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Amdt 158 to DPR-52
See Correction Letter of 12-13-88

Docket Nos. 50-259/260/296

November 22, 1988

Mr. Oliver D. Kingsley, Jr.
Senior Vice President, Nuclear Group
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. Kingsley:

SUBJECT: BROWNS FERRY TECHNICAL SPECIFICATIONS CHANGE TO REFLECT REVISED ASME SECTION XI PUMP AND VALVE PROGRAM (TAC 00250/251/252) (TS 235)

The Commission has issued the enclosed Amendment Nos. 159 155 and 130 to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3, respectively. These amendments are in response to your application dated October 27, 1987. The proposed amendment will add an ASME Section XI pump and valve program definition and change applicable testing frequencies to reference this definition.

A copy of the Safey Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Original Signed by
Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Special Projects

Enclosures:
1. Amendment No. 159 to License No. DPR-33
2. Amendment No. 155 to License No. DPR-52
3. Amendment No. 130 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

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Correction Letter of 12-13-88

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 155
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 27, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 155 , 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 its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne Black

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 22, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 155

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf pages* are provided to maintain document completeness.

REMOVE	INSERT
1.0-11	1.0-11*
1.0-12	1.0-12
3.3/4.3-11	3.3/4.3-11*
3.3/4.3-12	3.3/4.3-12
3.3/4.4-1	3.3/4.4-1
3.3/4.4-2	3.3/4.4-2*
3.5/4.5-1	3.5/4.5-1
3.5/4.5-2	3.5/4.5-2
3.5/4.5-3	3.5/4.5-3*
3.5/4.5-4	3.5/4.5-4
3.5/4.5-7	3.5/4.5-7
3.5/4.5-8	3.5/4.5-8*
3.5/4.5-9	3.5/4.5-9
3.5/4.5-10	3.5/4.5-10*
3.5.4.5-12	3.5/4.5-12*
3.5/4.5-13	3.5/4.5-13
3.5/4.5-14	3.5/4.5-14
3.5/4.5-15	3.5/4.5-15*
3.5/4.5-34	3.5/4.5-32*
3.5/4.5-35	3.5/4.5-33
3.6/4.6-9	3.6/4.6-9*
3.6/4.6-10	3.6/4.6-10
3.6/4.6-30	3.6/4.6-30
3.6/4.6-31	3/6/4.6-31*
3.7/4.7-9	3.7/4.7-9*
3.7/4.7-10	3.7/4.7-10
3.7/4.7-11	3.7/4.7-11
3.7/4.7-12	3.7/4.7-12
3.7.4.7-17	3.7/4.7-17
3.7.4.7-18	3.7/4.7-18
3.7/4.7-21	3.7/4.7-21*
3.7/4.7-22	3.7/4.7-22
3.7/4.7-49	3.7/4.7-49*
3.7/4.7-50	3.7/4.7-50

MM. Surveillance Requirements for ASME Section XI Pump and Valve Program - Surveillance requirements for Inservice Testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

1. Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these technical specifications:

ASME Boiler and Pressure Vessel 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

3. The provisions of Specification 1.0.LL are applicable to the above required frequencies for performing inservice testing activities.

4. Performance of the above inservice testing activities shall be in addition to other specified surveillance requirements.

5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any technical specification.

Table 1.1

SURVEILLANCE FREQUENCY NOTATION

NOTATION		FREQUENCY
S	(Shift)	At least once per 12 hours.
D	(Daily)	At least once per normal calendar 24 hour day (midnight to midnight).
W	(Weekly)	At least once per 7 days.
M	(Monthly)	At least once per 31 days.
Q	(Quarterly)	At least once per 3 months or 92 days.
SA	(Semi-Annually)	At least once per 6 months or 184 days.
Y	(Yearly)	At least once per year or 366 days.
R	(Refueling)	At least once per operating cycle.
S/U	(Start-Up)	Prior to each reactor startup.
N.A.		Not applicable.
P	(Prior)	Completed prior to each release.

