licensee intends for the modification to ensure that the existing analysis continue to remain bounding of proposed plant operation.

The steam release resulting from a rupture of a main steam pipe will result in an increase in steam flow, a reduction of coolant temperature and pressure, and an increase in core reactivity. The core reactivity increase may cause a power level increase and a decrease in shutdown margin. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review verified that proposed LCO revisions ensure that the existing analyses would remain bounding of permitted plant operation.

In its January 18, 2013, submittal, the licensee stated that TS Figure 3.7.18-1 provides a curve of SG level as a function of main steam superheat. The licensee stated that the proposed new curve, which reflects the dimensional characteristics and thermal performance differences of the new SGs, was obtained via the same methodology as the existing curve. The licensee also stated that given the proposed curve, the existing UFSAR MSLB analysis remains bounding.

To review the proposed change, the NRC staff requested additional information about the methodology used to develop the proposed, replacement curve, and verified that the licensee had correctly identified a complete set of licensing basis accident and transient analyses where the initial SG secondary inventory is a significant initial condition.

In its supplemental letter dated September 27, 2013, the licensee provided additional information about the methodology used to develop the new LCO curve. The licensee stated that the curve was developed using a computer code that solves for the steady-state conditions in once-through SGs. The code, VAGEN, predicts the steam superheat produced for a given SG mass inventory. The same computer code had been used for the previous curve; however, a newer version was used to develop the proposed curve. The licensee also stated that the newer version was validated using plant data, and that the validation indicated that the newer version more accurately simulated existing plant conditions than the previous.

It is important to note that VAGEN is being used to evaluate a set of initial conditions that are determined by the plant licensing basis safety analysis. Therefore, the analysis of record is the

MSLB analysis that is discussed in UFSAR Section 15.4.4. The VAGEN analyses merely translate the SG mass, which can't be readily monitored, into a surrogate value – the SG operate range level as a function of steam superheat – which can be readily monitored. In a supplemental letter dated January 10, 2014, the licensee clarified that the mass is determined by running a one-dimensional computer code. The computer code divides the tube bundle region into a series of control volumes where the local heat transfer coefficients, pressure drop, and two-phase flow behavior are determined from correlations, and the velocity, temperature, density, steam quality, and pressure drop are provided as output. The average liquid density in each node is multiplied by the volume and the total mass is thus determined.

Since the licensee used the same code to determine the proposed LCO as that used to determine the previous LCO, and since the updated version of VAGEN was validated and showed better agreement with plant data, the NRC staff determined that the LCO, which prescribes SG operate range level as a function of steam superheat, continues to ensure that the maximum SG liquid inventory assumed in the MSLB analyses is bounding of the design characteristics of the replacement SG. Based on this consideration, the NRC staff concluded that the proposed LCO is acceptable. Since the existing analysis remains bounding of plant operation with the new SGs, the NRC staff determined that the applicable design criteria, relating to adequate reactivity control, RCBP integrity, and adequate core cooling, remain satisfied in accordance with the current licensing basis.

SCOPE OF LICENSEE'S ANALYSIS

The NRC staff reviewed the scope of the licensee's evaluation supporting this change. In its review, the NRC staff referred to AREVA Licensing Topical Report BAW-10193NP-A, "RELAP5/MOD2-B&W for Safety Analysis of B&W [Babcock & Wilcox]-Designed Pressurized Water Reactors" (ADAMS Accession No. ML003682985). This report contains general information about significant parameters in the various safety analyses performed for B&W nuclear power plants. In particular, Appendix A of the report describes a "Large Detail Model" that is "used for those transients in which the performance of the SG and/or secondary plant dominate the transient." Table A.1 of the report identifies 15 different events that are analyzed using the large detail model.

In addition, the licensee's TS BASES, provided as Attachment 3 to the January 18, 2013, submittal, provided information that did not exclude other events from consideration as applicable safety analyses for the LCO. The requirements contained in TS LCO 3.7.18, "Steam Generator Level," are based:

> ... *primarily* [emphasis added] on preserving the initial condition assumptions for the steam generator inventory used in the main steam line break (MSLB) accident analysis."

Based on the vendor topical report and the language in the TS BASES, the NRC staff determined that confirmation was required and that MSLB was the only event relying on the initial conditions for which LCO 3.7.18 applies. In a supplement dated December 13, 2013, the licensee provided the following additional information:

> [...] Although technically correct, inclusion of the emphasized words in these statements provides the unintended implication that other design basis accident analyses may

> assume maximum steam generator secondary side inventory as an initial condition for the respective events.
>
> The change proposed to LCO 3.7.18 ensures that the maximum steam generator inventory allowed during plant operation remains bounded by the inventory that was used as an initial condition in analyzing the main steam line break. Ensuring operation within the limits of this analysis does not alter the relationship of the main steam line break event to other analyzed events. The proposed change to LCO 3.7.18 will not have any impact on the plant response to, or the analyses of, any other events identified.

