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

May 11, 2017

Mr. Samuel L. Belcher
Senior Vice President and Chief Nuclear Officer
FirstEnergy Nuclear Operating Company
341 White Pine Drive
Akron, OH 44320

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2; DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1; AND PERRY NUCLEAR POWER PLANT, UNIT NO. 1 – ISSUANCE OF AMENDMENTS RE: APPLICATION TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-545 REVISION 3, "TS INSERVICE TESTING PROARAM REMOVAL & CLARIFY SR USAGE RULE APPLICATION TO SECTION 5.5, TESTING." (CAC NOS. MF7771, MF7772, MF7773, AND MF7774)

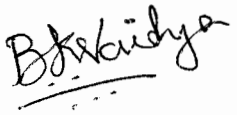
Dear Mr. Belcher:

The U.S. Nuclear Regulatory Commission (NRC or Commission) has issued the enclosed Amendment No. 298 to Facility Operating License No. DPR-66 and Amendment No. 186 to Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit Nos. 1 and 2, respectively, and Amendment No. 295 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1, and Amendment No. 175 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit No. 1. The amendments are in response to your application dated May 24, 2016, as supplemented by letter dated October 25, 2016.

The amendments eliminated the technical specification (TS), Section 5.5, "Inservice Testing Program," to remove requirements duplicated in American Society of Mechanical Engineers (ASME) Code for Operations and Maintenance of Nuclear Power Plants (OM Code), Case OMN-20, "Inservice Test Frequency." A new defined term, "INSERVICE TESTING PROGRAM," was added to TS Section 1.1, "Definitions." This change to the TS is consistent with Technical Specification Task Force-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR [surveillance requirement] Usage Rule Application to Section 5.5 Testing."

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

Sincerely,



Bhalchandra Vaidya, Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-334, 50-412, 50-346, 50-440

Enclosures:
1. Amendment No. 298 to DPR-66
2. Amendment No. 186 to NPF-73
3. Amendment No. 295 to NPF-3
4. Amendment No. 175 to NPF-58
5. Safety Evaluation

cc w/encls: Distribution via Listserv

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

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION. LLC

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 298
License No. DPR-66

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

   A. The application for amendment by FirstEnergy Nuclear Operating Company (the licensee) dated May 24, 2016, as supplemented by letter dated October 25, 2016, 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 Renewed Facility Operating License No. DPR-66 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 298, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 150 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Renewed Facility Operating License

Date of Issuance: May 11, 2017

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

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION, LLC

OHIO EDISON COMPANY

THE TOLEDO EDISON COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 186
License No. NPF-73

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

   A. The application for amendment by FirstEnergy Nuclear Operating Company (the licensee) dated May 24, 2016, as supplemented by letter dated October 25, 2016, 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 1;

   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 Renewed Facility Operating License No. NPF-73 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 186 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 150 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Renewed Facility Operating License

Date of Issuance: May 11, 2017

ATTACHMENT TO LICENSE AMENDMENT NOS. 298 AND 186

BEAVER POWER STATION, UNITS 1 AND 2

RENEWED FACILITY OPERATING LICENSE NOS. DPR-66 AND NPF-73

DOCKET NOS. 50-334 AND 50-412

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

Remove	Insert
License DPR-66	License DPR-66
Page 3	Page 3
License NPF-73	License NPF-73
Page 4	Page 4
TSs	TSs
1.1-3	1.1-3
3.4.10-2	3.4.10-2
3.5.2-3	3.5.2-3
3.6.3-5	3.6.3-5
3.6.6-1	3.6.6-1
3.6.6-2	3.6.6-2
3.6.7-2	3.6.7-2
3.7.1-2	3.7.1-2
3.7.2-2	3.7.2-2
3.7.3-2	3.7.3-2
3.7.5-4	3.7.5-4
5.5-3	5.5-3
5.5-4	5.5-4
5.5-5	5.5-5
5.5-6	5.5-6

(3) FENOC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) FENOC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) FENOC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter 1: 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 a steady state reactor core power level of 2900 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 298, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Auxiliary River Water System

(Deleted by Amendment No. 8)

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the term or conditions of any lease agreements executed as part of these transactions; (ii) the BVPS Operating Agreement, (iii) the existing property insurance coverage for BVPS Unit 2, and (iv) any action by a lessor or others that may have adverse effect on the safe operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations set forth in 10 CFR Chapter 1 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 a steady state reactor core power level of 2900 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 186 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

1.1 Definitions

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).
LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank,

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE, or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE, and

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.7	Verify, by visual inspection, that accessible regions of the ECCS containment sump suction inlet are not restricted by debris and that the accessible regions of the strainers show no evidence of structural distress or abnormal corrosion.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	Verify the isolation time of each automatic power operated containment isolation valve that is not locked, sealed, or otherwise secured in position, and required to be closed during accident conditions, is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.3.5	Verify each automatic power operated containment isolation valve that is not locked, sealed or otherwise secured in position, and required to be closed during accident conditions, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.6 Quench Spray (QS) System

LCO 3.6.6 Two QS trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One QS train inoperable.	A.1 Restore QS train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each QS manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.2	Verify each QS pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.6.3	Verify each QS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.4	Verify each QS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.5	Verify each spray nozzle is unobstructed.	Following maintenance that results in the potential for nozzle blockage

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 84 hours
E. Three or more RS subsystems inoperable.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Verify each RS manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.6.7.2	Verify each RS pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.7.3	Verify on an actual or simulated actuation signal(s): a. Each RS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position, and b. Each RS pump starts automatically.	In accordance with the Surveillance Frequency Control Program
SR 3.6.7.4	Verify each spray nozzle is unobstructed.	Following maintenance that results in the potential for nozzle blockage

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	B.2 ---------- **- NOTE -** Only required in MODE 1. ---------- Reduce the Power Range Neutron Flux - High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	36 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	AND	
OR	C.2 Be in MODE 4.	12 hours
One or more steam generators with ≥ 4 MSSVs inoperable.		

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	---------- **- NOTE -** Only required to be performed in MODES 1 and 2. ---------- Verify each required MSSV lift setpoint per Table 3.7.1-2a (Unit 1), Table 3.7.1-2b (Unit 2) in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift setting shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	---------- **- NOTE -** Only required to be performed in MODES 1 and 2. ---------- Verify the isolation time of each MSIV is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.2.2	---------- **- NOTE -** Only required to be performed in MODES 1 and 2. ---------- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the isolation time of each MFIV, MFRV, and MFRV bypass valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.3.2	Verify each MFIV, MFRV, and MFRV bypass valve actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required AFW train inoperable in MODE 4. OR Required feedwater injection header inoperable in MODE 4.	F.1 Initiate action to restore AFW train to OPERABLE status with a capability of providing flow to the steam generator(s).	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.5.1	---------- **- NOTE -** AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. ---------- Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	---------- **- NOTE -** Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 600 psig in the steam generator. ---------- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM

5.5 Programs and Manuals

5.5.2 Radioactive Effluent Controls Program (continued)

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I,

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I, and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.3 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Table 4.1-10 (Unit 1) and UFSAR Table 3.9N-1 (Unit 2), cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.4 Deleted

5.5 Programs and Manuals

5.5.5 Steam Generator (SG) Program

A Steam Generator Program for Unit 1 and Unit 2 shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program for Unit 1 shall include the provisions of Specification 5.5.5.1 and the Steam Generator Program for Unit 2 shall include the provisions of Specification 5.5.5.2.