LIMITING CONDITIONS FOR OPERATION

3.3.C. Scram Insertion Times

2. The average of the scram insertion times for the three fastest OPERABLE control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

a. The maximum scram insertion time for 90% insertion of any OPERABLE control rod shall not exceed 7.00 seconds.

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% k. If this limit is exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken as appropriate.

SURVEILLANCE REQUIREMENTS

4.3.C. Scram Insertion Times

2. At 16-week intervals, 10% of the OPERABLE control rod drives shall be scram-timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

D. Reactivity Anomalies

During the STARTUP test program and STARTUP following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.E. If Specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN CONDITION within 24 hours.	4.3.E. Surveillance requirements are as specified in 4.3.C and .D above.
F. Scram Discharge Volume (SDV)	F. Scram Discharge Volume (SDV)
1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.	1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.
	1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.
2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.	2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.
3. If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.	3. No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the operating status of the Standby Liquid Control System.

Objective

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without the use of control rods.

Specification

A. Normal System Availability

1. Except as specified in 3.4.B.1, the Standby Liquid Control System shall be OPERABLE at all times when there is fuel in the reactor vessel and the reactor is not in a shutdown condition with Specification 3.3.A.1 satisfied.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective

To verify the operability of the Standby Liquid Control System.

Specification

A. Normal System Availability

The operability of the Standby Liquid Control System shall be verified by the performance of the following tests:

1. Verify pump OPERABILITY in accordance with Specification 1.0.MM.

2. At least once during each operating cycle:

a. Check that the setting of the system relief valves is 1,425 ± 75 psig.

b. Manually initiate the system, except explosive valves. Visually verify flow by pumping boron solution through the recirculation path and back to the Standby Liquid Control Solution Tank. After pumping boron solution, the system shall be flushed with demineralized water. Verify minimum

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

4.4.A Normal System Applicability

4.4.A.2.b. (Cont'd)

pump flow rate of 39 gpm against a system head of 1275 psig by pumping demineralized water from the Standby Liquid Control Test Tank.

c. Manually initiate one of the Standby Liquid Control System loops and pump demineralized water into the reactor vessel.

This test checks explosion of the charge associated with the tested loop, proper operation of the valves, and pump operability. Replacement charges shall be selected such that the age of charge in service shall not exceed five years from the manufacturer's assembly date.

d. Both systems, including both explosive valves, shall be tested in the course of two operating cycles.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and containment cooling systems.

Objective

To assure the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. The CSS shall be OPERABLE:

(1) PRIOR TO STARTUP from a COLD CONDITION, or

(2) when there is irradiated fuel in the vessel and when the reactor vessel pressure is greater than atmospheric pressure, except as specified in Specification 3.5.A.2.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the surveillance requirements of the core and containment cooling systems when the corresponding limiting condition for operation is in effect.

Objective

To verify the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. Core Spray System Testing.

	Item	Frequency
a.	Simulated Automatic Actuation test	Once/ Operating Cycle
b.	Pump Operability	Per Specification 1.0.MM
c.	Motor Operated Valve Operability	Per Specification 1.0.MM
d.	System flow rate: Each loop shall deliver at least 6250 gpm against a system head corresponding to a	Once/3 months

