December 2, 1997

Mr. John K. Wood
Vice President - Nuclear, Davis-Besse
Centerior Service Company
c/o Toledo Edison Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: AMENDMENT NO. 218 TO FACILITY OPERATING LICENSE NO. NPF-3 - DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1 (TAC NOS. M97441, M97902, AND M98520)

Dear Mr. Wood:

The Commission has issued the enclosed Amendment No. 218 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station (DBNPS), Unit 1. The amendment revises the Technical Specifications (TSs) in response to your License Amendment Requests (LARs) dated December 11, 1996 (LAR 95-27, TAC No. M97441, as supplemented by letter dated January 6, 1997), January 30, 1997 (LAR 95-24, TAC No. M97902, as supplemented by letter dated September 15, 1997), and April 18, 1997 (LAR 96-14, TAC No. M98520).

This amendment extends surveillance requirement intervals from 18 to 24 months based on the results of the DBNPS Instrument Drift Study; revises setpoints based on the Drift Study and guidance in NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April 1995; and revises TS 2.2, "Limiting Safety System Settings," based on the results of the revised Framatome Reactor Protection System string error and setpoint allowable value calculations and guidance in NUREG-1430, Revision 1. Additional administrative changes have also been made.

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

Sincerely,

Original signed by:

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures: 1. Amendment No. 218 to License No. NPF-3
2. Safety Evaluation

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

December 2, 1997

Mr. John K. Wood
Vice President - Nuclear, Davis-Besse
Centerior Service Company
c/o Toledo Edison Company
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Oak Harbor, OH 43449-9760

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Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures:
1. Amendment No. 218 to License No. NPF-3
2. Safety Evaluation

cc w/encls: See next page

cc:

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

TOLEDO EDISON COMPANY

CENTERIOR SERVICE COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 218
License No. NPF-3

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

   A. The applications for amendment by the Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company (the licensees) dated December 11, 1996 (as supplemented by letter dated January 6, 1997), January 30, 1997 (as supplemented by letter dated September 15, 1997), and April 18, 1997, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 218, are hereby incorporated in the license. The Toledo Edison Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented not later than 120 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: December 2, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 218

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

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

Remove	Insert
TS 1-8	TS 1-8
TS 2-4 through 2-6 (3 pages)	TS 2-4 through 2-6 (3 pages)
TS Bases 2-4 through 2-7 (4 pages)	TS Bases 2-4 through 2-7 (4 pages)
TS 3/4 3-1	TS 3/4 3-1
TS 3/4 3-7 through 3/4 3-9 (3 pages)	TS 3/4 3-7 through 3/4 3-9 (3 pages)
TS 3/4 3-12	TS 3/4 3-12
TS 3/4 3-12a	TS 3/4 3-12a
TS 3/4 3-13	TS 3/4 3-13
TS 3/4 3-21 through 3/4 3-23 (3 pages)	TS 3/4 3-21 through 3/4 3-23 (3 pages)
TS 3/4 3-28	TS 3/4 3-28
TS 3/4 3-30	TS 3/4 3-30
TS 3/4 3-34	TS 3/4 3-34
TS 3/4 3-43	TS 3/4 3-43
TS 3/4 3-49	TS 3/4 3-49
TS 3/4 3-50	TS 3/4 3-50
TS 3/4 4-4	TS 3/4 4-4
TS 3/4 4-14	TS 3/4 4-14
TS 3/4 5-2	TS 3/4 5-2
TS 3/4 5-4	TS 3/4 5-4
TS 3/4 7-5	TS 3/4 7-5
TS 3/4 7-5a	TS 3/4 7-5a
TS Bases 3/4 3-1	TS Bases 3/4 3-1
TS Bases 3/4 3-1a	TS Bases 3/4 3-1a
TS Bases 3/4 5-2a	TS Bases 3/4 5-2a
TS Bases 3/4 5-2b	TS Bases 3/4 5-2b

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 6 months.*
E	At least once per 18 months.*
R	At least once per 24 months.*
S/U	Prior to each reactor startup.
N/A	Not applicable.

*In these Technical Specifications, 6 months is defined to be 184 days, 18 months is defined to be 550 days, and 24 months is defined to be 730 days.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM SETPOINTS

2.2.1 The Reactor Protection System instrumentation setpoints shall be set consistent with the Allowable Values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a Reactor Protection System instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Allowable Value.

Table 2.2-1 Reactor Protection System Instrumentation Trip Setpoints

Functional unit	Allowable values
1. Manual reactor trip	Not applicable.
2. High flux	≤105.1% of RATED THERMAL POWER with four pumps operating*
	≤80.6% of RATED THERMAL POWER with three pumps operating*
3. RC high temperature	≤618°F*
4. Flux --Δflux/flow(1)	Pump allowable values not to exceed the limit lines shown in the CORE OPERATING LIMITS REPORT for four and three pump operation.*
5. RC low pressure(1)	≥1900.0 psig*
6. RC high pressure	≤2355.0 psig*
7. RC pressure-temperature(1)	≥(16.00 T_{out}°F - 7957.5) psig*
8. High flux/number of RC pumps on(1)	≤55.1% of RATED THERMAL POWER with one pump operating in each loop*
	≤0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop*
	≤0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating*
9. Containment pressure high	≤4 psig*

Table 2.2-1. (Cont'd)

(1) Trip may be manually bypassed when RCS pressure ≤1820 psig by actuating shutdown bypass provided that:

a. The high flux trip setpoint is ≤5% of RATED THERMAL POWER.

b. The shutdown bypass high pressure trip setpoint of ≤1820 psig is imposed.

c. The shutdown bypass is removed when RCS pressure >1820 psig.

*Allowable value for CHANNEL FUNCTIONAL TEST.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The reactor protection system instrumentation Allowable Values specified in Table 2.2-1 have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

The shutdown bypass provides for bypassing certain functions of the reactor protection system in order to permit control rod drive tests, zero power PHYSICS TESTS and certain startup and shutdown procedures. The purpose of the shutdown bypass high pressure trip is to prevent normal operation with shutdown bypass activated. This high pressure setpoint is lower than the normal low pressure setpoint so that the reactor must be tripped before the bypass is initiated. The high flux setpoint of ≤5.0% prevents any significant reactor power from being produced. Sufficient natural circulation would be available to remove 5.0% of RATED THERMAL POWER if none of the reactor coolant pumps were operating.

Manual Reactor Trip

The manual reactor trip is a redundant channel to the automatic reactor protection system instrumentation channels and provides manual reactor trip capability.

High Flux

A high flux trip at high power level (neutron flux) provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.

