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

August 13, 1984

Docket Nos. 50-259/260/296

Posted
Amdt. 102
to DPR-52
(See CORRECTION
Letter of 9-19-84)

Mr. Hugh G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

The Commission has issued the enclosed Amendment Nos. 108, 102 and 75 to Facility Operating License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Unit Nos. 1, 2 and 3. These amendments are in response to your application dated April 9, 1984 (TVA BFNP TS 197).

The amendments change the Technical Specifications to allow extension of surveillance intervals, clarify various instrumentation requirements, make an editiorial correction and make corrections necessitated by plant modifications and changes to regulations.

A copy of the Safety Evaluation is also enclosed.

Sincerely,

Richard J. Clark

Richard J. Clark, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 108 to License No. DPR-33
2. Amendment No. 102 to License No. DPR-52
3. Amendment No. 75 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

Mr. Hugh G. Parris
Tennessee Valley Authority
Browns Ferry Nuclear Plant, Units 1, 2 and 3

cc:

H. S. Sanger, Jr., Esquire
General Counsel
Tennessee Valley Authority
400 Commerce Avenue
E 11B 330
Knoxville, Tennessee 37902

Mr. Ron Rogers
Tennessee Valley Authority
400 Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Mr. Charles R. Christopher
Chairman, Limestone County Commission
Post Office Box 188
Athens, Alabama 35611

Ira L. Meyers, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36130

Mr. H. N. Culver
249A HBD
400 Commerce Avenue
Tennessee Valley Authority
Knoxville, Tennessee 37902

James P. O'Reilly
Regional Administrator
Region II Office
U. S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

U. S. Environmental Protection
Agency
Region IV Office
Regional Radiation Representative
345 Courtland Street, N. W.
Atlanta, Georgia 30308

Resident Inspector
U. S. Nuclear Regulatory Commission
Route 2, Box 311
Athens, Alabama 35611

Mr. Donald L. Williams, Jr.
Tennessee Valley Authority
400 West Summit Hill Drive, W10B85
Knoxville, Tennessee 37902

George Jones
Tennessee Valley Authority
Post Office Box 2000
Decatur, Alabama 35602

Mr. Oliver Havens
U. S. Nuclear Regulatory Commission
Reactor Training Center
Osborne Office Center, Suite 200
Chattanooga, Tennessee 37411

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. 102
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 April 9, 1984, 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. 102 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 13, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 102

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

1. Remove the following pages and replace with identically numbered pages.

   7, 55, 58, 61, 70, 78, 112, 231, 232, 251a, 255, 266, 277

2. The marginal lines on these pages denote the area being changed.

1.0 DEFINITIONS (Cont'd)

10. Logic - A logic is an arrangement of relays, contacts, and other components that produces a decision output.

(a) Initiating - A logic that receive signals from channels and produces decision outputs to the actuation logic.

(b) Actuation - A logic that receives signals (either from initiation logic or channels) and produces decision outputs to accomplish a protective action.

W. Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or components to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).

X. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed.

Y. Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents.

Z. Reportable Event - A reportable event shall be any of those conditions specified in section 50.73 to 10 CFR Part 50.

Surveillance Interval - Each Surveillance Requirement shall be performed within the specified time interval with:

1. A maximum allowable extension not to exceed 25% of the surveillance interval, but:

2. The combined time interval for any 3 consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

Amendment No. ~~38~~, ~~49~~, ~~82~~, 102

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level (6)	≥ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation (Groups 2, 3, and 6) c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure	100 ± 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D, SW #1)	≥ 470" above vessel zero	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PS-64-56A-D)	≤ 2.5 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
2	Instrument Channel - High Radiation Main Steam Line Tunnel (6)	3 times normal rated full power background (13)	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Low Pressure Main Steam Line	≥ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2 (3)	Instrument Channel - High Flow Main Steam Line	≤ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation

Amendment No. ~~82~~, 102

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Group 2 (Initiating) Logic	N/A	A or (B and E)	1. Refer to Table 3.7.A for list of valves.
1	Group 2 (RHR Isolation-Actuation) Logic	N/A	D	
1	Group 8 (Tip-Actuation) Logic	N/A	J	
1	Group 2 (Drywell Sump Drains-Actuation) Logic	N/A	K	
1	Group 2 (Reactor Building & Refueling Floor, and Dry-well Vent and Purge-Actuation) Logic	N/A	F and G	1. Part of Group 6 Logic.
2	Group 3 (Initiating) Logic	N/A	C	1. Refer to Table 3.7.A for list of valves.
1	Group 3 (Actuation) Logic	N/A	C	
1	Group 6 Logic	N/A	F and G	1. Refer to Table 3.7.A for list of valves.
1	Group 8 (Initiating) Logic	N/A	J	1. Refer to Table 3.7.A for list of valves. 2. Same as Group 2 initiating logic.
1	Reactor Building Isolation (refueling floor) Logic	N/A	H or F	1. Logic has permissive to refueling floor static pressure regulator.
1	Reactor Building Isolation (reactor zone) Logic	N/A	H or G or A	1. Logic has permissive to reactor zone static pressure regulator.