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.A Core Spray System (CSS)	4.5.A Core Spray System (CSS)
	4.5.A.1.d (Cont'd) 105 psi differential pressure between the reactor vessel and the primary containment.
	e. Check Valve — Per Specification 1.0.MM.
2. If one CSS loop is inoperable, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS loop and the RHR system (LPCI mode) and the diesel generators are OPERABLE.	2. When it is determined that one core spray loop is inoperable, at a time when operability is required, the other core spray loop and the RHRS (LPCI mode) shall be demonstrated to be OPERABLE immediately. The OPERABLE core spray loop shall be demonstrated to be OPERABLE daily thereafter.
3. If Specification 3.5.A.1 or Specification 3.5.A.2 cannot be met, the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.	
4. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one core spray loop with one OPERABLE pump and associated diesel generator shall be OPERABLE, except with the reactor vessel head removed as specified in 3.5.A.5 or PRIOR TO STARTUP as specified in 3.5.A.1.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.A Core Spray System (CSS) 5. When irradiated fuel is in the reactor vessel and the reactor vessel head is removed, core spray is not required provided work is not in progress which has the potential to drain the vessel, provided the fuel pool gates are open and the fuel pool is maintained above the low level alarm point, and provided one RHRSW pump and associated valves supplying the standby coolant supply are OPERABLE.	

LIMITING CONDITIONS FOR OPERATION

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

1. The RHRS shall be OPERABLE:

   (1) PRIOR TO STARTUP from a COLD CONDITION, or

   (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

2. With the reactor vessel pressure less than 105 psig, the RHR may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water, provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.

SURVEILLANCE REQUIREMENTS

4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

1. a. Simulated Automatic Actuation Test — Once/Operating Cycle

   b. Pump Operability — Per Specification 1.0.MM

   c. Motor Operated valve operability — Per Specification 1.0.MM

   d. Pump Flow Rate — Once/3 months

   e. Testable Check Valve — Per Specification 1.0.MM

   Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12,000 gpm against an indicated system pressure of 250 psig.

2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.B <u>Residual Heat Removal System (RHRS)</u> (LPCI and Containment Cooling)	4.5.B. <u>Residual Heat Removal System (RHRS)</u> (LPCI and Containment Cooling)
8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be shutdown and placed in the COLD SHUTDOWN CONDITION within 24 hours.	8. No additional surveillance required.
9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE.	9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.
10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.	10. No additional surveillance required.
11. When there is irradiated fuel in the reactor and the reactor vessel pressure is greater than atmospheric, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)	11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE monthly when the cross-connect capability is required.

LIMITING CONDITIONS FOR OPERATION

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

12. If three RHR pumps or associated heat exchangers located on the unit cross-connection in the adjacent units are INOPERABLE for any reason (including valve inoperability, pipe break, etc.), the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pump and associated diesel generator are OPERABLE.

13. If RHR cross-connection flow or heat removal capability is lost, the unit may remain in operation for a period not to exceed 10 days unless such capability is restored.

14. All recirculation pump discharge valves shall be OPERABLE prior to reactor startup (or closed if permitted elsewhere in these specifications).

SURVEILLANCE REQUIREMENTS

4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

12. When it is determined that three RHR pumps or associated heat exchangers located on the unit cross-connection in the adjacent units are INOPERABLE at a time when operability is required, the remaining RHR pump and associated heat exchanger on the unit cross-connection shall be demonstrated to be OPERABLE immediately and every 15 days thereafter until the INOPERABLE pump and associated heat exchanger are returned to normal service.

13. No additional surveillance required.

14. All recirculation pump discharge valves shall be tested for operability during any period of reactor Cold Shutdown exceeding 48 hours, if operability tests have not been performed during the preceding 31 days.

LIMITING CONDITIONS FOR OPERATION

3.5.C <u>RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)</u>

1. PRIOR TO STARTUP from a COLD CONDITION, 9 RHRSW pumps must be OPERABLE, with 7 pumps (including one of pumps D1, D2, B2 or B1) assigned to RHRSW service and 2 automatically starting pumps assigned to EECW service.

SURVEILLANCE REQUIREMENTS

4.5.C <u>RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)</u>

1. a. Each of the RHRSW pumps normally assigned to automatic service on the EECW headers will be tested automatically each time the diesel generators are tested. Each of the RHRSW pumps and all associated essential control valves for the EECW headers and RHR heat exchanger headers shall be demonstrated to be OPERABLE in accordance with Specification 1.0.MM.

   b. Annually each RHRSW pump shall be flow-rate tested. To be considered OPERABLE, each pump shall pump at least 4500 gpm through its normally assigned flow path.