During normal station operation, reactor trip is initiated when the reactor power level reaches the Allowable Value ≤ 105.1% of rated power. Due to transient overshoot, heat balance, and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which was used in the safety analysis.

RC High Temperature

The RC high temperature trip ≤618°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Flux -- ΔFlux/Flow

The power level Allowable Value produced by the reactor coolant system flow is based on a flux-to-flow ratio which has been established to accommodate flow decreasing transients from high power where protection is not provided by the high flux/number of reactor coolant pumps on trips.

The power level Allowable Value produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate.

For safety calculations the instrumentation errors for the power level were used. Full flow rate is defined as the flow calculated by the heat balance at 100% power. At the time of the calibration the RCS flow will be greater than or equal to the value in Table 3.2-2.

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kW/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by a flux-to-flow ratio such that the boundaries of the figure in the CORE OPERATING LIMITS REPORT are produced.

RC Pressure - Low, High, and Pressure Temperature

The high and low trips are provided to limit the pressure range in which reactor operation is permitted.

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RC high pressure setpoint is reached before the high flux setpoint. The Allowable Value for RC high pressure, 2355 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RC high pressure trip is backed up by the pressurizer code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, $\leq$ 2525 psig. The RC high pressure trip also backs up the high flux trip.

The RC low pressure, 1900.0 psig, and RC pressure-temperature ($16.00\ T_{out}$ - 7957.5) psig, Allowable Values have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

High Flux/Number of Reactor Coolant Pumps On

In conjunction with the flux - Δflux/flow trip the high flux/number of reactor coolant pumps on trip prevents the minimum core DNBR from decreasing below the minimum allowable DNB ratio by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

Containment High Pressure

The Containment High Pressure Allowable Value $\leq$ 4 psig, provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the containment vessel or a loss-of-coolant accident, even in the absence of a RC Low Pressure trip.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the Reactor Protection System instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each Reactor Protection System instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The total bypass function shall be demonstrated OPERABLE at least once per REFUELING INTERVAL during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per REFUELING INTERVAL. Each test shall include at least one channel per function such that all channels are tested at least once every N times the REFUELING INTERVAL where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 4.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. High Flux	S	D(2), and Q(6,9)	N.A.	1, 2
3. RC High Temperature	S	R	SA(9)	1, 2
4. Flux - ΔFlux - Flow	S(4)	M(3) and Q(6,7,9)	N.A.	1, 2
5. RC Low Pressure	S	R	SA(9)	1, 2
6. RC High Pressure	S	R	SA(9)	1, 2
7. RC Pressure-Temperature	S	R	SA(9)	1, 2
8. High Flux/Number of Reactor Coolant Pumps On	S	Q(6,9)	N.A.	1, 2
9. Containment High Pressure	S	E	SA(9)	1, 2
10. Intermediate Range, Neutron Flux and Rate	S	E(6)	N.A.(5)	1, 2 and *
11. Source Range, Neutron Flux and Rate	S	E(6)	N.A.(5)	2, 3, 4 and 5
12. Control Rod Drive Trip Breakers	N.A.	N.A.	M(8,9) and S/U(1)(8)	1, 2 and *
13. Reactor Trip Module Logic	N.A.	N.A.	M(9)	1, 2 and *
14. Shutdown Bypass High Pressure	S	R	SA(9)	2**, 3**, 4**, 5**
15. SCR Relays	N.A.	N.A.	R	1, 2 and *

TABLE 4.3-1 (Continued)

NOTATION

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - When THERMAL POWER [TP] is above 50% of RATED THERMAL POWER [RTP] and at a steady state, compare out-of-core measured AXIAL POWER IMBALANCE [API_o] to incore measured AXIAL POWER IMBALANCE [API_I] as follows:

$$\frac{RTP}{TP} [API_o - API_I] = \text{Offset Error}$$

Recalibrate if the absolute value of the Offset Error is $\geq$ 2.5%.

(4) - AXIAL POWER IMBALANCE and loop flow indications only.

(5) - CHANNEL FUNCTIONAL TEST is not applicable. Verify at least one decade overlap prior to each reactor startup if not verified in previous 7 days.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Flow rate measurement sensors may be excluded from CHANNEL CALIBRATION. However, each flow measurement sensor shall be calibrated at least once each REFUELING INTERVAL.

(8) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of both the undervoltage and shunt trip devices of the Reactor Trip Breakers.

(9) - Performed on a STAGGERED TEST BASIS.

* - With any control rod drive trip breaker closed.

** - When Shutdown Bypass is actuated.

INSTRUMENTATION

3/4.3.2 SAFETY SYSTEM INSTRUMENTATION

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Safety Features Actuation System (SFAS) functional units shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4, with the exception of Instrument Strings Functional Units d and e and Interlock Channels Functional Unit a which shall be set consistent with the Allowable Value column of Table 3.3-4, and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With a SFAS functional unit trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the functional unit inoperable and apply the applicable ACTION requirement of Table 3.3-3, until the functional unit is restored to OPERABLE status with the trip setpoint adjusted consistent with Table 3.3-4.

b. With a SFAS functional unit inoperable, take the action shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each SFAS functional unit shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST during the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of functional units affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per REFUELING INTERVAL during CHANNEL CALIBRATION testing of each functional unit affected by bypass operation.

4.3.2.1.3 The SAFETY FEATURES RESPONSE TIME of each SFAS function shall be demonstrated to be within the limit at least once per REFUELING INTERVAL. Each test shall include at least one functional unit per function such that all functional units are tested at least once every N times the REFUELING INTERVAL where N is the total number of redundant functional units in a specific SFAS function as shown in the "Total No. of Units" Column of Table 3.3-3.

TABLE 3.3-3 (Continued)
TABLE NOTATION

*	Trip function may be bypassed in this MODE with RCS pressure below 1800 psig. Bypass shall be automatically removed when RCS pressure exceeds 1800 psig.
**	Trip function may be bypassed in this MODE with RCS pressure below 660 psig. Bypass shall be automatically removed when RCS pressure exceeds 660 psig.
***	DELETED
****	This instrumentation, or the containment purge and exhaust system noble gas monitor (with the containment purge and exhaust system in operation), must be OPERABLE during CORE ALTERATIONS or movement of irradiated fuel within containment to meet the requirements of Technical Specification 3.9.4. When using the containment purge and exhaust system noble gas monitor, SFAS is not required to be OPERABLE in MODE 6.
*****	All functional units may be bypassed for up to one minute when starting each Reactor Coolant Pump or Circulating Water Pump.
******	When either Decay Heat Isolation Valve is open.
#	The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 10 - With the number of OPERABLE functional units one less than the Total Number of Units, STARTUP and/or POWER OPERATION may proceed provided both of the following conditions are satisfied:

a. The inoperable functional unit is placed in the tripped condition within one hour.

b. The Minimum Units OPERABLE requirement is met; however, one additional functional unit may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 11 - With any component in the Output Logic inoperable, trip the associated components within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE Units one less than the Total Number of Units, restore the inoperable functional unit to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 13 - a. With less than the Minimum Units OPERABLE and indicated reactor coolant pressure $\geq$ 328 psig, both Decay Heat Isolation Valves (DH11 and DH12) shall be verified closed.

b. With Less than the Minimum Units OPERABLE and indicated reactor coolant pressure < 328 psig operation may continue; however, the functional unit shall be OPERABLE prior to increasing indicated reactor coolant pressure above 328 psig.