In addition to the above discussion, which was provided by the licensee, the NRC staff specifically identified three licensing basis events that had the potential to be affected by SG secondary characteristics which are: (1) the main feedwater line break (MFWLB); (2) the SGTR, and (3) the ECCS evaluation. The NRC staff was specifically concerned that each of these events would be sensitive to secondary side initial conditions, and thus possibly affected by the proposed LCO revision.

The licensee's December 13, 2013, response specifically addressed each of these events. For the MFWLB, the licensee stated that no specific value was considered for an SG secondary inventory in the licensing basis MFWLB, and the mass released during the MFWLB is from the downcomer region of the SG, from the feedwater piping, not from the tubesheet region. The licensee stated that the SGTR also does not specifically consider SG secondary inventory, because the analysis calculates the atmospheric release that is introduced, through the SG, from the primary coolant. Finally, the licensee stated that the ECCS evaluations are dominated by factors other than initial SG inventory.

Since the licensee provided additional detail indicating that the initial SG inventory was not a significant parameter considered in each of the three licensing basis analyses identified by the NRC staff, the NRC staff determined that the licensee had appropriately identified the licensing basis event affected by the proposed LCO revision. This information verified the licensee's statement that "the proposed change to LCO 3.7.18 will not have any impact on the plant response to, or the analyses of, any other events identified."

PROPOSED CHANGES TO LCO 3.7.18.c and 3.7.18.d

The TS LCOs 3.7.18.c and 3.7.18.d, provide the SG inventory limitations for MODE 3 with both main feedwater pumps not capable of supplying feedwater (3.7.18.c), and with one main feedwater pump not capable of supplying feedwater (3.7.18.d). The licensee stated in its January 18, 2013, submittal:

> The total mass and energy available for release in MODE 3 is a function of both the initial SG level and the available feedwater capacity, which is assumed in the MSLB analyses to continue to feed the SG for some length of time following the main steam line break.

The MODE 3 MSLB analyses address the various feedwater capacities and the maximum allowed SG level. The LCO restrictions in MODE 3 limit initial SG level, based on available feedwater capability, to ensure that the MODE 1 analysis remains bounding.

The licensee stated that the maximum allowable SG inventory allowed in TS LCO 3.7.18.c was being reduced to reflect secondary geometry characteristics of the replacement SGs (RSGs). Since the licensee is not proposing to modify the feedwater system, and the MODE 1 MSLB analysis remains unchanged, the NRC staff determined that it is appropriate to revise the maximum allowable level in MODE 3 to reflect the dimensional characteristics of the replacement SGs. The proposed revision is acceptable because it ensures that the plant will operate within the initial conditions assumed in the design basis MSLB analysis, consistent with 10 CFR 50.36 requirements applicable to LCOs.

The proposed revision to TS LCO 3.7.18.d is similarly acceptable; however, the licensee proposed to reduce the maximum allowable feedwater capacity from "one or both MFW pumps capable," to "one MFW pump capable," rather than reducing the initial level. Since the licensee also stated that the mass and energy available for release is a function of both the initial SG level and the available feedwater capacity, it is acceptable to reduce the allowable feedwater supply capability in order to ensure that the MSLB analytic results remain bounding. Based on the licensee's proposed reduction in feedwater input capability associated with LCO 3.7.18.d, the NRC staff determined that the revised LCO was acceptable.

Since the licensee provided information that demonstrated that: (1) the existing analysis remains bounding of plant operation with the new SGs, and (2) the MSLB analysis is the only licensing basis analysis affected by the revised LCO, the NRC staff concluded that the proposed LCO revision is acceptable.

3.2 Replacement Steam Generator

The facility currently has two B&W once-through SGs with mill-annealed Alloy 600 tubes. The NRC has approved multiple amendments related to the current SGs at DBNPS, including alternate repair methods for the SG tubes. The two RSGs were manufactured by B&W Canada and contain thermally treated Alloy 690 tubes.

As discussed on page 3 of the submittal, the licensee is proposing to remove the TS requirements associated with the current alternate repair methods, and is not proposing any alternate repair methods for the RSGs. These requirements are currently contained in LCO 3.4.17 (SG tube Integrity), including its associated Actions and SRs TS 5.5.8.c. (Provisions for SG tube repair criteria), TS 5.5.8.d (Provisions for SG inspections), TS 5.5.8.f (Provisions for SG tube repair methods), and 5.6.6 (SG tube inspection report). The alternate tube repair methods were supported by analyses that were developed for the licensee's current SGs and these analyses are not applicable to the RSGs due to the new tube material; therefore, the NRC staff finds removal of the alternate repair methods acceptable.

