December 12, 1983

Docket No. 50-259

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 No. 92 to Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit 1. This amendment changes the Technical Specifications in partial response to your application of July 13, 1983 (TVA BFNP TS 190).

The amendment revises the Technical Specifications to reflect modifications performed during the current refueling outage. The amendment does not address the analog trip system modification or the RPS power supply monitoring modification.

A copy of the Safety Evaluation is also enclosed.

Sincerely,

Original signed by/

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

Enclosures:
1. Amendment No. 92 to License No. DPR-33
2. Safety Evaluation

cc w/enclosures:
See next page

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

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 92
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 13, 1983, 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 License No. DPR-33 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical specifications contained in Appendices A and B, as revised through Amendment No. 92, 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: December 12, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 92

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

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

ii	105
iii	105a
iv	126
v	145
vi	181
31	231
32	232
40	233
42	234
66	251a
73	258
78	267
79	268
80	273
98	321
99	322
102	323

2. The marginal lines on these pages indicate the revised area.

LIST OF TABLES

3.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective

To assure the operability of the reactor protection system.

Specification

A. When there is fuel in the vessel, the setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.A.

B. Two RPS power monitoring channels for each inservice RPS MG sets or alternate source shall be operable.

1. With one RPS electric power monitoring channel for inservice RPS MG set or alternate power supply inoperable, restore the inoperable channel to operable status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

4.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.A and 4.1.B respectively.

C. When it is determined that a channel is failed in the unsafe condition, the other RPS channels that monitor the same variable shall be functionally tested immediately before the trip system containing the failure is tripped. The trip system containing the unsafe failure may untripped for short periods of time to allow functional testing of the other trip system. The trip system may be in the untripped position for no more than eight hours per functional test period for this testing.

3.1 REACTOR PROTECTION SYSTEM

B.2 With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one to operable status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

4.1 REACTOR PROTECTION SYSTEM

B. The RPS power monitoring system instrumentation shall be determined operable:

1. At least once per 6 months by performance of channel functional tests,

TABLE 4.1.B
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled startups (6)	Note (4)
APRM High Flux			
Output Signal	B	Heat Balance	Once every 7 days
Flow Bias Signal	B	Calibrate Flow Bias Signal (7)	Once/operating cycle
LPRM Signal	B	TIP System Traverse (8)	Every 1000 Effective Full Power Hours
High Reactor Pressure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	A	Standard Pressure Source	Every 3 Months
Reactor Low Water Level	A	Pressure Standard	Every 3 Months
High Water Level in Scram Discharge Volume Float Switches (LS-85-45C-F)	A	Calibrated Water Column (5)	Note (5)
High Water Level in Scram Discharge Volume Electronic Level Switches (LS-85-45-A, B, G, H)	B	Calibrated Water Column	Once/Operating Cycle (9)
Turbine Condenser Low Vacuum	A	Standard Vacuum Source	Every 3 Months
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 Months
Turbine First Stage Pressure Permissive (PT-1-81A and B, PT-1-91A and B)	B	Standard Pressure Source	Once/Operating Cycle (9)
Turbine Stop Valve Closure	A	Note (5)	Note (5)

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection trip system is supplied, via a separate bus, by its own high inertia, ac motor-generator set. Alternate power is available to either Reactor Protection System bus from an electrical bus that can receive standby electrical power. The RPS monitoring system provides an isolation between non-class 1E power supply and the class 1E RPS bus. This will ensure that failure of a non-class 1E reactor protection power supply will not cause adverse interaction to the class 1E Reactor Protection System.

The reactor protection system is made up of two independent trip systems (refer to Section 7.2, FSAR). There are usually four channels provided to monitor each critical parameter, with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic such that either channel trip will trip that trip system. The simultaneous tripping of both trip systems will produce a reactor scram.

This system meets the intent of IEEE - 279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each trip system logic has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor. Three APRM instrument channels are provided for each protection trip system.

Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast closure, turbine stop valve closure and loss of condenser vacuum are discussed in Specification 2.1 and 2.2.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
1	Core Spray Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	ADS Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems and valves.
1	HPCI Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	RCIC Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1(2)	Instrument Channel - Condensate Header Low Level (LS-73-55A & B)	$\geq$ Elev. 551'	A	1. Below trip setting will open HPCI suction valves to the suppression chamber.
1(2)	Instrument Channel - Suppression Chamber High Level	$\leq$ 7" above normal water level	A	1. Above trip setting will open HPCI suction valves to the suppression chamber.
2(2)	Instrument Channel - Reactor High Water Level	$\leq$ 583" above vessel zero.	A	1. Above trip setting trips RCIC turbine.
1	Instrument Channel - RCIC Turbine Steam Line High Flow	$\leq$ 450" H_2O (7)	A	1. Above trip setting isolates RCIC system and trips RCIC turbine.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum No. Operable Per Trip Sys (5)	Function	Trip Level Setting
2(1)	APRM Upscale (Flow Bias)	≤0.66W+42% (2)
2(1)	APRM Upscale (Startup Mode) (8)	≤12%
2(1)	APRM Downscale (9)	≥3%
2(1)	APRM Inoperative	(10b)
1(7)	RBM Upscale (Flow Bias)	≤ 0.66W+40% (2) (13)
1(7)	RBM Downscale (9)	≥ 3%
1(7)	RBM Inoperative	(10c)
3(1)	IRM Upscale (8)	≤ 108/125 of full scale
3(1)	IRM Downscale (3) (8)	≥ 5/125 of full scale
3(1)	IRM Detector not in Startup Position (8)	(11)
3(1)	IRM Inoperative (8)	(10a)
2(1) (6)	SRM Upscale (8)	≤ 1×10^5 counts/sec.
2(1) (6)	SRM Downscale (4) (8)	≥ 3 counts/sec.
2(1) (6)	SRM Detector not in Startup Position (4) (8)	(11)
2(1) (6)	SRM Inoperative (8)	(10a)
2(1)	Flow Bias Comparator	≤ 10% difference in recirculation flows
2(1)	Flow Bias Upscale	≤ 115% recirculation flow
1(1)	Rod Block Logic	N/A
2(1)	RSCS Restraint (PS-85-61A and PS-85-61B)	147 psig turbine first-stage pressure
1(12)	High Water Level in West Scram Discharge Tank (LS-85-45L)	≤25 gal.
1(12)	High Water Level in East Scram Discharge Tank (LS-85-45M)	≤25 gal.

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 0-1200 psig	(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)
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)

TABLE 3.2.F

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	H_2M - 76 - 94 H_2M - 76 - 104	Drywell and Torus Hydrogen Concentration	0.1 - 20%	(1)
2	PdI-64-137 PdI-64-138	Drywell to Suppression Chamber Differential Pressure	Indicator 0 to 2 psid	(1) (2) (3)
1/Valve		Relief Valve Tailpipe Thermocouple Temperature or Acoustic Monitor on Relief Valve Tailpipe		(5)
2	LI-64-159A XR-64-159	Suppression Chamber Water Level-Wide Range	Indicator, Recorder 0-240"	(1) (2) (3)
2	PI-64-39A XR-64-159 PI-64- 160A XR-64-159	Drywell Pressure Low Range Drywell Pressure Wide Range	Indicator, Recorder -5 to +5 psig Indicator, Recorder 0-300 psig	(1) (2) (3)
2	TI-64-161 TR-64-161 TI-64-162 TR-64-162	Suppression Pool Bulk Temperature	Indicator, Recorder 30° - 230° F	(1) (2) (3) (4) (6)

NOTES FOR TABLE 3.2.F

(1) From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

(2) From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made operable.

(3) If the requirements of notes (1) and (2) cannot be met, and if one of the indications cannot be restored in (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold condition within 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

(5) From and after the date that both the acoustic monitor and the temperature indication on any one valve fails to indicate in the control room, continued operation is permissible during the succeeding thirty days, unless one of the two monitoring channels is sooner made available. If both the primary and secondary indication on any SRV tailpipe is inoperable, the torus temperature will be monitored at least once per shift to observe any unexplained temperature increase which might be indicative of an open SRV.

(6) A channel consists of 8 sensors, one from each alternating torus bay. Seven sensors must be operable for the channel to be operable.

TABLE 4.2.B (Continued)

Function	Functional Test	Calibration	Instrument Check
Instrument Channel RHR Pump Discharge Pressure	(1)	once/3 months	none
Instrument Channel Core Spray Pump Discharge Pressure	(1)	once/3 months	none
Core Spray Sparger to RPV d/p	(1)	once/3 months	once/day
Trip System Bus Power Monitor	once/operating cycle	N/A	none
Instrument Channel Condensate Header Low Level (LS-73-56A,B)	(1)	once/3 months	none
Instrument Channel Suppression Chamber High Level	(1)	once/3 months	none
Instrument Channel Reactor High Water Level	(1)	once/3 months	once/day
Instrument Channel RCIC Turbine Steam Line High Flow	(1)	once/3 months	none
Instrument Channel RCIC Steam Line Space High Temperature	(1)	once/3 months	none