ACTION 14 - With less than the Minimum Units OPERABLE and indicated reactor coolant pressure < 328 psig, operation may continue; however, the functional unit shall be OPERABLE prior to increasing indicated reactor coolant pressure above 328 psig, or the inoperable functional unit shall be placed in the tripped state.

ACTION 15 - a. With the number of OPERABLE units one less than the Minimum Units Operable per Bus, place the inoperable unit in the tripped condition within one hour. For functional unit 4.a the sequencer shall be placed in the tripped condition by physical removal of the sequencer module. The inoperable functional unit may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

b. With the number of OPERABLE units two less than the Minimum Units Operable per Bus, declare inoperable the Emergency Diesel Generator associated with the functional units not meeting the required minimum units OPERABLE and take the ACTION required of Specification 3.8.1.1.

TABLE 3.3-4

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
INSTRUMENT STRINGS		
a. Containment Radiation	< 4 x Background at RATED THERMAL POWER	< 4 x Background at RATED THERMAL POWER#
b. Containment Pressure - High	≤ 18.4 psia	≤ 18.52 psia#
c. Containment Pressure - High-High	≤ 38.4 psia	≤ 38.52 psia#
d. RCS Pressure - Low	N.A.	≥ 1576.2 psig##
e. RCS Pressure - Low-Low	N.A.	≥ 441.42 psig##
f. BWST Level	≥ 89.5 and ≤ 100.5 in H_2O	≥ 88.3 and ≤ 101.7 in H_2O#
SEQUENCE LOGIC CHANNELS		
a. Essential Bus Feeder Breaker Trip (90%)	≥ 3744 volts for ≤ 7.8 sec	≥ 3558 volts ≤ 7.8 sec
b. Diesel Generator Start, Load Shed on Essential Bus (59%)	≥ 2071 and ≤ 2450 volts for 0.5 ± 0.1 sec	≥ 2071 and ≤ 2450 volts for 0.5 ± 0.1 sec#
INTERLOCK CHANNELS		
a. Decay Heat Isolation Valve and Pressurizer Heater	N.A.	< 328 psig##*

#Allowable Value for CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION
*Referenced to the RCS Pressure instrumentation tap.
Allowable Value for CHANNEL FUNCTIONAL TEST

TABLE 4.3-2

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. INSTRUMENT STRINGS				
a. Containment Radiation - High	S	E	M	1, 2, 3, 4, 6 #
b. Containment Pressure - High	S	E	M(2)	1, 2, 3
c. Containment Pressure - High-High	S	E	M(2)	1, 2, 3
d. RCS Pressure - Low	S	R	M	1, 2, 3
e. RCS Pressure - Low-Low	S	R	M	1, 2, 3
f. BWST Level - Low-Low	S	E	M	1, 2, 3
2. OUTPUT LOGIC				
a. Incident Level #1: Containment Isolation	S	E	M	1, 2, 3, 4, 6 #
b. Incident Level #2: High Pressure Injection and Starting Diesel Generators	S	E	M	1, 2, 3, 4
c. Incident Level #3: Low Pressure Injection	S	E	M	1, 2, 3, 4
d. Incident Level #4: Containment Spray	S	E	M	1, 2, 3, 4
e. Incident Level #5: Containment Sump Recirculation Permissive	S	E	M	1, 2, 3, 4
3. MANUAL ACTUATION				
a. SFAS (Except Containment Spray and Emergency Sump Recirculation)	NA	NA	M(1)	1, 2, 3, 4, 6 #
b. Containment Spray	NA	NA	M(1)	1, 2, 3
4. SEQUENCE LOGIC CHANNELS	S	NA	M	1, 2, 3, 4

TABLE 4.3-2 (Continued)

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. INTERLOCK CHANNELS				
a. Decay Heat Isolation Valve	S	R	**	1, 2, 3
b. Pressurizer Heater	S	R	**	3 ##

**See Specification 4.5.2.d.1

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per REFUELING INTERVAL. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days.

(2) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either vacuum or pressure to the appropriate side of the transmitter.

These surveillance requirements in conjunction with those of Section 4.9.4 apply during CORE ALTERATIONS or movement of irradiated fuel within the containment only if using the SFAS area radiation monitors listed in Table 3.3-3, Items 1a, 2a, and 3a, in lieu of the containment purge and exhaust system noble gas monitor.

When either Decay Heat Isolation Valve is open.

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.2 The Steam and Feedwater Rupture Control System (SFRCS) instrumentation channels shown in Table 3.3-11 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-12, with the exception of the Steam Generator Level-Low Functional Unit which shall be set consistent with the Allowable Value column of Table 3.3-12, and with RESPONSE TIMES as shown in Table 3.3-13.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With a SFRCS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-12, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-11, until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with Table 3.3-12.

b. With a SFRCS instrumentation channel inoperable, take the action shown in Table 3.3-11.

SURVEILLANCE REQUIREMENTS

4.3.2.2.1 Each SFRCS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST during the MODES and at the frequencies shown in Table 4.3-11.

4.3.2.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per REFUELING INTERVAL during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.2.3 The STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM RESPONSE TIME of each SFRCS function shall be demonstrated to be within the limit at least once per REFUELING INTERVAL. Each test shall include at least one channel per function such that all channels are tested at least once every N times the REFUELING INTERVAL where N is the total number of redundant channels in a specific SFRCS function as shown in the "Total No. of Channels" Column of Table 3.3-11.