6. Channel shared by RPS and Primary Containment & Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

7. A train is considered a trip system.

8. Two out of three SGTS trains required. A failure of more than one will require action A and F.

9. There is only one trip system with auto transfer to two power sources.

10. Refer to Table 3.7.A and its notes for a listing of Isolation Valve Groups and their initiating signals.

11. A channel may be placed in an inoperable status for up to four hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

12. A channel contains four sensors, all of which must be operable for the channel to be operable.

Power operations permitted for up to 30 days with 15 of the 16 temperature switches operable.

13. The nominal setpoints for alarm and reactor trip (1.5 and 3.0 times background, respectively) are established based on the normal background at full power. The allowable setpoints for alarm and reactor trip are 1.2-1.8 and 2.4-3.6 times background, respectively.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
1(10)	RHR Area Cooler Fan Logic	N/A	A	
1(10)	Core Spray Area Cooler Fan Logic	N/A	A	
1(11)	Instrument Channel - Core Spray Motors A or C Start	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(11)	Instrument Channel - Core Spray Motors B or D Start	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(12)	Instrument Channel - Core Spray Loop 1 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(12)	Instrument Channel - Core Spray Loop 2 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(13)	RHRSW Initiate Logic	N/A	(14)	
1	RPT logic	N/A	(17)	1. Trips recirculation pumps on turbine control valve fast closure or stop valve closure > 30% power.

Amendment No. ~~35~~, 102

Amendment No. 62, ~~82~~, 102

TABLE 3.2.F
SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	LI-3-46 A LI-3-46 B	Reactor Water Level	Indicator - 155" to +60"	(1) (2) (3)
2	PI-3-54 PI-3-61	Reactor Pressure	Indicator 0-1500 psig	(1) (2) (3)
2	PR-64-50 PI-64-67	Drywell Pressure	Recorder 0-80 psia Indicator 0-80 psia	(1) (2) (3)
2	TI-64-52 TR-64-52	Drywell Temperature	Recorder, Indicator 0-400°F	(1) (2) (3)
1	TR-64-52	Suppression Chamber Air Temperature	Recorder 0-400°F	(1) (2) (3)
2	TI-64-55 TIS-64-55	Suppression Chamber Water Temperature	Indicator, 0-400°F	(1) (2) (3)
2	LI-64-54 A LI-64-66	Suppression Chamber Water Level	Indicator -25" to +25"	(1) (2) (3)
1	N/A	Control Rod Position	6V Indicating Lights)	(1) (2) (3) (4)
1	N/A	Neutron Monitoring	SRM, IRM, LPRM 0 to 100% power)	
1	PS-64-67	Drywell Pressure	Alarm at 35 psig)	(1) (2) (3) (4)
1	TR-64-52 and PS-64-58 B and IS-64-67	Drywell Temperature and Pressure and Timer	Alarm if temp. > 281°F and pressure > 2.5 psig after 30 minute delay)	
1	LI-84-2A	CAD Tank "A" Level	Indicator 0 to 100%	(1)
1	LI-84-13A	CAD Tank "B" Level	Indicator 0 to 100%	(1)

and trips the recirculation pumps.
The low reactor water level instrumentation that is set to trip when reactor water level is 17.7" (378" above vessel zero) above the top of the active fuel (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation, and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and, in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus, the results given above are applicable here also.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steam line valve closure limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in these areas. Trips are provided on this instrumentation and, when exceeded, cause closure of isolation valves. The setting of 200°F for the main steam line tunnel detector is low enough to detect leaks of the order of 15 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established nominal setting of 3 times normal background and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.2 FSAR. An alarm with a nominal setpoint of 1.5 x normal full-power background is provided also.

Pressure instrumentation is provided to close the main steam isolation valves in Run Mode when the main steam line pressure drops below 825 psig.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7 CONTAINMENT SYSTEMS	4.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

g. Local Leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves, which are not part of a water-sealed system, at not less than 49.6 psig (except for the main steam isolation valves, see 4.7.A.2.i) and not less than 54.6 psig for water-sealed valves each operating cycle. Bolted double-gasketed seals shall be tested whenever the seal is closed after being opened and at least once per operating cycle. Acceptable methods of testing are halide gas detection, soap bubbles, pressure decay, hydrostatically pressurized fluid flow or equivalent.