LIMITING CONDITIONS FOR OPERATION

3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Cont'd)

2. During reactor power operation, RHRSW pumps must be OPERABLE and assigned to service as indicated in Table 3.5-1 for the specified time limits.

3. During unit 2 power operation, any two RHRSW pumps (D1, D2, B1, and B2) normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection must be OPERABLE except as specified in 3.5.C.4 and 3.5.C.5 below.

SURVEILLANCE REQUIREMENTS

4.5.C. RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Cont'd)

2. a. If no more than two RHRSW pumps are INOPERABLE, increased surveillance is not required.

   b. When three RHRSW pumps are INOPERABLE, the remaining pumps and associated essential control valves shall be operated daily.

   c. When four RHRSW pumps are INOPERABLE, the remaining pumps and associated essential control valves shall be operated daily.

3. Routine surveillance for these pumps is specified in 4.5.C.1.

LIMITING CONDITIONS FOR OPERATION

3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)

4. Three of the D1, D2, B1, B2 RHRSW pumps assigned to the RHR heat exchanger supplying the standby coolant supply connection may be INOPERABLE for a period not to exceed 30 days provided the OPERABLE pump is aligned to supply the RHR heat exchanger header and the associated diesel generator and essential control valves are OPERABLE.

5. The standby coolant supply capability may be INOPERABLE for a period not to exceed 10 days.

6. If Specifications 3.5.C.2 through 3.5.C.5 are not met, an orderly shutdown shall be initiated and the unit placed in the Cold Shutdown condition within 24 hours.

7. There shall be at least 2 RHRSW pumps, associated with the selected RHR pumps, aligned for RHR heat exchanger service for each reactor vessel containing irradiated fuel.

SURVEILLANCE REQUIREMENTS

4.5.C. RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Cont'd)

4. When it is determined that three of the RHRSW pumps supplying standby coolant are INOPERABLE at a time when operability is required, the OPERABLE RHRSW pump on the same header and the RHR heat exchanger header and associated essential control valves shall be demonstrated to be OPERABLE immediately and every 15 days thereafter.

LIMITING CONDITIONS FOR OPERATION

3.5.D. Equipment Area Coolers

1. The equipment area cooler associated with each RHR pump and the equipment area cooler associated with each set of core spray pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE.

2. When an equipment area cooler is not OPERABLE, the pump(s) served by that cooler must be considered inoperable for technical specification purposes.

E. High Pressure Coolant Injection System (HPCIS)

1. The HPCI system shall be OPERABLE:

(1) PRIOR TO STARTUP from a COLD CONDITION, or

(2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 122 psig, except as specified in Specification 3.5.E.2.

SURVEILLANCE REQUIREMENTS

4.5.D. Equipment Area Coolers

1. Each equipment area cooler is operated in conjunction with the equipment served by that particular cooler; therefore, the equipment area coolers are tested at the same frequency as the pumps which they serve.

E. High Pressure Coolant Injection System (HPCIS)

1. HPCI Subsystem testing shall be performed as follows:

a.	Simulated Automatic Actuation Test	Once/ operating cycle
b.	Pump Operability	Per Specification 1.0.MM
c.	Motor Operated Valve Operability	Per Specification 1.0.MM
d.	Flow Rate at normal reactor vessel operating pressure	Once/3 months

LIMITING CONDITIONS FOR OPERATION

3.5.E High Pressure Coolant Injection System (HPCIS)

2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS (LPCI), and RCICS are OPERABLE.

3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 122 psig or less within 24 hours.

F. Reactor Core Isolation Cooling System (RCICS)

1. The RCICS shall be OPERABLE:

(1) PRIOR TO STARTUP from a COLD CONDITION, or

(2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 122 psig, except as specified in 3.5.F.2.