TABLE 4.2.B (Continued)

Function	Functional Test	Calibration	Instrument Check
Instrument Channel HPCI Turbine Steam Line High Flow	(1)	once/3 months	none
Instrument Channel HPCI Steam Line Space High Temperature	(1)	once/3 months	none
Core Spray System Logic	once/6 months	(6)	N/A
RCIC System (Initiating) Logic	once/6 months	N/A	N/A
RCIC System (Isolation) Logic	once/6 months	(6)	N/A
HPCI System (Initiating) Logic	once/6 months	(6)	N/A
HPCI System (Isolation) Logic	once/6 months	(6)	N/A
ADS Logic	once/6 months	(6)	N/A
LPCI (Initiating) Logic	once/6 months	(6)	N/A
LPCI (Containment Spray) Logic	once/6 months	(6)	N/A
Core Spray System Auto Initiation Inhibit (Core Spray Auto Initiation)	once/6 months (7)	N/A	N/A
LPCI Auto Initiation Inhibit (LPCI Auto Initiation)	once/6 months (7)	N/A	N/A

TABLE 4.2.C
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE ROD BLOCKS

Function	Functional Test	Calibration (17)	Instrument Check
APRM Upscale (Flow Bias)	(1) (13)	once/3 months	once/day (8)
APRM Upscale (Startup Mode)	(1) (13)	once/3 months	once/day (8)
APRM Downscale	(1) (13)	once/3 months	once/day (8)
APRM Inoperative	(1) (13)	N/A	once/day (8)
RBM Upscale (Flow Bias)	(1) (13)	once/6 months	once/day (8)
RBM Downscale	(1) (13)	once/6 months	once/day (8)
RBM Inoperative	(1) (13)	N/A	once/day (8)
IRM Upscale	(1) (2) (13)	once/3 months	once/day (8)
IRM Downscale	(1) (2) (13)	once/3 months	once/day (8)
IRM Detector not in Startup Position	(2) (once/operating cycle)	once/operating cycle (12)	N/A
IRM Inoperative	(1) (2) (13)	N/A	N/A
SRM Upscale	(1) (2) (13)	once/3 months	once/day (8)
SRM Downscale	(1) (2) (13)	once/3 months	once/day (8)
SRM Detector not in Startup Position	(2) (once/operating cycle)	once/operating cycle (12)	N/A
SRM Inoperative	(1) (2) (13)	N/A	N/A
Flow Bias Comparator	(1) (15)	once/operating cycle (20)	N/A
Flow Bias Upscale	(1) (15)	once/3 months	N/A
Rod Block Logic	(16)	N/A	N/A
RSCS Restraint	(1)	once/3 months	N/A
West Scram Discharge Tank Water Level High (LS-85-45L)	once/quarter	once/operating cycle	N/A
East Scram Discharge Tank Water Level High (LS-85-45M)	once/quarter	once/operating cycle	N/A

TABLE 4.2.F
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
1) Reactor Water Level	Once/6 months	Each Shift
2) Reactor Pressure	Once/6 months	Each Shift
4) Drywell Temperature	Once/6 months	Each Shift
5) Suppression Chamber Air Temperature	Once/6 months	Each Shift
8) Control Rod Position	NA	Each Shift
9) Neutron Monitoring	(2)	Each Shift
10) Drywell Pressure (PS-64-67)	Once/6 months	NA
11) Drywell Pressure (PS-64-58B)	Once/6 months	NA
12) Drywell Temperature (TR-64-52)	Once/6 months	NA
13) Timer (IS-64-67)	Once/6 months	NA
14) CAD Tank Level	Once/6 months	Once/day
15) Containment Atmosphere Monitors	Once/6 months	Once/day
16) Drywell to Suppression Chamber Differential Pressure	Once /6 months	Each Shift

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TABLE 4.2.F
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
17 Relief valve Tailpipe Thermocouple Temperature	NA	Once/month (24)
18 Acoustic Monitor on Relief Valve Tailpipe	Once/cycle (25)	Once/month (26)
19. Suppression Chamber Water Level-Wide Range (LI-64-159A) (XR-64-159)	Once/cycle	Once/month
20. Drywell Pressure - Low Range (PI-64-39A) (XR-64-159)	Once/cycle	Once/shift
21. Drywell Pressure - Wide Range (PI-64-160A)(XR-64-159)	Once/cycle	Once/shift
22. Suppression Pool Bulk Temperature (TI-64-161) (TR-64-161) (TI-64-162((TR-64-162)	Once/cycle	Once/shift

3.3 Reactivity Control

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.