The personnel air lock shall be tested at 6-month intervals at an internal pressure of not less than 49.6 psig. In addition, if the personnel air lock is opened during periods when containment integrity is not required, a test at the end of such a period will be conducted at not less than 49.6 psig. If the personnel air lock is opened during a period when containment integrity is required, a test at≥2.5 psig shall be conducted within 3 days after being opened. If the air lock is opened more frequently than once every 3 days, the air lock shall be tested at least once every 3 days during the period of frequent openings.

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

The total leakage from all penetrations and isolation valves shall not exceed 60 percent of L_a per 24 hours. Leakage from containment isolation valves that terminate below suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available to ensure the sealing function for at least 30 days at a pressure of 54.6 psig. Leakage from containment isolation valves that are in closed-loop, seismic class I lines that will be water sealed during a DBA will be measured but will be excluded when computing the total leakage. Penetrations and isolation valves are identified as follows:

(1) Testable penetrations with double O-ring seals - Table 3.7.B,

(2) Testable penetrations with testable bellows Table 3.7.C,

(3) Isolation valves without fluid seal - Table 3.7.D,

(4) Testable electrical penetrations - Table 3.7.H, and

(5) Isolation valves sealed with fluid - Tables 3.7.E, and 3.7.F.

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (Sec.)	Normal Position	Action On Initiating Signal
6	Torus Hydrogen Sample Line Valves Analyzer A (FSV-76-55, 56)		2	NA	Note 1	SC
6	Torus Oxygen Sample Line Valves Analyzer A (FSV-76-53, 54)		2	NA	Note 1	SC
6	Drywell Hydrogen Sample Line Valves Analyzer A (FSV-76-49, 50)		2	NA	Note 1	SC
6	Drywell Oxygen Sample Line Valves Analyzer A (FSV-76-51, 52)		2	NA	Note 1	SC
6	Sample Return Valves - Analyzer A FSV-76-57, 58)		2	NA	0	GC
6	Torus Hydrogen Sample Line Valves Analyzer B (FSV-76-65, 66)		2	NA	Note 1	SC
6	Torus Oxygen Sample Line Valves-Analyzer B (FSV-76-63, 64)		2	NA	Note 1	SC
6	Drywell Hydrogen Sample Line Valves-Analyzer B (FSV-76-59, 60)		2	NA	Note 1	SC
6	Drywell Oxygen Sample Line Valves-Analyzer B (FSV-76-61, 62)		2	NA	Note 1	SC
6	Sample Return Valves-Analyzer B (FSV-76-67, 68)		2	NA	0	GC

Note 1: Analyzers are such that one is sampling drywell hydrogen and oxygen (valves from drywell open-valves from torus closed) while the other is sampling torus hydrogen and oxygen (valves from torus open - valves from drywell closed)

Group 7: The valves in Group 7 are automatically actuated by only the following condition:

1. The respective turbine steam supply valve not fully closed.

Group 8: The valves in Group 8 are automatically actuated by only the following condition:

1. High Drywell pressure

2. Reactor vessel low water level (538")

Amendment No. ~~90~~, 102

Correction letter of 9-19-84

Table 3.7.4 (Continued)

X-107B	Spare (testable)
X-108A	Power
X-108B	CRD Rod Position Indic.
X-109	CRD Rod Position Indic.
X-110A	Power
X-110B	CRD Rod Position Indic.
X-200A-SC	S/RV Test Instrumentation (Temporary)
X-219	Suppression Chamber Vacuum Breaker

BASES

Group 1 - process lines are isolated by reactor vessel low water level (490") 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 are also closed when process instrumentation detects excessive main steam line flow, high radiation, low pressure, or main steam space high temperature.

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 non-safety 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 vlaves 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 75 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NOS. 1, 2 AND 3

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

1.0 INTRODUCTION

By letter dated April 9, 1984 (TVA BFNP TS 197) the Tennessee Valley Authority (the licensee or TVA) requested amendments to Facility Operating License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3. The proposed amendments would allow extension of surveillance intervals, clarify various instrumentation requirements, and make corrections to reflect plant modifications and changes to regulations.

2.0 EVALUATION

Surveillance Intervals (Units 1, 2 and 3)

The definition of Surveillance Interval (Section 1.0.2) would be revised to allow extension of a surveillance interval by 25% with the limitation that three consecutive intervals do not exceed 3.25 times the single specified interval.

The BWR Standard Technical Specifications, NUREG-0123, Revision 3, served as the basis in assessing the conformance of the licensee's proposed Technical Specification change. The Standard Technical Specifications, (and the associated Bases) are recognized by the staff as an acceptable implementation of the applicable requirements of 10 CFR 50.36.

We have reviewed the proposed change and find the licensee's proposed Technical Specification change to be consistent with Paragraph 4.0.2 of the BWR Standard Technical Specifications. Therefore, we conclude that the change is acceptable.