SURVEILLANCE REQUIREMENTS

4.5.E. High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

e.	Flow Rate at 150 psig	Once/ operating cycle

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

2. When it is determined that the HPCIS is inoperable, the ADS actuation logic, the RCICS, the RHRS (LPCI), and the CSS shall be demonstrated to be OPERABLE immediately. The RCICS and ADS logic shall be demonstrated to be OPERABLE daily thereafter.

F. Reactor Core Isolation Cooling System (RCICS)

1. RCIC Subsystem testing shall be performed as follows:

a.	Simulated Automatic Actuation Test	Once/ operating cycle
b.	Pump Operability	Per Specification 1.0.MM
c.	Motor-Operated Valve Operability	Per Specification 1.0.MM

LIMITING CONDITIONS FOR OPERATION

3.5.F. Reactor Core Isolation Cooling System (RCICS)

2. If the RCICS is INOPERABLE, the reactor may remain in operation for a period not to exceed 7 days if the HPCIS is OPERABLE during such time.

3. If Specifications 3.5.F.1 or 3.5.F.2 are not met, an orderly shutdown shall be initiated and the reactor shall be depressurized to less than 122 psig within 24 hours.

G. Automatic Depressurization System (ADS)

1. Four of the six valves of the Automatic Depressurization System shall be OPERABLE:

   (1) prior to a STARTUP from a Cold Condition, or,

SURVEILLANCE REQUIREMENTS

4.5.F. Reactor Core Isolation Cooling System (RCICS)

4.5.F.1 (Cont'd)

d.	Flow Rate at normal reactor vessel operating pressure	Once/3 months
e.	Flow Rate at 150 psig	Once/ operating cycle

The RCIC pump shall deliver at least 600 gpm during each flow test.

2. When it is determined that the RCICS is INOPERABLE, the HPCIS shall be demonstrated to be OPERABLE immediately.

G. Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:

   a. A simulated automatic actuation test shall be performed prior to STARTUP after each

3.5.M. References

1. **Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant** Unit 2, **NEDO - 24088-1 and Addenda.**

2. **"BWR Transient Analysis Model Utilizing the RETRAN Program," TVA-TR81-01-A.**

3. **Generic Reload Fuel Application, Licensing Topical Report, NEDE - 24011-P-A and Addenda.**

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment, and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested in accordance with Specification 1.0.MM to assure their operability. A simulated automatic actuation test once each cycle combined with testing of pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventive maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, cause the outage, then the demonstration of operability should be thorough enough to assure that a generic problem does not exist. For example, if an out-of-service period was caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

Whenever a CSCS system or loop is made inoperable because of a required test or calibration, the other CSCS systems or loops that are required to be operable shall be considered operable if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Redundant operable components are subjected to increased testing during equipment out-of-service times. This adds further conservatism and increases assurance that adequate cooling is available should the need arise.

Maximum Average Planar LHGR, LHGR, and MCPR

The MAPLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

LIMITING CONDITIONS FOR OPERATION

3.6.C. Coolant Leakage

1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

   b. Anytime the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below.

   c. During the first 24 hours in the RUN mode following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

SURVEILLANCE REQUIREMENTS

4.6.C. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.

LIMITING CONDITIONS FOR OPERATION

3.6.C Coolant Leakage

2. Both the sump and air sampling systems shall be OPERABLE during REACTOR POWER OPERATION. From and after the date that one of these systems is made or found to be inoperable for any reason, reactor power operation is permissible only during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

   The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be shutdown in the COLD SHUTDOWN CONDITION within 24 hours.

3.6.D. Relief Valves

1. When more than one relief valves are known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.C Coolant Leakage

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

4.6.D. Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

3.6.C/4.6.C (Cont'd)

reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the unit should be shut down to allow further investigation and corrective action.