TABLE 3.3-12

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINTS	ALLOWABLE VALUES
1. Steam Line Pressure - Low	≥ 591.6 psig	≥ 591.6 psig* ≥ 586.6 psig**
2. Steam Generator Level - Low(1)	N.A.	≥ 16.9"*
3. Steam Generator Feedwater Differential Pressure - High(2)	≤ 197.6 psid	≤ 197.6 psid* ≤ 199.6 psid**
4. Reactor Coolant Pumps - Loss of	High ≤ 1384.6 amps Low ≥ 106.5 amps	≤ 1384.6 amps# ≥ 106.5 amps#

(1) Actual water level above the lower steam generator tubesheet.
(2) Where differential pressure is steam generator minus feedwater pressure.
*Allowable Value for CHANNEL FUNCTIONAL TEST
**Allowable Value for CHANNEL CALIBRATION
#Allowable Value for CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION

TABLE 4.3-11

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Instrument Channel			
a. Steam Line Pressure - Low	S	E	M
b. Steam Generator Level - Low	S	R	M
c. Steam Generator - Feedwater Differential Pressure - High	S	E	M
d. Reactor Coolant Pumps - Loss of	S	E	M
2. Manual Actuation	NA	NA	R

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area Emergency Ventilation System Actuation	S	E	M	**
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity RCS Leakage Detection	S	E	M	1, 2, 3 & 4
ii. Particulate Activity RCS Leakage Detection	S	E	M	1, 2, 3 & 4

**With fuel in the storage pool or building

INSTRUMENTATION

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5.1 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.

3.3.3.5.2 The control circuits and transfer switches required for a serious control room or cable spreading room fire shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than required by Table 3.3-9, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. With one or more control circuits or transfer switches required for a serious control room or cable spreading room fire inoperable, restore the inoperable circuit(s) or switch(es) to OPERABLE status within 30 days, or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the circuit(s) or switch(es) to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

4.3.3.5.2 At least once per REFUELING INTERVAL, verify each control circuit and transfer switch required for a serious control room or cable spreading room fire is capable of performing the intended function.

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	SG Outlet Steam Pressure	M	R
2.	RC Loop Outlet Temperature	M	R
3.	RC Loop Pressure	M	R
4.	Pressurizer Level	M	R
5.	SG Startup Range Level	M	R
6.	Containment Vessel Post-Accident Radiation		
	a.) Containment High Range Radiation	M	R
	b.) Containment Wide Range Noble Gas	M	E
7.	High Pressure Injection Flow	M	E
8.	Low Pressure Injection (DHR) Flow	M	E
9.	Auxiliary Feedwater Flow Rate	M	E
10.	RC System Subcooling Margin Monitor	M	R
11.	PORV Position Indicator	M	R
12.	PORV Block Valve Position Indicator	M	R
13.	Pressurizer Safety Valve Position Indicator	M	R
14.	BWST Level	S	E
15.	Containment Normal Sump Level	M	R
16.	Containment Wide Range Water Level	M	R

TABLE 4.3-10 (Continued)

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
17.	Containment Wide Range Pressure	M	R
18.	Incore Thermocouples	M	E
19.	Reactor Coolant Hot Leg Level (Wide Range)	M	R
20.	Neutron Flux (Wide Range)	M	E**
21.	Neutron Flux (Source Range)	M	E**

**Neutron detectors may be excluded from CHANNEL CALIBRATION.

SAFETY VALVES AND PILOT OPERATED RELIEF VALVE - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of $\leq$ 2525 psig.* When not isolated, the pressurizer pilot operated relief valve shall have a trip setpoint of $\geq$ 2435 psig and an allowable value of $\geq$2435 psig.**

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 For the pressurizer code safety valves, there are no additional Surveillance Requirements other than those required by Specification 4.0.5. For the pressurizer pilot operated relief valve a CHANNEL CALIBRATION check shall be performed each REFUELING INTERVAL.

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* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** Allowable value for CHANNEL CALIBRATION check.

SURVEILLANCE REQUIREMENTS (Continued)

b. Containment sump level and flow monitoring system-performance of CHANNEL CALIBRATION at least once each REFUELING INTERVAL.

c. Containment atmosphere gaseous monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days, and within 6 hours of each solution volume increase of $\geq$ 80 gallons that is not the result of addition from the borated water storage tank (BWST), by verifying the boron concentration of the CFT solution.

c. At least once per 31 days by verifying that power to the isolation valve operator is disconnected by locking the breakers in the open position.

d. At least once per REFUELING INTERVAL by verifying that each core flooding tank isolation valve opens automatically and is interlocked against closing whenever the Reactor Coolant System pressure exceeds 800 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once each REFUELING INTERVAL, or prior to operation after ECCS piping has been drained by verifying that the ECCS piping is full of water by venting the ECCS pump casings and discharge piping high points.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment emergency sump and cause restriction of the pump suction during LOCA conditions. This visual inspection shall be performed:

   1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

   2. For all areas of containment affected by an entry, at least once daily while work is ongoing and again during the final exit after completion of work (containment closeout) when CONTAINMENT INTEGRITY is established.

d. At least once each REFUELING INTERVAL by:

   1. Verifying that the interlocks:

      a) Close DH-11 and DH-12 and deenergize the pressurizer heaters, if either DH-11 or DH-12 is open and a simulated reactor coolant system pressure which is greater than the Allowable Value (<328 psig) is applied. The interlock to close DH-11 and/or DH-12 is not required if the valve is closed and 480 V AC power is disconnected from its motor operators.

      b) Prevent the opening of DH-11 and DH-12 when a simulated or actual reactor coolant system pressure which is greater than the Allowable Value (<328 psig) is applied.

   2. a) A visual inspection of the containment emergency sump which verifies that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

      b) Verifying that on a Borated Water Storage Tank (BWST) Low-Low Level interlock trip, with the motor operators for the BWST outlet isolation valves and the containment emergency sump recirculation valves energized, the BWST Outlet Valve HV-DH7A (HV-DH7B) automatically close in $\leq$75 seconds after the operator manually pushes the control switch to open the Containment Emergency Sump Valve HV-DH9A (HV-DH9B) which should be verified to open in $\leq$75 seconds.

   3. Deleted

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying that each valve (power operated or automatic) in the flow path is in its correct position.

2. Verifying that all manual valves in the auxiliary feedwater pump suction and discharge lines that affect the system's capacity to deliver water to the steam generator are locked in their proper position.

3. Verifying that valves CW 196, CW 197, FW 32, FW 91 and FW 106 are closed.

c. At least once each REFUELING INTERVAL by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Steam and Feedwater Rupture Control System actuation test signal.

2. Verifying that each pump starts automatically upon receipt of a Steam and Feedwater Rupture Control System actuation test signal. The provisions of Specification 4.0.4 are not applicable for entry in MODE 3.

3. Verifying that there is a flow path from each auxiliary feedwater pump to both steam generators by pumping water from the Condensate Storage Tank with each pump to both steam generators.