5.5.5.1 Unit 1 SG Program

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. Provisions for 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.

5.5 Programs and Manuals

5.5.5.1 Unit 1 SG Program (continued)

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 also not to exceed 1 gpm per SG, except during a SG tube rupture.

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.

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

5.5 Programs and Manuals

5.5.5.1 Unit 1 SG Program (continued)

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 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

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

FIRSTENERGY NUCLEAR OPERATING COMPANY

AND

FIRSTENERGY NUCLEAR GENERATION, LLC

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 295
Renewed License No. NPF-3

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

   A. The application for amendment by FirstEnergy Nuclear Operating Company (the licensee) dated May 24, 2016, as supplemented by letter dated October 25, 2016, 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 Renewed 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. 295, are hereby incorporated in the renewed license. FENOC shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 150 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Renewed Facility Operating License

Date of Issuance: May 11, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 295

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

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

Remove	Insert
License NPF-3	License NPF-3
Page L-5	Page L-5
TSs	TSs
1.1-2	1.1-2
3.4.10-1	3.4.10-1
3.4.12-2	3.4.12-2
3.5.2-2	3.5.2-2
3.6.3-6	3.6.3-6
3.6.6-3	3.6.6-3
3.7.1-2	3.7.1-2
3.7.2-2	3.7.2-2
3.7.3-2	3.7.3-2
5.5-4	5.5-4

2.C. This renewed 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 renewed license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 295, are hereby incorporated in the renewed 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 renewed license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the renewed 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

1.1 Definitions

CHANNEL CHECK (continued)

the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total steps.

CONTROL RODS CONTROL RODS shall be all full length safety and regulating rods that are used to shut down the reactor and control power level during maneuvering operations.

CORE OPERATING LIMITS REPORT (COLR) The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.3. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or those listed in ICRP 30, Supplement to Part 1, page 192-212, table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity".

Ē - AVERAGE DISINTEGRATION ENERGY Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

INSERVICE TESTING PROGRAM The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE with lift settings ≤ 2525 psig.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time of Condition A not met. OR Two pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and Associated Completion Time not met.	D.1	Disable capability of both high pressure injection pumps to inject water into the RCS.	1 hour
	AND		
	D.2	Disable makeup pump suction automatic transfer to the borated water storage tank on low makeup tank level.	8 hours
	AND		
	D.3	Verify makeup tank level ≤ 73 inches.	8 hours
	AND		
	D.4	Verify RCS pressure and pressurizer level in Acceptable Region of Figure 3.4.12-1 or 3.4.12-2, as applicable.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify RCS to DHR isolation valves open with control power removed.	24 hours
SR 3.4.12.2	Verify DHR System relief valve lift setpoint ≤ 330 psig in accordance with the INSERVICE TESTING PROGRAM.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.2	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.2.3	Verify ECCS piping is full of water by venting the ECCS pump casings and discharge piping high points.	24 months AND Prior to declaring ECCS OPERABLE after draining ECCS piping
SR 3.5.2.4	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.5.2.5	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.5.2.6	Verify the correct position of each mechanical stop for the following valves: a. DH-14A; and b. DH-14B.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.3	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---------- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.4	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.3	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.6.4	Verify each required containment air cooling train starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.6.5	Verify each required containment air cooling train cooling water flow rate is ≥ 1150 gpm.	24 months
SR 3.6.6.6	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.6.6.7	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.6.6.8	Verify each spray nozzle is unobstructed.	Following maintenance that could result in nozzle blockage.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR One or more steam generators with less than two MSSVs OPERABLE. OR One or more steam generators with no MSSVs with a lift setting of 1050 psig ± 3% OPERABLE.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	----------NOTE---------- Only required to be performed in MODES 1 and 2. ---------- Verify each MSSV lift setpoint per Table 3.7.1-1 in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Verify isolation time of each MSIV is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.2.2	Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	24 months

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two valves in the same flow path inoperable.	D.1 Isolate affected flow path.	8 hours
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the isolation time of each MFSV is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.3.2	Verify the isolation time of each MFCV and SFCV is within limits.	24 months
SR 3.7.3.3	Verify each MFSV, MFCV, and SFCV actuates to the isolation position on an actual or simulated actuation signal.	24 months

5.5 Programs and Manuals

5.5.5 Allowable Operating Transient Cycles Program

This program provides controls to track the UFSAR, Section 5, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel. Inservice inspection of each reactor coolant pump flywheel shall be performed every 10 years. The inservice inspection shall be either an ultrasonic examination of the volume from the inner bore of the flywheel to the circle of one-half the outer radius, or a surface examination of exposed surfaces of the disassembled flywheel. The recommendations delineated in Regulatory Positions C.4.b(3), (4), and (5) of Regulatory Guide 1.14, Revision 1, August 1975, shall apply.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program Surveillance Frequency.

5.5.7 Deleted

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

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION, LLC

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 175
License No. NPF-58

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

   A. The application for amendment by FirstEnergy Nuclear Operating Company (the licensee) dated May 24, 2016, as supplemented by letter dated October 25, 2016, 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 Renewed Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 175, are hereby incorporated into the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 150 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: May 11, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 175

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

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

Remove	Insert
License NPF-58	License NPF-58
Page 4	Page 4
TSs	TSs
1.0-3	1.0-3
3.1-22	3.1-22
3.4-10	3.4-10
3.4-15	3.4-15
3.5-4	3.5-4
3.5-9	3.5-9
3.6-17	3.6-17
3.6-25	3.6-25
3.6-28	3.6-28
3.6-41	3.6-41
3.6-67	3.6-67
5.0-10	5.0-10
5.0-15a	5.0-15a

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I 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 reactor core power levels not in excess of 3758 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 175, are hereby incorporated into the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

a. FirstEnergy Nuclear Generation, LLC

1.1 Definitions (continued)

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. Exceptions are stated in the individual surveillance requirements.
END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC – RPT SYSTEM RESPONSE TIME shall be that time interval from initial movement of the associated turbine stop valve or the turbine control valve to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. Exceptions are stated in the individual surveillance requirements.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position, or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 32.4 gpm at a discharge pressure ≥ 1220 psig.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction is unblocked.	In accordance with the Surveillance Frequency Control Program AND Once within 24 hours after pump suction piping temperature is restored to ≥ 70°F