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.

2. In the event any SDV drain or vent valve becomes inoperable, reactor operation may continue provided the redundant drain or vent valve is operable.

3. If redundant drain or vent valves become inoperable, the reactor shall be in hot standby within 24 hours.

4.3 Reactivity Control

E. Surveillance requirements are as specified in 4.3.C and .D above.

F. Scram Discharge Volume (SDV)

1.a. The scram discharge volume drain and vent valves shall be verified open prior to each starup 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 monthly.

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. No additional surveillance required.

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

1. The RHRS shall be operable:

   (1) prior to a reactor 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 RHRS 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.

3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain operable.

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

1. a.	Simulated Automatic Actuation Test	Once/ Operating Cycle
b.	Pump Operability	Once/ month
c.	Motor Operated valve operability	Once/ month
d.	Pump Flow Rate	Once/3 months
e.	Test Check Valve	Once/ Operating Cycle

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12000 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.

3. When it is determined that one RHR pump (LPCI mode) is inoperable at a time when operability is required, the remaining RHR pumps (LPCI mode) and active components in both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators shall be demonstrated to be operable immediately and daily thereafter.

3.6.C Coolant Leakage

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 Condition within 24 hours.

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.

E. Jet Pumps

1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be shutdown in the Cold Condition within 24 hours.

4.6.C Coolant Leakage

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. Once during each operating cycle, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

3. The integrity of the relief/safety valve bellows shall be continuously monitored.

4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the startup or run modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

   a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

3.7 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 $\geq$ 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	SURVEILLANCE REQUIREMENTS
3.7 CONTAINMENT SYSTEMS	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.

3.7.A Primary Containment

4.7.A Primary Containment

within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

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.

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 set point 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 c, below.

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

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 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 exercised through an opening-closing cycle every month.

   b. When it is determined that two vacuum breakers are inoperable for opening at a time when operability is all other vacuum breaker

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	SC
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	SC

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)

TABLE 3.7.D

AIR TESTED ISOLATION VALVES

Valve	Valve Identification
1-14	Main Steam
1-15	Main Steam
1-26	Main Steam
1-27	Main Steam
1-37	Main Steam
1-38	Main Steam
1-51	Main Steam
1-52	Main Steam
1-55	Main Steam Drain
1-56	Main Steam Drain
2-1192	Service Water
2-1383	Service Water
3-554	Feedwater
3-558	Feedwater
3-568	Feedwater
3-572	Feedwater
32-62	Drywell Compressor Suction
32-63	Drywell Compressor Suction
32-336	Drywell Compressor Return
32-2163	Drywell Compressor Return
32-2516	Drywell Compressor Return
32-2521	Drywell Compressor Return
33-1070	Service Air
33-785	Service Air
43-13	Reactor Water Sample Lines
43-14	Reactor Water Sample Lines
63-525	Standby Liquid Control Discharge
63-526	Standby Liquid Control Discharge
64-17	Drywell and Suppression Chamber Air Purge Inlet
64-18	Drywell Air Purge Inlet
64-19	Suppression Chamber Air Purge Inlet
64-20	Suppression Chamber Vacuum Relief
64-c.v.	Suppression Chamber Vacuum Relief
64-21	Suppression Chamber Vacuum Relief
64-c.v.	Suppression Chamber Vacuum Relief
64-29	Drywell Main Exhaust
64-30	Drywell Main Exhaust
64-32	Suppression Chamber Main Exhaust
64-33	Suppression Chamber Main Exhaust
64-31	Drywell exhaust to Standby Gas Treatment
64-34	Suppression Chamber to Standby Gas Treatment
64-139	Drywell pressurization, Compressor Suction
64-140	Drywell pressurization, Compressor Discharge
68-508	CRD to RC Pump Seals
68-523	CRD to RC Pump Seals
68-550	CRD to RC Pump Seals
68-555	CRD to RC Pump Seals

BASES

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination, ensure that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

During initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required, there will be no pressure on the system thus greatly reducing the changes of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect to minimize the probability of an accident occuring.

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 49.6 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50 (type A, B, and C tests).

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat release during primary system blowdown from 1,035 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting form isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water levels given in the specification, containment pressure during the design basis accident is approximately 49 psig, which is below the maximum of 62 psig. The maximum water level indications of -1 inch corresponds to a downcomer submergence of 3 feet 7 inches and a water volume of 127,800 cubic feet with or 128,700 cubic feet without the drywell-suppression chamber differential pressure control. The minimum water level indication of -6.25 inches with differential pressure control and -7.25 inches without differential pressure control corresponds to a downcomer submergence of approximately 3 feet and water volume of approximately 123,000 cubic feet. Maintaining the water level between these levels will ensure that the torus water volume and downcomer submergence are within the aforementioned limits during normal plant operation. Alarms, adjusted for instrument error, will notify the operator when the limits of the torus water level are approached.