The flow paths shall be verified by either steam generator level change or Auxiliary Feedwater Safety Grade Flow Indication. Verification of the Auxiliary Feedwater System's flow capacity is not required.

d. The Auxiliary Feed Pump Turbine Steam Generator Level Control System shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days, and a CHANNEL CALIBRATION at least once each REFUELING INTERVAL.

e. The Auxiliary Feed Pump Suction Pressure Interlocks shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days, and a CHANNEL CALIBRATION at least once each REFUELING INTERVAL.

SURVEILLANCE REQUIREMENTS (Continued)

f. After any modification or repair to the Auxiliary Feedwater System that could affect the system's capability to deliver water to the steam generator, the affected flow path shall be demonstrated available as follows:

1. If the modification or repair is downstream of the test flow line, each auxiliary feed pump(s) associated with the affected flow path shall pump water from the Condensate Storage Tank to the steam generator(s) associated with the affected flow path; and the flow path availability will be verified by steam generator level change or Auxiliary Feedwater Safety Grade Flow Indication.

2. If the modification or repair is upstream of the test flow line, the auxiliary feed pump shall pump water through the Auxiliary Feedwater System to the test flow line; and the flow path availability will be verified by flow indication in the test flow line.*

This Surveillance Testing shall be performed prior to entering MODE 3 if the modification is made in MODES 4, 5 or 6. Verification of the Auxiliary Feedwater System's flow capacity is not required.

g. Following each extended cold shutdown (> 30 days in MODE 5), by:

1. Verifying that there is a flow path from each auxiliary feedwater pump to both steam generators by pumping Condensate Storage Tank water with each pump to both steam generators. The flow paths shall be verified by either steam generator level change or Auxiliary Feedwater Safety Grade Flow Indication. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

Verification of the Auxiliary Feedwater System's flow capacity is not required.

4.7.1.2.2 The Auxiliary Feed Pump Turbine Inlet Steam Pressure Interlocks shall be demonstrated OPERABLE when the steam line pressure is greater than 275 psig, by performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days, and a CHANNEL CALIBRATION at least once each REFUELING INTERVAL. The CHANNEL FUNCTIONAL TEST shall be performed within 24 hours after exceeding 275 psig during each plant startup, if the test has not been performed within the last 31 days.

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* When conducting tests of an auxiliary feedwater train in MODES 1, 2, and 3 which require local manual realignment of valves that make the train inoperable, the Motor Driven Feedwater Pump and its associated flow paths shall be OPERABLE per Specification 3.7.1.7 during the performance of this surveillance. If the Motor Driven Feedwater Pump or an associated flow path is inoperable, a dedicated individual shall be stationed at the realigned auxiliary feedwater train's valves (in communication with the control room) able to restore the valves to normal system OPERABLE status.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION

The OPERABILITY of the RPS, SFAS and SFRCS instrumentation systems ensure that 1) the associated action and/or trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for RPS, SFAS and SFRCS purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

For the RPS, SFAS Table 3.3-4 Functional Unit Instrument Strings d and e and Interlock Channel a, and SFRCS Table 3.3-12 Functional Unit 2:

Only the Allowable Value is specified for each Function. Nominal trip setpoints are specified in the setpoint analysis. The nominal trip setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the specific setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis to account for instrument uncertainties appropriate to the trip parameter. These uncertainties are defined in the specific setpoint analysis.

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Setpoints must be found within the specified Allowable Values. Any setpoint adjustment shall be consistent with the assumptions of the current specific setpoint analysis.

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis.

The frequency is justified by the assumption of an 18 or 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(Next page is B3/4 3-1a)

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION (Continued)

The measurement of response time at the specified frequencies provides assurance that the RPS, SFAS, and SFRCS action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

The actuation logic for Functional Units 4.a., 4.b., and 4.c. of Table 3.3-3, Safety Features Actuation System Instrumentation, is designed to provide protection and actuation of a single train of safety features equipment, essential bus or emergency diesel generator. Collectively, Functional Units 4.a., 4.b., and 4.c. function to detect a degraded voltage condition on either of the two 4160 volt essential buses, shed connected loads, disconnect the affected bus(es) from the offsite power source and start the associated emergency diesel generator. In addition, if an SFAS actuation signal is present under these conditions, the sequencer channels for the two SFAS channels which actuate the train of safety features equipment powered by the affected bus will automatically sequence these loads onto the bus to prevent overloading of the emergency diesel generator. Functional Unit 4.a. has a total of four units, one associated with each SFAS channel (i.e., two for each essential bus). Functional Units 4.b. and 4.c. each have a total of four units, (two associated with each essential bus); each unit consisting of two undervoltage relays and an auxiliary relay.

An SFRCS channel consists of 1) the sensing device(s), 2) associated logic and output relays (including Isolation of Main Feedwater Non Essential Valves and Turbine Trip), and 3) power sources.

The SFRCS response time for the turbine stop valve closure is based on the combined response times of main steam line low pressure sensors, logic cabinet delay for main steam line low pressure signals and closure time of the turbine stop valves. This SFRCS response time ensures that the auxiliary feedwater to the unaffected steam generator will not be isolated due to a SFRCS low pressure trip during a main steam line break accident.

Safety-grade anticipatory reactor trip is initiated by a turbine trip (above 45 percent of RATED THERMAL POWER) or trip of both main feedwater pump turbines. This anticipatory trip will operate in advance of the reactor coolant system high pressure reactor trip to reduce the peak reactor coolant system pressure and thus reduce challenges to the pilot operated relief valve. This anticipatory reactor trip system was installed to satisfy Item II.K.2.10 of NUREG-0737. The justification for the ARTS turbine trip arming level of 45% is given in BAW-1893, October, 1985.

Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

Containment Emergency Sump Recirculation Valves DH-9A and DH-9B are de-energized during MODES 1, 2, 3 and 4 to preclude postulated inadvertent opening of the valves in the event of a Control Room fire, which could result in draining the Borated Water Storage Tank to the Containment Emergency Sump and the loss of this water source for normal plant shutdown. Re-energization of DH-9A and DH-9B is permitted on an intermittent basis during MODES 1, 2, 3 and 4 under administrative controls. Station procedures identify the precautions which must be taken when re-energizing these valves under such controls.

Borated Water Storage Tank (BWST) outlet isolation valves DH-7A and DH-7B are de-energized during MODES 1, 2, 3, and 4 to preclude postulated inadvertent closure of the valves in the event of a fire, which could result in a loss of the availability of the BWST. Re-energization of valves DH-7A and DH-7B is permitted on an intermittent basis during MODES 1, 2, 3, and 4 under administrative controls. Station procedures identify the precautions which must be taken when re-energizing these valves under such controls.