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 Safety/Relief Valves (S/RVs)

LCO 3.4.4 The safety function of seven S/RVs shall be OPERABLE,

AND

The relief function of six additional S/RVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required S/RVs inoperable.	A.1 Be in MODE 3.	12 hours
	AND	
	A.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify the safety function lift setpoints of the required S/RVs are as follows: Number of S/RVs — Setpoint (psig) 8 — 1165 ± 34.9 6 — 1180 ± 35.4 5 — 1190 ± 35.7	In accordance with the INSERVICE TESTING PROGRAM

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify equivalent leakage of each RCS PIV is ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at an RCS pressure ≥ 1040 psig and ≤ 1060 psig.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.2	----------NOTE---------- Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than the residual heat removal cut in permissive pressure in MODE 3, if capable of being manually realigned and not otherwise inoperable. ---------- Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify ADS accumulator supply pressure is ≥ 150 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.4	Verify each ECCS pump develops the specified flow rate with sufficient pump total head to overcome the total system resistance which includes the specified reactor-to-containment wetwell differential pressure. SYSTEM / FLOW RATE / REACTOR-TO-CONTAINMENT WETWELL DIFFERENTIAL PRESSURE LPCS / ≥ 6110 gpm / ≥ 128 psid LPCI / ≥ 7100 gpm / ≥ 24 psid HPCS / ≥ 6110 gpm / ≥ 200 psid	In accordance with the INSERVICE TESTING PROGRAM

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.5	Verify each required ECCS pump develops the specified flow rate with sufficient pump total head to overcome the total system resistance which includes the specified reactor to containment wetwell differential pressure. SYSTEM — FLOW RATE — REACTOR TO CONTAINMENT WETWELL DIFFERENTIAL PRESSURE LPCS — ≥ 6110 gpm — ≥ 128 psid LPCI — ≥ 7100 gpm — ≥ 24 psid HPCS — ≥ 6110 gpm — ≥ 200 psid	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.2.6	----------NOTE---------- Vessel injection/spray may be excluded. ---------- Verify each required ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.5	Verify the isolation time of each power operated and each automatic PCIV, except MSIVs, is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.1.3.6	----------NOTE---------- Only required to be met in MODES 1, 2, and 3. ---------- Perform leakage rate testing for each primary containment purge valve with resilient seals.	In accordance with the Surveillance Frequency Control Program AND Once within 92 days after opening the valve
SR 3.6.1.3.7	Verify the isolation time of each MSIV is ≥ 2.5 seconds, and ≤ 5 seconds.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.1.3.8	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.7.1	----------NOTE---------- RHR containment spray subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below the RHR cut in permissive pressure in MODE 3 if capable of being manually realigned and not otherwise inoperable. ---------- Verify each RHR containment spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.7.2	Verify each RHR pump develops a flow rate of ≥ 5250 gpm on recirculation flow through the associated heat exchangers to the suppression pool.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.1.7.3	Verify each RHR containment spray subsystem automatic valve in the flow path actuates to its correct position on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.7.4	Verify each spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.9.1	Verify the isolation time of each valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.3.1	Verify each RHR suppression pool cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct positon or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.6.2.3.2	Verify each RHR pump develops a flow rate ≥ 7100 gpm through the associated heat exchangers to the suppression pool.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.3.1	Verify each 24 inch and 36 inch drywell purge supply and exhaust isolation valve is sealed closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.5.3.2	Deleted.	
SR 3.6.5.3.3	----------NOTES---------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for drywell isolation valves that are open under administrative controls. ---------- Verify each drywell isolation manual valve and blind flange that is not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed.	Prior to entering MODE 2 or 3 from MODE 4, if not performed in the previous 92 days
SR 3.6.5.3.4	Verify the isolation time of each power operated and each automatic drywell isolation valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.5.3.5	Verify each automatic drywell isolation valve actuates to the isolation position on an actual or simulated isolation signal.	In accordance with the Surveillance Frequency Control Program

5.5 Programs and Manuals (continued)

5.5.6 Deleted.

5.5.7 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2.

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

- BN-TOP-1 methodology may be used for Type A tests.

- The corrections to NEI 94-01 which are identified on the Errata Sheet attached to the NEI letter, "Appendix J Workshop Questions and Answers," dated March 19, 1996 are considered an integral part of NEI 94-01.

- The containment isolation check valves in the Feedwater penetrations are tested per the INSERVICE TESTING PROGRAM.

- The provisions of NEI 94-01, Section 9.2.3 are revised to include the following exception: The first Type A test performed after the Type A test completed on July 1, 1994 shall be completed no later than June 29, 2009.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident is 6.40 psig. For conservatism P_a is defined as 7.80 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.20% of primary containment air weight per day at the peak containment pressure (P_a).

Leakage rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. However, during the first unit startup following testing performed in accordance with this Program, the leakage rate acceptance criteria are $< 0.6\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for the Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is ≤ 2.5 scfh when tested at $\geq P_a$.

   2) For each door, leakage rate is ≤ 2.5 scfh when the gap between the door seals is pressurized to $\geq P_a$.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

Nothing in these Technical Specifications shall be construed to modify the testing frequencies required by 10 CFR 50, Appendix J.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR PLANT-SPECIFIC ADOPTION OF

TECHNICAL SPECIFICATIONS TASK FORCE TRAVELER TSTF-545, REVISION 3,

"TS INSERVICE TESTING PROGRAM REMOVAL & CLARIFY SR USAGE RULE

APPLICATION TO SECTION 5.5 TESTING,"

USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS

RELATED TO

AMENDMENT NO. 298 TO BEAVER VALLEY POWER STATION, UNIT NO. 1

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-66,

DOCKET Mo. 50-334,

AMENDMENT NO. 186 TO BEAVER VALLEY POWER STATION, UNIT NO. 2

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-73,

DOCKET NO. 412,

AMENDMENT NO. 295 TO DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-3,

DOCKET NO. 50-346

AND

AMENDMENT NO. 175 PERRY NUCLEAR POWER PLANT, UNIT NO. 1

TO FACILITY OPERATING LICENSE NO. NPF-58,

DOCKET NO. 50-440

FIRSTENERGY NUCLEAR OPERATING COMPANY

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC or Commission) dated May 24, 2016, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16148A047), as supplemented by letter dated October 25, 2016 (ADAMS Accession No. ML16299A113), FirstEnergy Nuclear Operating Company (FENOC, the licensee) requested changes to the technical specifications (TSs) for FENOC facilities at Beaver Valley Power Station, Units 1 and 2 (BVPS 1 and 2), Davis-Besse Nuclear Power Station, Unit 1 (DBNPS), and Perry Nuclear Power Plant, Unit 1 (PNPP).