The maximum permissible bulk pool temperature is limited by the potential for stable and complete condensation of steam discharged from safety relief valves and adequate core spray pump net positive suction head. At reactor vessel pressures above approximately 555 psig, the bulk pool temperature shall not exceed 180°F. At pressures below approximately 240 psig, the bulk temperature may be as much as 184°F. At intermediate pressures, linear interpolation of the bulk temperature is permitted.

BASES

They also represent the bounding upper limits that are used in suppression pool temperature response analyses for safety relief valve discharge and LOCA cases. The actions required by specification 3.7.c-f assure the reactor can be depressurized in a timely manner to avoid exceeding the maximum bulk suppression pool water limits. Furthermore, the 184°F limit provides that adequate RHR and core spray pump NPSH will be available without dependency on containment overpressure.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability. Under full power operation conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a peak long term water temperature which is sufficient for complete condensation.

Limiting suppression pool temperature to 105°F during RCIC, HPCI, or relief valve operation when decay heat and stored energy is removed from the primary system by discharging reactor steam directly to the suppression chamber ensures adequate margin for controlled blowdown anytime during RCIC operation and ensures margin for complete condensation of steam from the design basis loss-of-coolant accident.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated form that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

If a loss-of-coolant accident were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62 psig code permissible pressures even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212°F provides additional margin above that available at 330°F.

BASES

The interior surfaces of the drywell and suppression chamber are coated as necessary to provide corrosion protection and to provide a more easily decontaminable surface. The surveillance inspection of the internal surfaces each operating cycle assures timely detection of corrosion. Dropping the torus water level to one foot below the normal operating level enables an inspection of the suppression chamber where problems would first begin to show.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be about 49 psig which would rapidly reduce to less than 30 psig within 20 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig within 25 seconds, equalizes with drywell pressure, and decays with the drywell pressure decay.

The design pressure of the drywell and suppression chamber is 56 psig. The design leak rate is 0.5 percent per day at the pressure of 56 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 25 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The calculated radiological doses given in Section 14.9 of the FSAR were based on an assumed leakage rate of 0.635 percent at the maximum calculated pressure of 49.6 psig. The doses calculated by the NRC using this bases are 0.14 rem, whole body passing cloud gamma dose, and 15.0 rem, thyroid dose, which are respectively only 5×10^{-3} and 10^{-1} times the 10 CFR 100 reference doses. Increasing the assumed leakage rate at 49.6 psig to 2.0 percent as indicated in the specifications would increase these doses approximately a factor of 3, still leaving a margin between the calculated dose and the 10 CFR 100 reference values.

Establishing the test limit of 2.0%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly

3.11 FIRE PROTECTION SYSTEMS

D. ROVING FIRE WATCH

A roving fire watch will tour each area in which automatic fire suppression systems are to be installed (as described in the "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2," Section X) at intervals no greater than 2 hours. A keyclock recording type system shall be used to monitor the routes of the roving fire watch. The patrol will be discontinued as the automatic suppression systems are installed and made operable for each specified area.

4.11 FIRE PROTECTION SYSTEMS

3. The class A supervised detector alarm circuits will be tested once each two months at the local panels.

4. The circuits between the local panels in 4.11.C.3 and the main control room will be tested monthly.

5. Smoke detector sensitivity will be checked in accordance with manufacturer's instruction annually.

D. ROVING FIRE WATCH

A monthly walk-through by the Safety Engineer will be made to visually inspect the plant fire protection system for signs of damage, deterioration, or abnormal conditions which could jeopardize proper operation of the system.

3.11 FIRE PROTECTION SYSTEMS

E. Fire Protection Systems Inspection

All fire barrier penetrations, including cable penetration barriers, fire doors and fire dampers, in fire zone boundaries protecting safety related areas shall be functional at all times. With one or more of the required fire barrier penetrations non-functional within one hour establish a continuous fire watch on at least one side of the affected penetration or verify the OPERABILITY of fire detectors on at least one side of the non-functional fire barrier and establish an hourly fire watch patrol until the work is completed and the barrier is restored to functional status.

F. Fire Protection Organization

The minimum in-plant fire protection organization and duties shall be as depicted in Figure 6.3-1.