The Decay Heat Isolation Valve and Pressurizer Heater Interlock setpoint is based on preventing over-pressurization of the Decay Heat Removal System normal suction line piping. The value stated is the RCS pressure at the sensing instrument's tap. It has been adjusted to reflect the elevation difference between the sensor's location and the pipe of concern.

3/4.5.4 BORATED WATER STORAGE TANK

The OPERABILITY of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on the BWST minimum volume and boron concentration ensure that:

1) sufficient water is available within containment to permit recirculation cooling flow to the core following manual switchover to the recirculation mode, and

2) The reactor will remain at least 1% $\Delta k/k$ subcritical in the cold condition at $70^{\circ}F$, xenon free, while only crediting 50% of the control rods' worth following mixing of the BWST and the RCS water volumes.

These assumptions ensure that the reactor remains subcritical in the cold condition following mixing of the BWST and the RCS water volumes.

With either the BWST boron concentration or BWST borated water temperature not within limits, the condition must be corrected in eight hours. The eight hour limit to restore the temperature or boron concentration to within limits was developed considering the time required to change boron concentration or temperature and assuming that the contents of the BWST are still available for injection.

The bottom 4 inches of the BWST are not available, and the instrumentation is calibrated to reflect the available volume. The limits on water volume, and boron concentration ensure a pH value of between 7.0 and 11.0 of the solution sprayed within the containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 218 TO FACILITY OPERATING LICENSE NO. NPF-3

TOLEDO EDISON COMPANY

CENTERIOR SERVICE COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

DOCKET NO. 50-346

1.0 INTRODUCTION

By letters dated December 11, 1996 (as supplemented January 6, 1997), January 30, 1997 (as supplemented September 15, 1997), and April 18, 1997, Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, the licensees for the Davis-Besse Nuclear Power Station (DBNPS), Unit 1, requested NRC approval to implement amendments to its operating license NPF-3, by incorporating modifications to the Technical Specifications (TSs). The NRC staff reviewed these submittals and by letter dated June 11, 1997, the staff requested additional information. On July 10, 1997, a conference call was held with the licensees to further discuss the staff request, and on August 21, 1997, a meeting was held with the licensees to discuss their responses. By letter dated September 15, 1997, the licensees provided clarification to their original submittals.

In accordance with the licensees' submittals, the licensees have:

(a) Proposed revisions to extend surveillance requirement intervals from 18 to 24 months based on the results of the DBNPS Instrument Drift Study;

(b) Proposed setpoint revisions based on the results of the DBNPS Instrument Drift Study and guidance in NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April 1995;

(c) Proposed revisions to TS 2.2, "Limiting Safety System Settings," based on the results of revised Framatome Reactor Protection System string error and setpoint allowable value calculations and guidance in NUREG-1430, Revision 1; and

(d) Proposed administrative revisions supporting the preceding areas of revision.

2.0 BACKGROUND

Improved reactor fuels allow licensees to consider an increase in the duration of the fuel cycle for their facilities. The staff has reviewed requests for individual plants to modify TS surveillance intervals to be compatible with a 24-month fuel cycle. The staff issued Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," on April 2, 1991, to provide generic guidance to licensees for preparing such license amendment requests.

TSs that specify an 18-month surveillance interval could be changed to state that these surveillances are to be performed once per refueling interval. The notation for surveillance intervals would then be changed to include the definition of a "Refueling Interval" with the existing "R" notation for surveillances that are generally performed during a refueling outage. The frequency for the interval indicated by this notation would also be changed from 18 months to "at least once every 24 months." The TS provision to extend surveillances by 25 percent of the specified interval would extend the time limit for completing these surveillances from the existing limit of 22.5 months to a maximum of 30 months.

Licensees must address instrument drift when proposing an increase in the surveillance interval for calibrating instruments that perform safety functions which includes providing the capability for safe shutdown. The effect of the increased calibration interval on instrument errors must be addressed because instrument errors caused by drift were considered when determining safety system setpoints based on design basis accident analyses.

In accordance with GL 91-04, to provide an acceptable basis for increasing the calibration interval for instruments that are used to perform safety functions, the licensees should:

(1) Confirm that instrument drift as determined by as-found and as-left calibration data from surveillance and maintenance records has not, except on rare occasions, exceeded acceptable limits for a calibration interval;

(2) Confirm that the values of drift for each instrument type (make, model number, and range) and application have been determined with a high probability and a high degree of confidence. Provide a summary of the methodology and assumptions used to determine the rate of instrument drift with time based upon historical plant calibration data;

(3) Confirm that the magnitude of instrument drift has been determined with a high probability and a high degree of confidence for a bounding calibration interval of 30 months for each instrument type (make, model number, and range) and application that performs a safety function. Provide a list of the channels by TS section that identifies these instrument applications;

(4) Confirm that a comparison of the projected instrument drift errors has been made with the values of drift used in the setpoint analysis. If this results in revised setpoints to accommodate larger drift errors, provide proposed TS changes to update trip setpoints. If the drift errors result in a revised safety analysis to support existing setpoints, provide a summary of the updated analysis conclusions to confirm that safety limits and safety analysis assumptions are not exceeded;

(5) Confirm that the projected instrument errors caused by drift are acceptable for control of plant parameters to effect a safe shutdown with the associated instrumentation;

(6) Confirm that all conditions and assumptions of the setpoint and safety analyses have been checked and are appropriately reflected in the acceptance criteria of plant surveillance procedures for channel checks, channel functional tests, and channel calibrations;

(7) Provide a summary description of the program for monitoring and assessing the effects of increased calibration surveillance intervals on instrument drift and its effect on safety; and

(8) Maintain a program to monitor calibration results and the effect on instrument drift that will accompany the increase in calibration intervals.

For other 18-month surveillances, licensees should evaluate the effect on safety of the change in surveillance intervals to accommodate a 24-month fuel cycle. This evaluation should support a conclusion that the effect on safety is small. In addition, licensees should confirm that historical maintenance and surveillance data do not invalidate this conclusion. Licensees should confirm that the performance of surveillance at the bounding surveillance interval limit provided to accommodate a 24-month fuel cycle would not invalidate any assumption in the plant licensing basis. In consideration of these confirmations, the licensees need not quantify the effect of the change in surveillance intervals on the availability of individual systems or components.

3.0 EVALUATION

The licensees have proposed the following changes to the TSs:

Proposed Change

TS Table 1.2, "Frequency Notation," change the notation "R" to "E" to define a frequency of "at least once per 18 months," and redefine "R" to "at least once per 24 months." Also, the footnote with an asterisk has been revised to add that 24 months is defined to be 730 days.

Evaluation

The proposed change to TS Table 1.2 is an administrative change only consistent with the proposed revised surveillance interval, and is, therefore, acceptable.