Specifically, the licensee requested changes to the TSs consistent with Technical Specifications Task Force (TSTF) Standard Technical Specifications (STS) Change Traveler TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR [surveillance requirement] Usage Rule Application to Section 5.5 Testing," dated October 21, 2015 (ADAMS Accession No. ML15294A555).

Specifically, the licensee's proposed changes would revise the TS to eliminate TS 5.5.4 for BVPS 1 and 2, TS 5.5.7 for DBNPS, Unit 1, and TS 5.5.6 for PNPP, Unit 1, "Inservice Testing Program" and would add a new defined term, "INSERVICE TESTING PROGRAM," to the TSs. All existing references to the "Inservice Testing Program" in the plant-specific TS SRs are replaced with "INSERVICE TESTING PROGRAM" so that the SRs refer to the new definition in lieu of the deleted program.

The licensee's letter dated May 24, 2016, also included a request to use American Society of Mechanical Engineers (ASME) Code Case OMN-20, "Inservice Test Frequency," as an alternative to certain ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) requirements at BVPS 1 and 2, DBNPS, and PNPP. The U.S. Nuclear Regulatory Commission (NRC or Commission) considered this request separately from the proposed license amendment, and authorized the licensee's use of this alternative by letter dated January 23, 2017 (ADAMS Accession No. ML16328A125).

The licensee stated that the license amendment request is consistent with NRC-approved Traveler TSTF-545, Revision 3. However, Section 3.3.4 of this safety evaluation (SE) provides the NRC staff evaluation of the licensee's proposed variations to the TS changes from those described in TSTF-545, Revision 3. This TS improvement was made available via letter to the TSTF dated December 11, 2015 (ADAMS Package Accession No. ML15317A071), as part of the consolidated line item improvement process.

The supplement dated October 25, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 2, 2016, (81 FR 50732).

2.0 REGULATORY EVALUATION

2.1 Description of Inservice Testing Requirements and TSTF-545

An inservice test is a test to assess the operational readiness of a system, structure, or component after first electrical generation by nuclear heat. The ASME OM Code provides requirements for inservice testing of certain components in light-water nuclear power plants. The ASME OM Code identifies the components subject to the testing (i.e., pumps, valves,

pressure relief devices, and dynamic restraints), responsibilities, methods, intervals, parameters to be measured and evaluated, criteria for evaluating results, corrective actions, personnel qualification, and record keeping. Title 10 of the *Code of Federal Regulations* (10 CFR), Paragraph 50.55a(f), "Inservice testing requirements," requires that inservice tests of certain ASME Code Class 1, 2, and 3 components must meet the requirements of the ASME OM Code and applicable addenda. The TSs also prescribe inservice test requirements and frequencies for ASME Code Class 1, 2, and 3 components.

The regulation in 10 CFR 50.55a(f)(5)(ii) states that if a revised inservice test program for a facility conflicts with the TSs, the licensee must apply to the Commission for amendment of the TSs to conform the TSs to the revised program. TSTF-545, Revision 3, provides guidance to licensees on how to request license amendments that would eliminate conflicting requirements between 10 CFR 50.55a and the TSs. TSTF-545, Revision 3, proposes elimination of the Inservice Testing Program from the "Administrative Controls" section of the TSs. The TSs contain surveillances that require testing or test intervals in accordance with the inservice test. The elimination of the Inservice Testing Program from the TSs could cause uncertainty regarding the correct application of these SRs. Therefore, TSTF-545, Revision 3, also proposes adding a new definition, "INSERVICE TESTING PROGRAM," to the TSs which would be defined as the licensee program that fulfills the requirements of 10 CFR 50.55a(f). TSTF-545, Revision 3, proposes replacement of existing uses of the term, "Inservice Testing Program," with the defined term, as denoted by capitalized letters, throughout the TSs.

By letter dated December 11, 2015,[1] the NRC found the changes to the STSs proposed in TSTF-545, Revision 3, to be suitable for incorporation into the STSs and announced that licensees could request amending their licenses to adopt TSTF-545, Revision 3. The NRC published a notice of availability of TSTF-545, Revision 3, in the *Federal Register* (FR) on March 28, 2016 (81 FR 17208).

2.2 Proposed Technical Specifications Changes

The licensee requested to revise the plants' TSs by deleting the the Inservice Testing Program TSs from the Administrative Controls TS sections for BVPS 1 and 2 (TS 5.5.4), DBNPS (TS 5.5.7), and PNPP (TS 5.5.6), as follows, with proposed deletions shown as stricken text and additions as bolded text).

BVPS 1 and 2:

Deleted~~Inservice Testing Program~~

~~This program provides controls for inservise testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:~~

~~a. Testing frequensies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda are as follows:~~

~~ASME OM Code and applicable Addenda terminology for inservice testing activities~~	~~Required Frequencies for performing inservice testing activities~~

[1] ADAMS Package Accession No. ML15317A071.

~~Weekly~~	~~At least once per 7 days~~
~~Monthly~~	~~At least once per 31 days~~
~~Quarterly or every 3 months~~	~~At least once per 92 days~~
~~Semiannually or every 6 months~~	~~At least once per 184 days~~
~~Every 9 months~~	~~At least once per 276 days~~
~~Yearly or annually~~	~~At least once per 366 days~~
~~Biennially or every 2 years~~	~~At least once per 731 days~~

~~b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities,~~

~~c. The provisions of SR 3.0.3 are applicable to inservise testing activities, and~~

~~d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.~~

DBNPS:

Deleted~~Inservice Testing Program~~

~~This program provides controls for inservise testing of ASME Code Class 1, 2, and 3 pumps and valves.~~

~~a. Testing frequensies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Codes) and applicable Addenda are as follows:~~

~~ASME OM Code and applicable Addenda terminology for inservice testing activities~~	~~Required Frequencies for performing inservice testing activities~~
~~Monthly~~	~~At least once per 31 days~~
~~Quarterly or every 3 months~~	~~At least once per 92 days~~
~~Yearly or annually~~	~~At least once per 366 days~~
~~Biennially or every 2 years~~	~~At least once per 731 days~~

~~b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;~~

~~c. The provisions of SR 3.0.3 are applicable to inservise testing activities; and~~

~~d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.~~

PNPP:

Deleted~~Inservice Testing Program~~

~~This program provides controls for inservise testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:~~