4.11 FIRE PROTECTION SYSTEMS

E. Fire Protection Systems Inspections

Each required fire barrier penetration shall be verified to be functional at least once per 18 months by a visual inspection, and prior to restoring a fire barrier to functional status following repairs or maintenance by performance of a visual inspection of the affected fire barrier penetration.

F. Fire Protection Organization

No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION

3.11 FIRE PROTECTION SYSTEMS

G. Air Masks and Cylinders

A minimum of fifteen air masks and thirty 500 cubic inch air cylinders shall be available at all times except that a time period of 48 hours following emergency use is allowed to permit recharging or replacing.

H. Continuous Fire Watch

A continuous fire watch shall be stationed in the immediate vicinity where work involving open flame welding, or burning is in progress.

I. Open Flames, Welding, and Burning in the Cable Spreading Room

There shall be no use of open flame, welding, or burning in the cable spreading room unless the reactor is in the cold shutdown condition.

SURVEILLANCE REQUIREMENTS

4.11 FIRE PROTECTION SYSTEMS

G. Air Masks and Cylinders

No additional surveillance required.

H. Continuous Fire Watch

No additional surveillance required.

I. Open Flames, Welding, and Burning in the Cable Spreading Room

No additional surveillance required.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 92 TO FACILITY LICENSE NO. DPR-33

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-259

1.0 Introduction

By letter dated July 13, 1983 (TVA BFNP TS 190) the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit 1. The proposed amendment and revised Technical Specifications would (1) incorporate the limiting conditions for operation of the facility in the sixth fuel cycle following the fifth refueling of the reactor and (2) reflect modifications performed during the outage. This amendment addresses the changes to the Technical Specifications associated with all of the various modifications completed during this refueling outage except for the installation of an analog trip system and a reactor protection system power monitoring design modification. The latter modifications and the core reload are addressed in separate amendments.

2.0 Discussion

The modifications and the changes to the Technical Specifications were described in the Commission's Notice of this application published pursuant to Public Law 97-415 on October 28, 1983 (48 FR 49947).

3.0 Evaluation

3.1 Changes Related to Torus Modifications

One of the changes to the Technical Specifications (TS) is to revise the tables that list the surveillance instrumentation associated with the suppression pool bulk temperature. This modification provides an improved torus temperature monitoring system which consists of 16 sensors. This will provide a more accurate indication of the torus water bulk temperature as required by NUREG-0661 and will replace the suppression chamber water temperature instruments presently listed in the TS. The proposed changes to the TS place operability and calibration requirements on the new temperature monitoring system. The changes are acceptable.

Another change to the TS is to revise the bases for the present limits on temperature of water in the torus. The present bases for suppression pool temperature limits were founded on the Humboldt Bay and Bodega Bay tests. Consistent with the long-term torus integrity program of NUREG-0661 and NUREG-0783, the bases require change to account for steam mass fluxes through the safety/relief valve (S/RV) T-quenchers. The proposed bases describe assurances of stable and complete condensation of steam discharged through the S/RVs and adequate residual heat removal (RHR) and core spray pump net positive suction head. The bases do not contain any limits or action requirements; they provide the justification for the limiting conditions of operation and the surveillance requirements. The revised bases are acceptable. The staff will perform a post-implementation review of all the design modifications after TVA submits the required "Plant Unique Analysis" and results of the proof tests.

Section 4.5.B.1 of the Technical Specifications requires that every three months, the LPCI capability of the RHR pumps shall be demonstrated. In the tests, the pumps take suction from the torus and return the water to the torus. The pumps are required to demonstrate that two pumps in the same loop can deliver at least 15,000 gpm against an indicated system pressure (head) of 200 psig.

The two-pump 15,000 gpm LPCI test surveillance was determined to induce vibrations in the RHR return line to the torus. To eliminate the vibration, an orifice has been installed in the return line. However, installation of this orifice plate also decreases the suppression pool cooling mode of RHR operation from 15,000 gpm to approximately 12,000 gpm. A new containment cooling analysis was performed for this configuration, and it was determined that this flow rate induces a long-term suppression pool temperature well within that necessary for stable and complete steam condensation and for adequate RHR and core spray pumps net positive suction head. The revised test requirement is that the two pumps demonstrate that they can deliver 12,000 gpm against a higher head - 250 psig. The orifice is in the return line to the torus and does not change the volume of water that would be injected into the reactor during the LPCI mode. The 12,000 gpm at higher pump head pressure is equivalent to 15,000 gpm at lower discharge pressure. We conclude that the change has no adverse impact on the LPCI or containment cooling modes of RHR operation and is acceptable. As a supplement to this application, the licensee submitted a revised ECCS analysis by letter dated July 21, 1983. This was reviewed in conjunction with the core reload evaluation and as discussed in the reload amendment, was found to be acceptable. An identical change for Browns Ferry Unit 2 was approved by Amendment No. 85 to License No. DPR-52 issued March 11, 1983.