Proposed Change

This license amendment request will extend surveillance testing intervals for selected instrumentation from every 18 months to each refueling interval (24-month fuel cycle plus 25% allowable tolerance). The licensees proposed replacing "at least once per 18 months" with "at least once each REFUELING INTERVAL" for the TSs described below:

(a) TS Table 4.3-1, "Reactor Protection System (RPS) Instrumentation Surveillance Requirements," Functional Units 3, 4 (a change to Notation 7), 5, 6, 7, 14, and 15;

(b) TS Table 4.3-2, "Safety Features Actuation System (SFAS) Instrumentation Surveillance Requirements," Functional Units 1d, 1e, 5a, 5b, and 3 (manual actuation switch only);

(c) TS Table 4.3-6, "Remote Shutdown Monitoring (RSM) Instrumentation Surveillance Requirements," Functional Units 2 through 6;

(d) TS Table 4.3-10, "Post Accident Monitoring (PAM) Instrumentation Surveillance Requirements," Functional Units 1 through 5, 6a, 10 through 13, 15 through 17, and 19;

(e) TS Table 4.3-11, "Steam and Feedwater Rupture Control System (SFRCS) Instrumentation Surveillance Requirements," Functional Units 1b and 2;

(f) TS Section 4.4.3, "Safety Valves and Pilot Operated Relief Valve - Operating," CHANNEL CALIBRATION surveillance interval for pressurizer PORV;

(g) TS Section 4.4.6.1.b, "Reactor Coolant System Leakage," CHANNEL CALIBRATION surveillance interval for containment sump level and flow monitoring system;

(h) TS Sections 4.7.1.2.1.d, 4.7.1.2.1.e, and 4.7.1.2.2, "Auxiliary Feedwater System," CHANNEL CALIBRATION surveillance interval for Auxiliary Feedwater Pump (AFP) Turbine, Steam Generator Level Control System, AFP Suction Pressure Interlock, and AFP Turbine Inlet Steam Pressure Interlock;

(i) TS Section 4.5.1.d, "Emergency Core Cooling System," verification that each core flood tank isolation valve opens automatically and is interlocked against closing whenever the RCS pressure exceeds 800 psig; and

(j) TS Section 4.5.2.d, "Emergency Core Cooling System," verification that the interlocks for the Decay Heat Isolation Valves, pressurizer heater and borated water storage tank (BWST) low-low level interlock trip perform their function. (Note: The TS text change to "REFUELING INTERVAL" was addressed in Amendment No. 216, issued simultaneously with this amendment.)

Evaluation

The proposed changes allow the continued application of TS 4.0.2. This TS permits surveillance intervals to be increased up to 25% on a non-routine basis (30 months) in accordance with GL 91-04. A paragraph was added (Amendment 213, dated February 10, 1997) to TS Bases 4.0.2, consistent with GL 91-04, that ensures that surveillances are performed in an operational mode consistent with safe plant operation. This TS Bases section already included clarification that the allowable tolerance not be used as a convenience to repeatedly schedule the performance of surveillances at the maximum time limit permitted by the TS.

The licensees performed a safety assessment for the proposed changes to the surveillance test intervals in accordance with the GL 91-04 guidance stated above. This assessment entailed reviewing the historical maintenance and surveillance test data at the bounding surveillance interval limit, performing an evaluation to ensure that a 24-month surveillance test interval would not invalidate any assumption in the plant licensing bases, and the determination that the effect on safety of the surveillance interval extension is small. Only the period since 1985 was reviewed. However, this is most representative of current plant operating conditions since many changes to DBNPS occurred after the loss of feedwater event in 1985. This period includes five refueling outages and four operating cycles of test data and results.

The licensees performed analyses of drift for all affected instrument loops in order to establish the effect of a 30-month (24 months plus 25% allowable tolerance) calibration frequency on instrument performance using the in-house procedure, "Drift Data Analysis Methodology and Assumptions," Revision 1, dated May 8, 1996. The analyses were performed to verify that the surveillance interval extensions have an insignificant effect on plant safety and would not invalidate any assumptions in the plant licensing basis. Statistically based drift values were determined for the instruments involved. In the evaluation of drift data time dependency and the determination of a projected 95/95 confidence interval for expected 30-month drift, the licensees used one of the following three approaches:

(a) Drift is time independent, and the 95/95 confidence interval calculated from historical as-found/as-left data applies to a 30-month calibration interval;

(b) Drift is time dependent, and each individual drift data point (calculated for calibration intervals less than 30 months) is extrapolated to 30 months using a linear method; and

(c) Drift is time dependent, and each individual drift data point is extrapolated to 30 months by using the square root method. If this method is used, a justification is provided.

The above drift evaluation approaches are consistent with GL 91-04 and are, therefore, acceptable to the staff.

In GL 91-04, the staff identified the issues discussed in the background section of this evaluation pertaining to increasing the interval of instrument surveillance and identified specific actions that licensees should take to address these issues. The staff has evaluated the licensees' submittals to verify that the licensees have addressed these issues and provided an acceptable basis for increasing the calibration interval for instruments that are used to perform safety functions. Based on the evaluation as described above, the staff concludes that the licensees have confirmed that safety limits and safety analysis assumptions will not be exceeded after the worst case drift is considered for the instruments indicated above for which surveillance intervals will be extended to 24 months.

Proposed Change

The licensees proposed the following setpoint changes required as a result of the DBNPS Instrument Drift Study and based on NUREG-1430 Revision 1:

(a) TS Table 3.3-12, "SFRCS instrumentation Trip Setpoints," Functional Unit 2, allowable value has been changed from ≥ 15.6" to ≥ 16.9" and the trip setpoint has been deleted. In addition, TS 3.3.2.2 has been modified to be consistent with this change;

(b) TS Table 3.3-4, "SFAS Instrumentation Trip Setpoints," Functional Unit d, allowable value has been changed from ≥ 1615.75 psig to ≥ 1576.2 psig and the trip setpoint has been deleted;

(c) TS Table 3.3-4, "SFAS Instrumentation Trip Setpoints," Functional Unit e, allowable value has been changed from ≥ 415.75 psig to ≥ 441.42 psig and the trip setpoint has been deleted;

(d) TS Table 3.3-4, "SFAS Instrumentation Trip Setpoints," Functional Unit a, for Interlock channels (Decay Heat Isolation Valve and Pressurizer Heater), allowable value has been changed from < 443 psig to < 328 psig and the trip setpoint has been deleted; and

(e) TS 3.3.2.1, "SFAS Instrumentation," has been modified consistent with (b), (c), and (d) above.