~~a. Testing frequensies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:~~

~~ASME OM Code and applicable Addenda terminology for inservice testing activities~~	~~Required frequencies for performing inservice testing activities~~
~~Weekly~~	~~At least once per 7 days~~
~~Monthly~~	~~At least once per 31 days~~
~~Quarterly or every 3 months~~	~~At least once per 92 days~~
~~Semiannually or every 6 months~~	~~At least once per 184 days~~
~~Every 9 months~~	~~At least once per 276 days~~
~~Yearly or annually~~	~~At least once per 366 days~~
~~Biennially or every 2 years~~	~~At least once per 731 days~~

~~b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;~~

~~c. The provisions of SR 3.0.3 are applicable to inservise testing activities; and~~

~~d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.~~

BVPS 1 and 2 TS 5.5.4.b, DBNPS TS 5.5.7.b, and PNPP TS 5.5.6.b, which refer to SR 3.0.2, allow an extension of inservice testing intervals by up to 25 percent. If it is discovered that a surveillance associated with an inservice testing activity was not performed within the required interval, TS SR 3.0.3 for BVPS 1 and 2, DBNPS, and PNPP, allow the licensee to delay declaring the associated limiting condition for operation (LCO) not met in order to perform the missed surveillance. The licensee did not request changes to BVPS 1 and 2, DBNPS, and PNPP, TS SR 3.0.2 or TS SR 3.0.3.

The licensee requested to revise BVPS 1 and 2, DBNPS, and PNPP, Definitions section of the TSs by adding the term, "INSERVICE TESTING PROGRAM," with the following definition: "The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." The licensee also requested that all existing occurrences of "Inservice Testing Program" for BVPS, DBNPS and PNPP in the TS be replaced with "INSERVICE TESTING PROGRAM," so that the term refers to the new definition in lieu of the deleted program.

2.3 Regulatory Evaluation

The NRC staff considered the following regulatory requirements, guidance, and licensing information during its review of the proposed changes.

Technical Specifications

Paragraph 50.36(c) of 10 CFR requires TSs to include the following categories: (1) safety limits, limiting safety systems settings, and control settings; (2) LCOs; (3) SRs; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports.

inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.
Paragraph 50.36(c)(5) states that administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

The NRC staff's guidance for review of the TSs is in NUREG-0800, "Standard Review Plan," Chapter 16, "Technical Specifications," Revision 3, dated March 2010.[2] As described therein, as part of the regulatory standardization effort, the staff has prepared improved STSs for each of the light-water reactor nuclear steam supply systems and associated balance-of-plant equipment systems. The licensee's proposed amendments are based on TSTF-545, Revision 3, which is an NRC-approved change to the improved STSs. The staff's review included consideration of whether the proposed changes are consistent with TSTF-545, Revision 3. The staff gives special attention to TS provisions that depart from the improved STSs, as modified by NRC-approved TSTF Travelers, to determine whether proposed differences are justified by uniqueness in plant design or other considerations so that 10 CFR 50.36 is met.

Inservice Testing

Pursuant to 10 CFR 50.54, "Conditions of licenses," the applicable requirements of 10 CFR 50.55a are conditions of every nuclear power reactor operating license issued under 10 CFR Part 50. These requirements include inservice testing of pumps and valves at nuclear power reactors in accordance with the ASME OM Code as specified in 10 CFR 50.55a(f). Paragraph 50.55a(f) states that systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Boiler and Pressure Vessel Code and ASME OM Code as specified in the paragraph, and that each operating license for a boiling or pressurized water-cooled nuclear facility is subject to the requirements of 10 CFR 50.55a(f)(1) through (f)(6).

The ASME OM Code is a consensus standard, which is incorporated by reference into 10 CFR 50.55a. During the incorporation process, the NRC staff reviewed the ASME OM Code requirements for technical sufficiency and found that the ASME OM Code Inservice Testing program requirements were suitable for incorporation into the NRC's rules.

Paragraph 50.55(a)(f)(5)(ii) of 10 CFR states that if a revised inservice test program for a facility conflicts with the TSs for the facility, the licensee must apply for an amendment of the TSs to conform the TSs to the revised program.

NUREG-1482, Revision 2, "Guidelines for Inservice Testing at Nuclear Power Plants: Inservice Testing of Pumps and Valves and Inservice Examination and Testing of Dynamic Restraints (Snubbers) at Nuclear Power Plants," Final Report, published October 2013,[3] provides guidance for the inservice testing of pumps and valves.

NUREG-0800, Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints," Revision 3, dated March 2007,[4] provides guidance

---

[2] ADAMS Accession No. ML100351425.
[3] ADAMS Accession No. ML13295A020.
[4] ADAMS Accession No. ML070720041.

and acceptance criteria for the NRC staff's review of the inservice testing program for pumps and valves.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's application to determine if the proposed changes are consistent with the guidance, regulations, and licensing information discussed in Section 2.3 of this SE. In determining whether an amendment to a license will be issued, the Commission is guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. Among the considerations are whether the TSs, as amended, would provide the necessary administrative controls per 10 CFR 50.36(c)(5) (i.e., provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner). In making its determination as to whether to amend the license, the staff considered those regulatory requirements that are automatically conditions of the license through 10 CFR 50.54. Where the regulations already condition the license, there is no need for a duplicative requirement in the TSs; the regulations provide the necessary reasonable assurance of the health and safety of the public.

3.1 Deletion of the Inservice Testing Program from the TSs

For BVPS 1 and 2, DBNPS, and PNPP, the Inservice Testiing Program TSs are TS 5.5.4, TS 5.5.7, and TS 5.5.6, respectively, and it requires the licensee to have an inservice testing program that provides controls for inservice testing of ASME Code Class 1, 2, and 3 components (i.e., pumps and valves). Through 10 CFR 50.54, the applicable requirements of 10 CFR 50.55a are conditions of every nuclear power reactor operating license issued under 10 CFR Part 50. These requirements include 10 CFR 50.55a(f), which specifies the requirements for the inservice test of pumps and valves. Therefore, requiring the licensee to have an inservice testing program in TSs is duplicative of the license condition in 10 CFR 50.54. Thus, with the proposed TS changes, the licensee will still be required to maintain an inservice testing program in accordance with the ASME OM Code, as specified in 10 CFR 50.55a(f). For the reasons explained below, it is not necessary to have additional administrative controls in the TSs relating to the inservice testing program to assure operation of the facility in a safe manner.

Consideration of BVPS TS 5.5.4.a, DBNPS TS 5.5.7.a and PNPP TS 5.5.6.a

The ASME OM Code requires testing to normally be performed within certain time periods. BVPS 1 and 2 TS 5.5.4.a, DBNPS TS 5.5.7.a, and PNPP TS 5.5.6.a, set inservice test frequencies more precisely than those specified in the ASME OM Code and applicable addenda (e.g., "at least once per 31 days" contrasted with "monthly"). However, the NRC staff determined that the more precise inservice test frequencies are not necessary to assure operation of the facility in a safe manner. Therefore, the NRC staff determined that deletion of BVPS 1 and 2 TS 5.5.4.a, DBNPS TS 5.5.7.a, and PNPP TS 5.5.6.a, is acceptable.