Section 4.7.A.2.k of the present Technical Specifications requires that if extended relief valve operation causes the temperature of the suppression pool to exceed 130°F, the reactor shall be shutdown and the torus and drywell visually inspected for signs of distress or displacement. Since the torus is being extensively upgraded to withstand dynamic loading significantly beyond that originally expected, extended operation of relief valves above a suppression pool temperature of 130°F is not expected to be a safety concern warranting placing the reactor in cold shutdown and performeing a torus inspection. Therefore, this requirement is being deleted. We have determined that this change is acceptable for the following reasons:

a. Browns Ferry Unit 1 is using an epoxy-base coating on its drywell and torus surfaces to replace the old organic-base coating which is not stable above 150°F. The new coating is stable at temperatures higher than 212°F. The present Technical Specification limit of 130°F is no longer meaningful.

b. Browns Ferry Unit 1 has modified its SRV discharge line structure. The new T-quencher will reduce unstable condensation considerably. This will also reduce the possible damage to crating caused by condensation.

The present Technical Specifications in the "Bases" for primary containment, discuss the specific type of protective coatings applied to the drywell and torus surfaces to protect the steel from corrosion and minimize contamination of the water. There have been significant developments in protective coating technology since the Browns Ferry units were licensed. During the torus modifications, the licensee has throughly sandblasted all torus surfaces in each unit and is applying coatings that offer more potential for sealing the surfaces. Therefore, the "Bases" are being generalized so that a Technical Specification change will not be required if a different protective coating is applied. As the name implies, the "Bases" do not contain limits or surveillance requirements; rather, the bases set forth the reasons and justification for the limits and requirements. In this case, the former use of an organic-based coating was translated into an operating limit on water temperature in the suppression pool. With the new protective coatings, these Bases are no longer relevant. We find the proposed change acceptable. This same change was approved for Browns Ferry Unit 2 by Amendment No. 85 to License No. DPR-52 issued March 11, 1983.

3.2 Scram Discharge Instrument Volume

The SDVs and SDIVs are being modified to address inadequacies identified by the partial rod insertion event on Browns Ferry Unit 3 in June 1980. One of the modifications includes adding another valve in series to the existing drain and vent valves on the SDV and SDIV. Another modification includes adding electronic level switches to initiate a scram on a high level in the SDIV. On June 24, 1983, the Commission issued Orders for the Browns Ferry Nuclear Plant, Units 1 and 3 to install permanent Scram Discharge System modifications during the Cycle 5 outages for Units 1 and 3 in accordance with Generic Letter 81-09, BWR Scram Discharge System. (This is the Cycle 5 outage for Unit 1.) The modifications have been previously completed for Unit 2. Both the modification of the systems and submission of TS change to place operability and surveillance requirements on the new instruments and valves were required of the licensee to be in compliance with the Commission Order.

We have reviewed the proposed changes to Sections 3.3/4.3 of the Technical Specifications and conclude that the proposed changes are consistent with the staff guidelines as stated in the December 1, 1980 BWR Scram Discharge

System Safety Evaluation. Further, these same proposed changes have been previously approved for Browns Ferry Unit 2 by Amendment No. 85. Thus, we conclude that the proposed changes in the Technical Specifications for Unit 1 are acceptable.

3.3 Accident Monitor Instrumentation

Item II.F.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," requires all licensees to install five new monitoring systems and to provide onsite sampling/analysis capability for a specified range of radionuclides. For all six categories, NUREG-0737 states: "Changes to technical specifications will be required." During this refueling outage, the licensee has installed: a) a containment high-range radiation monitoring system, b) a drywell wide-range pressure monitoring system and c) a suppression chamber wide-range water level monitoring system. These three items were required by NUREG-0737, items II.F.1.3, II.F.1.4 and II.F.1.5, respectively. The changes to the TS, which track the model TS provided to the licensee by the staff, are to add operability and surveillance requirements on the new monitoring systems to the TS. By letter dated October 12, 1983, the licensee informed us that the design of necessary cable connections to the drywell penetration for the installation of the high-range containment radiation monitoring instrumentation is inadequate and therefore, the operability and full environmental qualification of the installed system is questionable. TVA committed to keep the existing containment radiation monitor in service until the new equipment fully meets all the requirements of NUREG-0737, Item II.F.1.3; the existing instrumentation serves the same function as the new equipment but will not measure as wide a range. The licensee requested that the proposed changes to the TS for Item II.F.1.3 be withdrawn and that the present surveillance requirements of the existing equipment remain in effect. Thus, the changes to the TS by this amendment are to add operability and surveillance requirements for the instrumentation added to meet the requirements of Items II.F.1.4 and II.F.1.5. The revisions also delete the requirements on the present drywell pressure and suppression chamber water level instruments since they are being replaced by the new instrumentation. The proposed changes track the model TS requirements provided the licensee and are acceptable.