Editorial Correction (Unit 1)

Technical Specification Section 4.1.C is presently placed between 4.1.A and 4.1.B. It is proposed that 4.1.C be made a part of 1.A. This will restore alphabetical order without affecting the actual surveillance requirements and is therefore an acceptable change.

Primary Containment Isolation Due to Reactor Low Water Level (Units 1, 2 and 3)

Table 3.2.A, "Primary Containment and Reactor Building Isolation Instrumentation" contains a remark which indicates that reactor low water level (538 inches above vessel zero) initiates primary containment isolation. This remark requires clarification as there are seven groups of primary containment isolation valves and reactor water level at 538 inches initiates isolation of only three groups (2, 3 and 6) of the seven, as noted in the notes to Table 3.7.A, "Primary Containment Isolation Valves." A proposed change would clarify Table 3.2.A and is acceptable based on the fact that requirements would not be added, deleted, or modified by the change.

Main Steamline Radiation Alarm and Trip Setpoints (Units 1, 2 and 3)

Specification 3.2.A specifies a setpoint of "three times normal rated full power background level" for main steamline high radiation instrument channels serving containment isolation functions. As discussed in Regulatory Guide 1.105, instruments should have a margin between the setpoint and allowable process value to allow for instrument drift during the surveillance interval. A change requested by the licensee would not change the instrument channel setpoint, but would define the margin. The margin proposed by the licensee is consistent with Standard Technical Specifications. Based on conformance to Regulatory Guide 1.105, this change is acceptable.

TIP Isolation (Units 1, 2 and 3)

The transversing incore probe (TIP) system is provided with automatic isolation. The isolation valves are designated as Group 8, as indicated in Table 3.7.A of the Technical Specifications. Initiating conditions for Group 8 isolation are high drywell pressure or reactor low water (538 inches). TIP isolation on reactor low water or high drywell pressure is sufficiently diverse and reliable to meet the acceptance criteria of Standard Review Plan Section 6.2.4. Proposed changes to the Technical Specifications 3.2.A, 3.7.D and 3.7.D (Bases) would correct descriptions of TIP isolation logic which are presently incorrect. These changes are therefore acceptable.

RHRSW Pump Instrumentation (Units 1, 2 and 3)

Browns Ferry Units 1, 2 and 3 have shared residual heat removal service water (RHRSW) headers and emergency equipment cooling water (EECW) headers. The headers are served by twelve RHRSW pumps; eight of which are provided with instrumentation for automatic starting. The licensee has proposed changes to the Technical Specifications to correct errors in Table 3.2.B. These changes will revise the pump automatic starting assignments to be consistent with the installed instrumentation as described in FSAR Figure 10.9.3, note 2. These changes are therefore acceptable.

Reactor Pressure Indicator Range (Units 1, 2 and 3)

Table 3.2.F of the Technical Specifications indicates that the reactor pressure indicators have a range of 0-1200 psig. The installed instruments have a range of 0-1500 psig. The licensee has requested that Table 3.2.F be revised to indicate the actual instrument range. Based on Regulatory Guide 1.105 guidance regarding margin between process limits and instrument limits, a range of 0-1500 is acceptable. The proposed change is therefore acceptable.

Air Lock Doors (Unit 2)

Air lock doors have been modified by adding strongbacks to permit testing by pressurizing the air lock to 49.6 psig (Pa) as required by Appendix J to 10 CFR Part 50. The licensee has requested changes to Technical Specification Section 4.7.A.2.g to reflect the revised test method. The new test method is consistent with 10 CFR 50 Appendix J Section III.D.2(b). The requested change is therefore acceptable.

H_2O_2 Monitoring System Isolation Valves (Unit 2)

In Amendment No. 82, Technical Specifications were revised to reflect installation of the Hays-Republic hydrogen-oxygen (H_2O_2) monitoring system. That amendment added a new page 251A to Table 3.7.A, "Primary Containment Isolation Valves," indicating that each sample line contains an inboard isolation valve and an outboard isolation valve. The correct configuration is (as indicated in Unit 1 - Amendment No. 92) two outboard isolation valves. Use of two outboard valves is consistent with Paragraph 5.2.3.5 of the Final Safety Analysis Report. The licensee has requested that page 251A be corrected to reflect the actual configuration. This change is acceptable.

Testable Electrical Penetrations (Units 1, 2 and 3)

The licensee has requested changes to Table 3.7.H, "Testable Electrical Penetrations," which would delete penetration "X-230 Containment Air Monitoring System" and add penetration "X-219 Suppression Chamber Vacuum Breaker." Penetration X-230 is not a testable penetration. Penetration X-219 is a testable penetration that was inadvertently omitted from the Technical Specifications. These changes are acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

The amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and a change in a surveillance requirement. 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 the 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 or environmental assessment need be prepared in connection with the issuance of the amendments.

4.0 CONCLUSION

We have 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: W. Long

Dated: August 13, 1984