Consideration of BVPS TS 5.5.4.b, DBNPS TS 5.5.7.b, and PNPP TS 5.5.6.b

BVPS 1 and 2 TS 5.5.4.b, DBNPS TS 5.5.7.b, and PNPP TS 5.5.6.b allow the licensee to extend, by up to 25 percent, the interval between inservice test activities, as required by BVPS 1 and 2 TS 5.5.4.a, DBNPS TS 5.5.7.a, and PNPP TS 5.5.6.a, and for other normal and accelerated frequencies specified as 2 years or less in the inservice testing program. Similar to BVPS 1 and 2, TS 5.5.4.b, DBNPS TS 5.5.7.b, and PNPP TS 5.5.6.b, the NRC authorization of ASME Code Case OMN-20, "Inservice Test Frequency," by letter dated January 23, 2017

(ADAMS Accession No. ML16328A125), also permits the licensee to extend the inservice testing intervals specified in the ASME OM Code by up to 25 percent.

The NRC staff determined that the BVPS 1 and 2 TS 5.5.4.b, DBNPS TS 5.5.7.b, and PNPP TS 5.5.6.b allowance to extend IST intervals is not needed to assure operation of the facility in a safe manner. Therefore, the NRC staff determined that deletion of BVPS 1 and 2 TS 5.5.4.b, DBNPS TS 5.5.7.b, and PNPP TS 5.5.6.b is acceptable. This deletion does not impact the licensee's ability to extend IST intervals using Code Case OMN-20, as authorized by the NRC.

Consideration of BVPS TS 5.5.4.c, DBNPS TS 5.5.7.c, and PNPP TS 5.5.6.c

BVPS 1 and 2 TS 5.5.4.c, DBNPS TS 5.5.7.c, and PNPP TS 5.5.6.c allow the licensee to use SR 3.0.3 when it discovers that an SR associated with an inservice test was not performed within its specified frequency. SR 3.0.3 allows the licensee to delay declaring a LCO not met in order to perform the missed surveillance. The use of SR 3.0.3 for inservice tests is limited to those required by an SR. In accordance with 10 CFR 50.55a, the licensee may also request relief from the ASME OM Code requirements to address issues associated with a missed inservice test. Deletion of BVPS TS 5.5.4.c, DBNPS TS 5.5.7.c, and PNPP TS 5.5.6.c does not change any of these requirements, and SR 3.0.3 will continue to apply to those inservice tests required by SRs. Based on the above, the NRC staff determined that this deletion is acceptable.

Consideration of BVPS TS 5.5.4.d, DBNPS TS 5.5.7.d, and PNPP TS 5.5.6.d

BVPS 1 and 2 TS 5.5.4.d, DBNPS TS 5.5.7.d, and PNPP TS 5.5.6.d state that nothing in the ASME OM Code shall be construed to supersede the requirements of any TS. However, the regulations in 10 CFR 50.55a(f)(5)(ii) address what to do if a revised inservice testing program for a facility conflicts with the TSs for the facility; they require the licensee to apply for an amendment to the TSs to conform the TSs to the revised program at least 6 months prior to the start of the period for which the provisions become applicable. Accordingly, there is no need for a TS stating how to address conflicts between the TSs and the inservice teting program because the regulations specify how conflicts must be resolved.

Conclusion Regarding Deletion of BVPS 1 and 2 TS 5.5.4, DBNPS TS 5.5.7, and PNPP TS 5.5.6

The NRC staff determined that the requirements currently in BVPS 1 and 2 TS 5.5.4, DBNPS TS 5.5.7, and PNPP TS 5.5.6 are not necessary to assure operation of the facility in a safe manner. Based on this evaluation, the staff concludes that deletion of BVPS 1 and 2 TS 5.5.4, DBNPS TS 5.5.7, and PNPP TS 5.5.6+ from the licensee's TSs is acceptable because it is not required by 10 CFR 50.36(c)(5).

3.2 Definition of INSERVICE TESTING PROGRAM and Revision of SRs

The licensee proposed to revise BVPS 1 and 2, DBNPS, and PNPP TS "Definitions" section to include the term, "INSERVICE TESTING PROGRAM," with the following definition: "The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." The proposed definition is consistent with the definition in TSTF-545, Revision 3. The NRC staff finds the definition acceptable because it correctly refers to the IST requirements in 10 CFR 50.55a(f).

The licensee requested that all existing references to the "Inservice Testing Program" in BVPS 1 and 2, DBNPS, and PNPP TS be revised to "INSERVICE TESTING PROGRAM" to reference the new TS defined term in lieu of the deleted inservice testing program TSs. The proposed change is consistent with the intent of TSTF-545, Revision 3, to replace the current references in TS with the new definition. The NRC staff verified that for each TS reference to the "Inservice Testing Program," the licensee proposed to change the reference to "INSERVICE TESTING PROGRAM." The proposed change does not alter how the inservice testing is performed. However, the inservice test frequencies could change because the TSs will no longer include the more precise test frequencies in TS 5.5.4.a for BVPS 1 and 2, TS 5.5.7.a for DBNPS, and TS 5.5.6.a for PNPP. As discussed in Section 3.1 of this SE, the staff determined that the TSs do not need to include the more precise testing frequencies currently in the TS 5.5.4.a for BVPS 1 and 2, TS 5.5.7.a for DBNPS, and TS 5.5.6.a for PNPP. Based on its review, the staff determined that revising TS to refer to the new definition is acceptable because any testing will continue to be performed in accordance with the requirements of 10 CFR 50.55a(f). The staff also determined that, with the proposed changes that allow less-precise testing frequencies, 10 CFR 50.36(c)(3) will continue to be met because testing will continue to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

3.3 Evaluations of Requested Variations from the Notice of Availability

In its application to adopt TSTF-545, Revision 3, dated May 24, 2016, the licensee proposed the several variations to TS changes for each facility. The following sections discuss the NRC staff evaluation of the variations for each facility.

3.3.1 Beaver Valley Power Station, Units 1 and 2

The BVPS TSs utilize different numbering and do not include all the specifications shown on the applicable Standard Technical Specifications, Westinghouse Plants - Specifications, NUREG-1431, Revision 4.0, (Westinghouse STS) pages in TSTF-545. The licensee proposed the following variations for BVPS 1 and 2:

1. The specified surveillance for Westinghouse STS SR 3.4.10.1 in TSTF-545 does not show a change to make "Inservice Testing Program" all capital letters. The specified surveillance for BVPS TS SR 3.4.10.1 will be changed to make "Inservice Testing Program" all capital letters.