Evaluation

In their submittal, the licensees stated that their calculations to revise the allowable value (AV) of functional units to support the surveillance test interval extension are based on the methodology described in Framatome documents 32-1172392-02, "Reactor Protection System String Error Calculations," and 32-1257719-02, "Davis-Besse Unit 1 RPS Allowable Values Calculations." This methodology is consistent with the guidance of Regulatory Guide (RG) 1.105, Revision 2, which endorses Instrument Society of America (ISA) S67.04-1982, regarding accounting for uncertainties in setpoint calculations, except for Section 4.3.1(2)(a), which states that the accuracy of test equipment for measuring setpoints shall be included in the allowances between AV and the trip setpoints. The licensees determined that test instrument uncertainty is sufficiently small such that it would not affect the AV significantly. Based on this, the staff finds the licensees' setpoint methodology acceptable.

Proposed Change

The licensees proposed the following changes to TS Section 2.2, "Limiting Safety System Settings," based on the results of the revised Framatome RPS string error and setpoint allowable value calculations and the guidance of NUREG-1430, Revision 1:

(a) TS Table 2.2.1, "RPS Instrumentation Trip Setpoints," Functional Unit 2, High Flux, changed allowable value from $<$ 104.9% to $\leq$ 105.1%;

(b) TS Table 2.2.1, "RPS Instrumentation Trip Setpoints," Functional Unit 5, RC low pressure, changed allowable value from $>$ 1900 psig to $\geq$ 1900 psig;

(c) TS Table 2.2.1, "RPS Instrumentation Trip Setpoints," Functional Unit 6, RC high pressure, changed allowable value from $<$ 2355 psig to $\leq$ 2355 psig; and

(d) Deleted the column for trip setpoint.

Evaluation

These TS changes are based on an acceptable setpoint methodology as discussed in the previous section of this evaluation and are consistent with NUREG-1430, Revision 1. Therefore, the staff finds them acceptable.

Administrative Changes

The licensees proposed the following administrative changes to the TSs:

(a) All surveillances for instruments in TS Tables 4.3.1, 4.3.2, 4.3.3, 4.3.10, and 4.3.11, which were previously identified as "R" but were not changed to the new refueling interval, have been designated as "E" to indicate that the surveillance interval for these instruments remains at 18 months;

(b) TS Section 2.2.1 and its associated ACTION statement have been revised to replace "trip setpoint" with "allowable value";

(c) TS Sections 4.5.2.d.1.a and 4.5.2.d.1.b have been revised to replace "trip setpoint" with "allowable value." Also, the allowable value as given in TS Table 3.3-4 was included in place of the trip setpoint value;

(d) TS Sections 4.3.1.1.2, 4.3.1.1.3, 4.3.2.1.2, 4.3.2.1.3, 4.3.2.2.2, 4.3.2.2.3, and 4.3.3.5.2; Notation 7 in TS Table 4.3-1; and Notation 1 in TS Table 4.3-2 were changed to replace "18 months" with "Refueling Interval";

(e) TS Section 3.3.2.1 ACTION "a" has been revised to state, "...trip setpoint adjusted consistent with Table 3.3.4" instead of "...consistent with the trip setpoint value";

(f) TS Table 3.3-3 Notation "**" has been revised from 600 psig to 660 psig. ACTIONs "13a," "13b," and "14" have been revised from < 438 psig to < 328 psig, and have been modified to clarify that "indicated" reactor coolant pressure is the referenced parameter;

(g) A footnote to TS Tables 3.3-4 and 2.2-1 has been added to include the allowable value for CHANNEL FUNCTIONAL TEST;

(h) TS Section 3.3.2.2 ACTION "a" has been revised to state, "...trip setpoint adjusted consistent with Table 3.3-12" instead of "...consistent with the trip setpoint value";

(i) TS Table 4.3-10, Instrument 6, "Containment Vessel Post-Accident Radiation," has been revised to list two separate instruments, (6a) Containment High Range Radiation and (6b) Containment Wide Range Noble Gas. The containment high range radiation monitor will have the surveillance frequency of 24 months while the containment wide range noble gas monitor will continue to be on an 18-month surveillance interval;

(j) TS Basis Sections 3/4.3.1, 3/4.3.2, 3/4.5.2, and 3/4.5.3 have been revised to include the bases for the changes in surveillance frequency and instrument setpoint/allowable value discussed previously in this evaluation; and

(k) TS Basis Section 2.2.1 has been revised to change trip setpoints to allowable values and add the new allowable value for high flux.

Evaluation

All of the above changes are administrative in nature as they reflect TS changes for new setpoint/allowable values, action statements, and surveillance intervals consistent with the 24-month refueling outage surveillance interval as discussed above. Therefore, the changes are acceptable.

Evaluation of Radiological Consequences

TS Amendment No. 181 was issued to DBNPS on November 19, 1993. This change allowed the licensees to utilize fuel with up to 5% enrichment, and extended fuel assembly average discharge burnup to 60,000 megawatt days per metric ton of uranium (MWD/MTU), so that extended fuel cycles would be possible. This license amendment request to revise certain TS surveillance requirements for conversion to a 24-month fuel cycle will not exceed the allowable 60,000 MWD/MTU fuel burnup limit. The licensees identified in the request that three design basis accidents impacted by the proposed changes are the 0.04 square foot small break loss-of-coolant accident, the letdown line break accident, and the steam generator tube rupture accident. The licensees concluded, and the staff agrees, that these three accidents do not involve any core damage and that only the radiological consequences resulting from the letdown line break accident is affected by the proposed changes. The licensees reanalyzed the radiological consequences resulting from the letdown line break accident and showed that the recalculated doses are higher than those previously reported but they are still within a small fraction of dose criteria specified in 10 CFR Part 100. The staff reviewed the radiological consequence analysis summary submitted by the licensees for the letdown line break accident and finds it to be acceptable.

Concluding Remarks

Based on its review, the staff concludes that the proposed DBNPS TS modifications to extend surveillance intervals for certain safety-related instrumentation components are consistent with the guidance in GL 91-04 and NUREG-1430, Revision 1, in that the licensees have demonstrated that the effect on safety of extending the surveillance interval to 24 months is negligible and the system will continue to perform within assumed limits during the longer surveillance interval. The staff finds that the setpoint methodology, drift analysis and changes proposed for the TS AVs and setpoints are in accordance with the guidance of RG 1.105 and are consistent with design basis accident analyses. Therefore, the staff finds the proposed TS modifications for a 24-month surveillance interval to be acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued proposed findings that the amendment involves no significant hazards consideration, and there has been no public comment on such findings (62 FR 2194, 62 FR 11498, 62 FR 30645). The supplemental information submitted by the licensees did not affect the proposed findings. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

Principal Contributor: H. Garg

Date: December 2, 1997