The 2 gpm limit for coolant leakage rate increases over any 24-hour period is a limit specified by NRC (Reference 2). This limit applies only during the RUN mode to avoid being penalized for the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pump is 50 gpm and the capacity of the drywell equipment sump pump is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

REFERENCE

1. Nuclear System Leakage Rate Limits (BFNP FSAR Subsection 4.10)
2. Safety Evaluation Report (SER) on IE Bulletin 82-03

3.6.D/4.6.D Relief Valves

To meet the safety basis, 13 relief valves have been installed on the unit with a total capacity of 84.1 percent of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves OPERABLE, results in adequate margin to the code allowable overpressure limit of 1,375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1,375 psig.

Experience in relief valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief valves are benchtested every second operating cycle to ensure that their setpoints are within the ± 1 percent tolerance. The relief valves are tested in place in accordance with Specification 1.0.MM to establish that they will open and pass steam.

3.6.D/4.6.D (Cont'd)

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

REFERENCES

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)

2. Amendment 22 in response to AEC Question 4.2 of December 6, 1971.

3. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)

4. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilleland to F. E. Kruesi, August 29, 1973

5. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda

3.6.E/4.6.E Jet Pumps

Failure of a jet pump nozzle assembly holddown mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Also, failure of the diffuser would eliminate the capability to reflood the core to two-thirds height level following a recirculation line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5 percent, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10 percent, riser and nozzle assembly integrity has been verified.

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10 percent or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the unit shut down for repairs. If the potential blowdown flow area is increased, the system

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

j. Continuous Leak Rate Monitor

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

k. Drywell and Torus Surfaces

The interior surfaces of the drywell and torus above the level one foot below the normal water line and outside surfaces of the torus below the water line shall be visually inspected each operating cycle for deterioration and any signs of structural damage with particular attention to piping connections and supports and for signs of distress or displacement.

LIMITING CONDITIONS FOR OPERATION

3.7.A Primary Containment

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when primary containment integrity is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be 0.5 psid.

b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate primary containment integrity.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c., below.

b. One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.

SURVEILLANCE REQUIREMENTS

4.7.A Primary Containment

3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation each three months.

b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.

b. When it is determined that two vacuum breakers are inoperable for opening at a time when operability is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

LIMITING CONDITIONS FOR OPERATION

3.7.A Primary Containment

3.7.A.4 (Cont'd)

c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.

d. If Specifications 3.7.A.4.a, .b, or .c cannot be met, the unit shall be placed in a COLD SHUTDOWN CONDITION in an orderly manner within 24 hours.

5. Oxygen Concentration

a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100/psig, except as specified in 3.7.A.5.b.

b. Within the 24-hour period subsequent to placing the reactor in the RUN MODE following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a COLD SHUTDOWN CONDITION within 24 hours.

d. If Specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN CONDITION within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.A Primary Containment

4.7.A.4 (Cont'd)

c. Each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches in accordance with Specification 1.0.MM.

d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

5. Oxygen Concentration

a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

LIMITING CONDITIONS FOR OPERATION

3.7.A. Primary Containment

6. Drywell-Suppression Chamber Differential Pressure

a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.1 psid except as specified in (1) and (2) below:

(1) This differential shall be established within 24 hours of achieving operating temperature and pressure. The differential pressure may be reduced to less than 1.1 psid 24 hours prior to a scheduled shutdown.

(2) This differential may be decreased to less than 1.1 psid for a maximum of four hours during required operability testing of the HPCI system, RCIC system and the drywell-pressure suppression chamber vacuum breakers.

b. If the differential pressure of Specification 3.7.A.6.a cannot be maintained and the differential pressure cannot be restored within the subsequent six-hour period, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

6. Drywell-Suppression Chamber Differential Pressure

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.C. Secondary Containment 4. If refueling zone secondary containment cannot be maintained, the following conditions shall be met: a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited. b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones.	
D. Primary Containment Isolation Valves 1. When primary containment integrity is required, all isolation valves listed in Table 3.7.A and all reactor coolant system instrument line flow check valves shall be OPERABLE except as specified in 3.7.D.2.	D. Primary Containment Isolation Valves 1. The primary containment isolation valves surveillance shall be performed as follows: a. At least once per operating cycle, the OPERABLE isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation, and in accordance with Specification 1.0.MM, tested for closure times.