2. There will be no change to the specified frequency in BVPS TS SR 3.4.14.1 since the specified frequency does not include a reference to the Inservice Testing Program as is shown in the specified frequency for SR 3.4.14.1 of the Westinghouse STS in TSTF-545.

3. Westinghouse STS SR 3.6.3.5 is numbered SR 3.6.3.4 in the BVPS TS. Therefore, the change to the specified frequency for SR 3.6.3.5 shown in TSTF-545 will be incorporated in the specified frequency for BVPS TS SR 3.6.3.4.

4. Since the BVPS TSs do not include SR 3.6.6A.4, SR 3.6.6B.4, SR 3.6.6C.2, or SR 3.6.12.1, as shown on the Westinghouse STS pages in TSTF-545, there will be no corresponding changes to BVPS TSs.

5. Westinghouse STS SR 3.6.6D.2 is numbered SR 3.6.6.2 in the BVPS TS. Therefore, the change to the specified frequency for SR 3.6.6D.2 shown in TSTF-545 will be incorporated in the specified frequency for BVPS TS SR 3.6.6.2.

6. Westinghouse STS SR 3.6.6E.5 is numbered SR 3.6.7.2 in the BVPS TS. Therefore, the change to the specified frequency for SR 3.6.6E.5 shown in TSTF-545 will be incorporated in the specified frequency for BVPS TS SR 3.6.7.2.

7. Westinghouse STS pages in TSTF-545 propose elimination of TS 5.5.8, "Inservice Testing Program," however, the Inservice Testing Program specification is numbered 5.5.4 in the BVPS TS. BVPS TS 5.5.4 will be revised to show that the Inservice Testing Program has been deleted and the subsequent TSs 5.5.5 through 5.5.15 will not be renumbered. BVPS TS do not include TS numbers 5.5.16 through 5.5.20 as shown in Westinghouse STS pages in TSTF-545.

8. Due to not renumbering specifications after BVPS TSs 5.5.4, changes related to section renumbering were not incorporated.

The NRC staff evaluated the following conforming changes and variations from TSTF-545, Revision 3, not previously addressed in this SE.

a. BVPS TS use different numbering than the improved STSs, on which TSTF-545, Revision 3, is based. The NRC staff finds the licensee's proposed deviations in numbering, format, and content editorial in nature and that the licensee's proposed TS changes are consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds that the licensee's proposed TS changes are acceptable.

b. The licensee proposed a change to BVPS TS SR 3.4.10.1 by making "Inservice Testing Program" all capital letters. Westinghouse STS SR 3.4.10.1 in TSTF-545 does not show a change to make "Inservice Testing Program" all capital letters. The NRC staff finds the licensee's proposed deviations editorial in nature and that the licensee's proposed TS changes are consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds that the licensee's proposed TS changes are acceptable.

c. The licensee is not proposing a change to the specified frequency in BVPS TS SR 3.4.14.1 since the specified frequency does not include a reference to the Inservice Testing Program as is shown in the specified frequency for SR 3.4.14.1 of the Westinghouse STS in TSTF-545. The NRC staff finds the licensee's proposed deviations editorial in nature and that the licensee's proposed TS changes are consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds that the licensee's proposed TS changes are acceptable.

d. BVPS TS does not include all the specifications shown on the applicable STS for which there is a reference to IST. Specifically, BVPS TS does not include TS SR 3.6.6A.4, SR 3.6.6B.4, SR 3.6.6C.2, or SR 3.6.12.1. Since BVPS TS does not contain the SR included in Westinghouse STS, the licensee proposed a deviation from TSTF-545 by not including the capitalized term. The NRC staff finds the licensee's proposed deviations consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds that the licensee's proposed TSs changes are acceptable.

e. The licensee proposed to replace the content of IST Program TSs with the word, "Deleted," and retain the existing numbering sequence. The staff finds that these proposed changes are consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds that the licensee's proposed TS changes are acceptable.

3.3.2 Davis-Besse Nuclear Power Station

The DBNPS TSs utilize different numbering and do not include all the specifications shown on the applicable Standard Technical Specifications, Babcock and Wilcox Plants – Specifications, NUREG-1430, Revision 4.0 (Babcock & Wilcox STS) pages in TSTF-545. The licensee proposed the following variations for DBNPS:

1. Reference to the "Inservice Testing Program" will be changed to all capital letters in the surveillance and frequency statements specified in DBNPS SR 3.4.12.2, even though this change is not specified in TSTF-545 for Babcock & Wilcox STS. DBNPS TS SR 3.4.12.2 is not included in the Babcock & Wilcox STS.

2. Babcock & Wilcox STS SR 3.4.14.1 is numbered SR 3.4.14.2 in the DBNPS TS. Since DBNPS TS SR 3.4.14.2 does not invoke the Inservice Testing Program, no change to this SR is necessary.

3. Babcock & Wilcox STS SR 3.5.2.4 is numbered SR 3.5.2.2 in the DBNPS TS. Therefore, the change to the specified frequency for SR 3.5.2.4 shown in TSTF-545 will be incorporated in the specified frequency for DBNPS TS SR 3.5.2.2.

4. Babcock & Wilcox STS SR 3.6.3.5 is numbered SR 3.6.3.4 in the DBNPS TS. Therefore, the change to the specified frequency for SR 3.6.3.5 shown in TSTF-545 will be incorporated in the specified frequency for DBNPS TS SR 3.6.3.4.

5. Babcock & Wilcox STS SR 3.6.6.4 is numbered SR 3.6.6.3 in the DBNPS TS. Therefore, the change to the specified frequency for SR 3.6.6.4 shown in TSTF-545 will be incorporated in the specified frequency for DBNPS TS SR 3.6.6.3.

6. There will be no change to the specified frequency in DBNPS TS SR 3.7.5.2 since the specified frequency does not include a reference to the Inservice Testing Program as is shown in the specified frequency for SR 3.7.5.2 of the Babcock & Wilcox STS in TSTF-545.

7. Babcock & Wilcox STS pages in TSTF-545 propose elimination of TS 5.5.8, "Inservice Testing Program;" however, the Inservice Testing Program specification is numbered 5.5.7 in the DBNPS TS. DBNPS TS 5.5.7 will be revised to show that the Inservice Testing Program has been deleted and the subsequent TSs 5.5.8 through 5.5.17 will not be renumbered. DBNPS TS do not include TS numbers 5.5.18, 5.5.19, or 5.5.20 as shown in Babcock & Wilcox STS pages in TSTF-545.

8. Due to not renumbering specifications after DBNPS TSs 5.5.7, changes related to section renumbering were not incorporated.