The licensee also proposed to delete the requirements for the special visual inspection of the internal auxiliary feed water (AFW) header in TS 5.5.8.g. The internal AFW header was abandoned-in-place after being damaged many years ago and the original SGs were modified to use an external AFW header. The NRC staff finds removal of these special visual inspection requirements acceptable since the RSGs do not have an internal AFW header.

In addition, the licensee proposed to adopt inspection requirements applicable to SGs with thermally treated Alloy 690 tubes, which is the tubing material used in the RSGs. The NRC staff finds these proposed changes acceptable since the licensee's RSGs have thermally treated Alloy 690 tubes, rather than mill-annealed Alloy 600 tubes, and the proposed changes are consistent with TSTF-510, Rev. 2, which the NRC staff has already reviewed and approved on a generic basis as discussed above. The performance-based inspection requirements in the TSs require that inspection intervals be established, to ensure that SG tube integrity is maintained until the next SG inspection.

3.3 Adoption of TSTF-510

In addition to the changes proposed to reflect the replacement of the SGs at DBNPS, Unit 1, as discussed on page 3 of the submittal, the licensee also proposed to adopt the identified changes specified in TSTF-510, Rev. 2, for TSs 3.4.17, 5.5.8, and 5.6.6. TSTF-510, Rev. 2, reflect licensees' early implementation experience with their current TSs, and are primarily editorial corrections, changes, and clarifications intended to improve internal consistency, consistency with implementing industry documents, and usability without changing the intent of the requirements. The proposed changes are an improvement to the existing SG inspection requirements and continue to provide assurance that the plant licensing basis will be maintained between SG inspections. The NRC staff approved TSTF-510, Rev. 2, for use with the consolidated line item process on October 19, 2011 (ADAMS Accession No. ML112101604). Other than the variations/deviations discussed above (to reflect the replacement of the SGs), and those discussed below, the licensee is not proposing any variations or deviations from the TS changes described in TSTF-510, Rev. 2.

The DBNPS TS utilize different numbering and titles than the STSs on which TSTF-510, Rev. 2, was based. These differences are editorial and do not affect the applicability of TSTF-510, Rev. 2, to the DBNPS TS. As a result, the NRC staff finds the numbering and editorial differences between changes approved by TSTF-510, Rev. 2, and changes proposed for DBNPS acceptable.

The licensee also proposed one additional change that went beyond the changes made by TSTF-510, Rev. 2. In TS 5.5.8.d, the licensee proposed to replace the words "tube repair criteria" with the words "tube plugging criteria." [This change was also evaluated generically as documented in a letter dated June 17, 2013 (ADAMS Accession No. ML13120A541).]
The replacement of "repair" with "plugging" in TS 5.5.8.d is an editorial change that makes the wording in the specification consistent with the approved TSTF-510, Rev. 2, and, therefore, the NRC staff finds this change acceptable.

Changes to the licensee's TS Bases were submitted for information with the licensee's submittal. The NRC staff noticed that the licensee's revised TS Bases differed from the revision to the TS Bases in TSTF-510, Rev. 2, making it unclear whether the licensee's SG program would be consistent with the fundamental premise for the requirements in TSTF-510, Rev. 2. As a result, the NRC staff asked the licensee how the condition monitoring (CM) and operational assessment (OA) tube integrity determinations would be performed at DBNPS. The licensee's September 27, 2013, supplement, confirmed that the CM and OA tube integrity determinations would remain consistent with the TSTF-510 TS, Rev. 2, Bases. The NRC staff notes, however, that the wording in the licensee's revised TS Bases (as provided in the letter dated

January 18, 2013), is not fully consistent with the information provided in their letter dated September 27, 2013 (i.e., the September 27th letter, provides more detail than the TS Bases dated January 18, 2013).

In summary, the NRC staff finds the proposed changes to the SG TS requirements acceptable, since the resultant TSs are consistent with TSTF-510, Rev. 2, and reflect the tube material, dimensional characteristics and thermal differences in the RSGs. The NRC staff's basis for concluding TSTF-510, Rev. 2, is acceptable is documented in the safety evaluation dated October 19, 2011 (ADAMS Accession No. ML112101513).

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 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent 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 (78 FR 16876, and16883; March 19, 2013). The Atomic Safety and Licensing Board, on August 12, 2013 (LBP-13-11), denied the petition for leave to intervene and that decision became final agency action on December 24, 2013. 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 NRC 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) there is reasonable assurance that 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: B. Parks, NRR
A. Johnson, NRR

Date of issuance: March 31, 2014

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

Sincerely,

/ RA /

Eva A. Brown, Senior Project Manager
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures:
1. Amendment No. 287 to NPF-3
2. Safety Evaluation

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Accession Number: Package: ML1414023A747, Amendement No.:ML14023A766

***by memo dated**

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