LIMITING CONDITIONS FOR OPERATION

3.7.D. Primary Containment Isolation Valves

2. In the event any isolation valve specified in Table 3.7.A becomes inoperable, reactor power operation may continue provided at least one valve in each line having an inoperable valve is OPERABLE and within 4 hours either:

   a. The inoperable valve is restored to OPERABLE status, or

   b. Each affected line is isolated by use of at least one deactivated containment isolation valve secured in the isolated position.

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.D. Primary Containment Isolation Valves

4.7.D.1 (Cont'd)

   b. In accordance with Specification 1.0.MM, all normally open power operated isolation valves shall be functionally tested.

   c. (Deleted)

   d. At least once per operating cycle, the operability of the reactor coolant system instrument line flow check valves shall be verified.

2. Whenever an isolation valve listed in Table 3.7.A is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

LIMITING CONDITIONS FOR OPERATION

3.7.F. Primary Containment Purge System

1. The primary containment shall be normally vented and purged through the primary containment purge system. The standby gas treatment system may be used when primary containment purge system is INOPERABLE.

2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

   b. The results of laboratory carbon sample analysis shall show ≥85% radioactive methyl iodide removal when tested in accordance with ASTM D3803 (130°C 95% R.H.).

   c. System flow rate shall be shown to be within ±10% of design flow when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS

4.7.F. Primary Containment Purge System

1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8.5 inches of water at system design flow rate (± 10%).

2. a. The tests and sample analysis of Specification 3.7.F.2 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first or after 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

   b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

   c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

LIMITING CONDITIONS FOR OPERATION

3.7.G. Containment Atmosphere Dilution System (CAD)

1. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE with:

   a. Two independent systems capable of supplying nitrogen to the drywell and torus.

   b. A minimum supply of 2,500 gallons of liquid nitrogen per system.

2. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE whenever the reactor is in the RUN MODE.

3. If one system is inoperable, the reactor may remain in operation for a period of 30 days provided all active components in the other system are OPERABLE.

SURVEILLANCE REQUIREMENTS

4.7.G. Containment Atmosphere Dilution System (CAD)

1. System Operability

   a. Cycle each solenoid operated air/nitrogen valve through at least one complete cycle of full travel in accordance with Specification 1.0.MM, and at least once per month verify that each manual valve in the flow path is open.

   b. Verify that the CAD System contains a minimum supply of 2,500 gallons of liquid nitrogen twice per week.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repaired and test repeated.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other systems must be tested daily. This substantiates the availability of the operable systems and thus reactor operation and refueling operation can continue for a limited period of time.

3.7.D/4.7.D Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a LOCA.

Group 1 - Process lines are isolated by reactor vessel low water level (378") in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in Group 1, except the reactor water sample line valves, are also closed when process instrumentation detects excessive main steam line flow, high radiation, low pressure, or main steam space high temperature. The reactor water sample line valves isolate only on reactor low water level at 378" or main steam line high radiation.

Group 2 - Isolation valves are closed by reactor vessel low water level (538") or high drywell pressure. The Group 2 isolation signal also "isolates" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the Group 2 isolation signal by a transient or spurious signal.

Group 3 - Process lines are normally in use, and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from nonsafety related causes. To protect the reactor from a possible pipe break in the system, isolation is provided by high temperature in the cleanup system area or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, a low-level isolation is provided.

Groups 4 and 5 - Process lines are designed to remain operable and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of Groups 4 and 5 process lines are therefore indicative of a condition which would render them inoperable.