The NRC staff evaluated the following conforming changes and variations from TSTF-545, Revision 3, not previously addressed in this SE.

a. DBNPS TS use different numbering than the improved STSs, on which TSTF 545, Revision 3, is based. The NRC staff finds the licensee's proposed deviations in numbering, format, and content are consistent with the intent of TSTF 545, Revision 3. Therefore, the staff finds that the licensee's proposed TS changes are acceptable.

b. The licensee proposed a change to DBNPS SR 3.4.12.2 by making "Inservice Testing Program" all capital letters. TSTF-545 does not show a change to make "Inservice Testing Program" all capital letters in Babcock & Wilcox STS. The NRC staff finds the licensee's proposed deviations are consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds that the licensee's proposed TS changes are acceptable.

c. The licensee is not proposing a change to the specified frequency in DBNPS TS SR 3.7.5.2 since the specified frequency does not include a reference to the Inservice Testing Program as is shown in the specified frequency for SR 3.7.5.2 of the Babcock & Wilcox STS in TSTF-545. The NRC staff finds the licensee's proposed deviations are consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds that the licensee's proposed TS changes are acceptable.

d. The licensee proposed to replace the content of IST Program TSs with the word, "Deleted," and retain the existing numbering sequence. The staff finds that these proposed changes are consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds that the licensee's proposed TS changes are acceptable.

3.3.3 Perry Nuclear Power Plant

The PNPP TSs utilize different numbering and do not include all the specifications shown on the applicable Standard Technical Specifications, General Electric BWR [boiling-water reactor]/6 Plants – Specifications, NUREG-1434, Revision 4.0, (General Electric BWR/6 STS) pages in TSTF-545. The licensee proposed the following variations for PNPP:

1. Reference to the "Inservice Testing Program" will be changed to all capital letters in the frequency statement specified in PNPP SR 3.6.1.9.1, even though this change is not specified in TSTF-545 for General Electric BWR/6 STS. PNPP SR 3.6.1.9.1 is not included in the General Electric BWR/6 STS.

2. Since the PNPP TSs do not include SR 3.6.4.2.2 as shown on the General Electric BWR/6 STS pages in TSTF-545, there will be no corresponding changes to PNPP TSs.

3. General Electric BWR/6 STS pages in TSTF-545 propose elimination of TS 5.5.7, "Inservice Testing Program," however, the Inservice Testing Program specification is numbered 5.5.6 in the PNPP TS. PNPP TS 5.5.6 will be revised to show that the Inservice Testing Program has been deleted and the subsequent TSs 5.5.7 through 5.5.14 will not be renumbered. PNPP TS do not include TS numbers 5.5.15, 5.5.16, or 5.5.17 as shown in General Electric BWR/6 STS pages in TSTF-545.

4. Due to not renumbering specifications after PNPP TSs 5.5.6, changes related to section renumbering were not incorporated.

5. Reference to the "Inservice Testing Program" will be changed to all capital letters in PNPP TS 5.5.12, "Primary Containment Leakage Rate Testing Program," even though this change is not specified in TSTF-545 for General Electric BWR/6 STS. In addition, a reference to

Specification 5.5.6 in Specification 5.5.12 will be deleted. PNPP TS 5.5.12 wording is different than the General Electric BWR/6 STS 5.5.13, "Primary Containment Leakage Rate Testing Program," wording.

The NRC staff evaluated the following conforming changes and variations from TSTF-545, Revision 3, not previously addressed in this SE.

a. The licensee proposed a change to PNPP SR 3.6.1.9.1 by making "Inservice Testing Program" all capital letters. TSTF-545 does not show a change to make "Inservice Testing Program" all capital letters in General Electric BWR/6 STS because PNPP SR 3.6.1.9.1 is not included in the General Electric BWR/6 STS. The NRC staff finds the licensee's proposed deviations are consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds that the licensee's proposed TS changes are acceptable.

b. The licensee is not proposing a change similar to General Electric BWR/6 STS SR 3.6.4.2.2 as shown on the pages in TSTF 545, since PNPP TS does not include an equivalent specification. The NRC staff finds the licensee's proposed changes are consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds that the licensee's proposed TS changes are acceptable.

c. The licensee proposed to capitalize all letters of reference to the "Inservice Testing Program" in PNPP TS 5.5.12, "Primary Containment Leakage Rate Testing Program." In addition, a reference to Technical Specification 5.5.6 in Technical Specification 5.5.12 will be deleted. PNPP TS 5.5.12 wording is different than STS wording. The NRC staff finds the licensee's proposed deviations are consistent with the intent of TSTF 545, Revision 3. Therefore, the staff finds that the licensee's proposed TSs changes are acceptable.

d. The licensee proposed to replace the content of IST Program TSs with the word, "Deleted," and retain the existing numbering sequence. The staff finds that these proposed changes are consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds that the licensee's proposed TS changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania and Ohio State officials were notified of the proposed issuance of the amendments on March 15, 2017. The State officials had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC 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 published in the *Federal Register* (FR) on August 2, 2016, (81 FR 50732). 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 Commission 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Bhalchandra Vaidya, NRR
Blake Purnell, NRR
Caroline Tilton, NRR
John Huang, NRR

Date of Issuance: May 11, 2017

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2; DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1; AND PERRY NUCLEAR POWER PLANT, UNIT NO. 1 – ISSUANCE OF AMENDMENTS RE: APPLICATION TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-545.REVISION 3, "TS INSERVICE TESTING PROARAM REMOVAL & CLARIFY SR USAGE RULE APPLICATION TO SECTION 5.5, TESTING." (TAC NOS. MF7771, MF7772, MF7773, AND MF7774) DATED MAY 11, 2017

DISTRIBUTION:

Public	LPL3 R/F	LPL1 R/F
RidsNrrDorlLpl3 Resource	RidsNrrDorlLpl1 Resource	
RidsNrrLALRonewicz Resource	RidsNrrLASRohrer Resource	BVaidya, NRR
RidsNrrPMDavisBesse	RidsNrrPMBeaver Valley	RidsNrrPMPerryResource
RidsAcrsMailCenter Resource	RidsOgcRp Resource	RidsRgn1MailCenter
RidsRgn3MailCenter	RecordsAmend Resource	
C. Tilton, NRR/STSB	J. Huang, NRR/EPNB	

ADAMS Accession No. ML17081A509
(*) No substantial change to SE Input Memo ML17074A227

OFFICE	NRR/LPL3/PM	NRR/LPL3/LA	NRR/EPNB/BC	NRR/STSB/BC*
NAME	BVaidya	SRohrer	DAlley	AKlein (JWhitman for)
DATE	4/5/17	3/28/17	5/4/17	4/17/17

OFFICE	OGC – NLO	NRR/LPL3/BC	NRR/LPL3/PM
NAME	AGhosh	DWrona	BVaidya
DATE	5/11/17	5/11/17	5/11/17

OFFICIAL RECORD COPY