3.4 NUREG-0737, Item II.K.3.15

TMI Action Plan Item II.K.3.15 requires licensees of BWRs to modify pipe-break-detection circuitry so that pressure spikes resulting from high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) initiation will not cause inadvertent system isolation. The licensee elected to employ the BWR Owners Group modification which incorporates a three-second time delay relay (TDR) to prevent spurious isolation. In our letter to the licensee of October 13, 1981, we requested the licensee to provide certain analyses and to "propose the appropriate Surveillance Requirements and Limiting Conditions of Operation for the HPCI and RCIC systems which address this item." The safety evaluation was provided by the licensee's letter of December 16, 1981. All of the Browns Ferry units have always had a three-second TDR on the HPCI system. During the current outage for Browns Ferry Unit 1, a TDR was added to the RCIC system. The proposed changes to the Technical Specifications requiring calibration and surveillance of the time delay relays is in accordance with the requirements of NUREG-0737, Item II.K.3.15 and the staff's follow up letter. The changes are acceptable. The same changes were made to the Browns Ferry Units 2 and 3 TS by Amendment No. 85 to License No. DPR-52, issued March 11, 1983 and by Amendment No. 51 to License No. DPR-68, issued March 29, 1982, respectively. Thus, NUREG-0737, Item II.K.3.15, is satisfactorily implemented for Browns Ferry Units 1, 2 and 3.

3.5 Redundant Air Supply to Drywell

During the current outage, TVA has installed a second discharge line from the drywell compressor into containment. This line was added to provide the capability for isolation of approximately one-half of the drywell suppression equipment in the case of a drywell line leak. This air supply will be used to supply two inboard main steam isolation valves (MSIVs), approximately one-half of the main steam relief valves (MSRVs), and approximately one-half of all other air-operated equipment in the drywell. This will significantly reduce the possibility of any one control air pipe break inside containment from requiring immediate shutdown and isolation due to MSIVs, MSRVs, and drywell coolers being inoperable. Since any line penetrating containment requires two isolation valves, the table in the Technical Specifications listing the isolation valves that must be periodically tested is being revised to add these two new isolation valves. The changes to the TS are acceptable.

3.6 Modification of Airlock Doors

Section III.D.2(b) of Appendix J, 10 CFR Part 50, requires that air locks shall be tested at 6 month intervals at an internal pressure not less than P_a. P_a is defined as "the calculated peak containment internal pressure related to the design basis accident and specified either in the technical specification or associated bases." Reactor plants designed prior to the issuance of Appendix J often do not have the capability to test airlocks at P_a without the installation of strongbacks or the performance of mechanical adjustments to the operating mechanisms of the inner doors. The reason for this is that the inner doors are designed to seat with accident pressure on the containment side of the door, and therefore, the operating mechanisms were not designed to withstand accident pressure in the opposite direction. When the airlock is pressurized for a local airlock test (i.e., pressurized between the doors), pressure is exerted on the airlock side of the inner door, causing the door to unseat and preventing the performance of a meaningful test. The strongback or mechanical adjustments prevent the unseating of the inner door, allowing the test to proceed. Section 4.7.A.2.g of the present TS requires that "the personnel air lock shall be tested at a pressure of 49.6 psig during each operating cycle." The proposed change to the TS is to require that "the personnel air lock shall be tested at 6-month intervals at an internal pressure of not less than 49.6 psig." This modification and the proposed change conform the requirement in the TS to Appendix J and are clearly acceptable.

3.7 Administrative Changes

Several administrative changes are being made to the Technical Specifications. These include revising the Table of Contents to reflect the changes discussed above, an editorial change and corrections to the list of sample valves to reflect the current plant configuration. The changes are primarily editorial or reformating of the present requirements without changing the requirements. We have reviewed the changes and find them acceptable.

4.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

5.0 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Reviewers: H. Shaw, J. Mauck and T. Chan

Dated: December 12, 1983