Group 6 - Lines are connected to the primary containment but not directly to the reactor vessel. These valves are isolated on reactor low water level (538"), high drywell pressure, or reactor building ventilation high radiation which would indicate a possible accident and necessitate primary containment isolation.

Group 7 - Process lines are closed only on the respective turbine steam supply valve not fully closed. This assures that the valves are not open when HPCI or RCIC action is required.

Group 8 - Line (traveling in-core probe) is isolated on high drywell pressure or reactor low water level (538"). This is to assure that this line does not provide a leakage path when containment pressure or reactor water level indicates a possible accident condition.

The maximum closure time for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

In satisfying this design intent, an additional margin has been included in specifying maximum closure times. This margin permits identification of degraded valve performance prior to exceeding the design closure times.

In order to assure that the doses that may result from a steam line break do not exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

These valves are highly reliable, have low service requirements, and most are normally closed. The initiating sensors and associated trip logic are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability in accordance with Specification 1.0.MM results in a greater assurance that the valve will be operable when needed.

The main steam line isolation valves are functionally tested per Specification 1.0.MM to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25-inch restricting orifice inside the primary containment and an excess flow check valve outside the primary containment.

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

SAFETY EVALUATION BY THE OFFICE OF SPECIAL PROJECTS

SUPPORTING AMENDMENT NO. 159 [redacted] TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 155 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3

DOCKET NOS. 50-259, 50-260 AND 50-296

1.0 INTRODUCTION

By letter dated October 27, 1987, Tennessee Valley Authority (The licensee) requested a change to the Browns Ferry Nuclear Plant, Unit 1, 2 and 3 Technical Specifications to reference the ASME Section XI Pump and Valve Program definition and to revise applicable surveillance tests frequencies consistent with the specified ASME program. The existing monthly pump and valve surveillance test frequencies were based upon earlier editions of the ASME code which have been revised in a more recent edition that specifies quarterly testing requirements.

2.0 EVALUATION

Testing in accordance with ASME Section XI is required by 10 CFR 50.55a(g). Adding an ASME Section XI Pump and Valve Program definition to the Technical Specifications and referencing this definition throughout applicable pump and valve surveillance test requirements in the Technical Specifications provides consistency between the requirements of the ASME Section XI Pump and Valve Program and the requirements of the Technical Specification.

The substantive changes involve the frequency of testing ASME Section XI pumps, motor operated valves, solenoid operated air/nitrogen valves, the closure time testing of main steam isolation valves, the exercising of vacuum breakers, and the operability testing of the scram discharge volume drain and vent valves. The reduction in the frequency of testing began with the 1979 edition/Winter 1979 addenda of the Code because monthly testing caused unnecessary wear to safety-related components and increased the probability of system misalignments resulting from returning the system to their normal configurations following testing.

The licensee, in a letter dated December 23, 1986, established the 1980 edition/Winter 1980 addenda as its ASME Section XI code of record per 50.55a(g)(4). The proposed changes reflect the current ASME Section XI Pump and Valve Program. Changing the frequency requirements to reflect the ASME Section XI Pump and Valve Program does not affect the intent of the technical

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specifications which is to perform pump and valve operability testing in accordance with the requirements of 10 CFR 50.55a.

The licensee's request to modify the technical specifications to include an ASME Section XI Pump and Valve Program definition and to reference this definition throughout applicable pump and valve surveillance test requirements, thereby decreasing the frequency of testing, is acceptable. The change is consistent with the requirements specified in 10 CFR 50.55a and is similar to that which has been reviewed and approved in the technical specifications for a number of recently licensed reactors, e.g., Hope Creek and Limerick 1.

Based on the above evaluation, the staff finds the proposed changes to the Technical Specification are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

The amendments involve a change to a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to surveillance requirements. The staff has determined that the amendments involve 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet 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 nor environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (53 FR 2325) on January 27, 1988 and consulted with the State of Alabama. No public comments were received and the State of Alabama did not have any comments.

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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: David Moran

Dated: November 22, 1988