FEB 15 1977

Docket No. 50-296

Tennessee Valley Authority
ATTN: Mr. Godwin Williams, Jr.
Manager of Power
818 Power Building
Chattanooga, Tennessee 37401

Gentlemen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 3 TO FACILITY OPERATING LICENSE FOR BROWNS FERRY NUCLEAR PLANT UNIT 3

The Nuclear Regulatory Commission has issued Amendment No. 3 to Facility Operating License No. DPR-68 to Tennessee Valley Authority. This amendment revises the Technical Specifications (Appendix A) of the Browns Ferry Nuclear Plant Unit 3 to provide a uniformity of statement with the Technical Specifications in effect for Units 1 and 2 of the Browns Ferry Nuclear Plant and to correct identified errors of understanding, omission, designation, grammar and spelling. In addition, the amendment addresses proposed modifications to certain valves, a part of the containment isolation system, which will be used to maintain a pressure differential between the drywell and torus atmospheres.

A copy of the Safety Evaluation and Notice of Issuance is also enclosed.

Sincerely,

Original Signed by
John F. Stolz

John F. Stolz, Chief
Light Water Reactors
Branch No. 1
Division of Project Management

Enclosures:
1. Amendment No. 3 to DPR-68
2. Safety Evaluation
3. Federal Register Notice

cc: See page 2

GD
const. 1

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DATE ➤	2/7/77	2/11/77	2/7/77			

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DISTRIBUTION OF AMENDMENT NO. 3 TO DPR-68 FOR BROWNS FERRY 3, DTD 2/15/77

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
License No. DPR-68

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

   A. The facility will operate in conformance with the provisions of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. 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;

   C. 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

   D. 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-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 3, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Light Water Reactors
Branch No. 1
Division of Project Management

Attachment:
Changes to the Appendix A
Technical Specifications

Date of Issuance: February 15, 1977

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
License No. DPR-68

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

   A. The facility will operate in conformance with the provisions of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. 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;

   C. 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

   D. 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-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 3, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

OFFICE➤						
SURNAME➤						
DATE➤						

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

FOR THE NUCLEAR REGULATORY COMMISSION
Original Signed by
John F. Stolz

John F. Stolz, Chief
Light Water Reactors
Branch No. 1
Division of Project Management

Attachment:
Changes to the Appendix A
Technical Specifications

Date of Issuance: February 15, 1977

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
License No. DPR-68

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

   A. 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;

   B. 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

   C. 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-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 3, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

OFFICE						
SURNAME						
DATE						

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-296

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 3 to Facility Operating License No. DPR-68 issued to Tennessee Valley Authority which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit 3, located in Limestone County, Alabama. The amendment is effective as of its date of issuance.

This amendment revises the Technical Specifications (Appendix A) of the Browns Ferry Nuclear Plant Unit 3 to provide a uniformity of statement with the Technical Specifications in effect for Units 1 and 2 of the Browns Ferry Nuclear Plant and to correct identified errors of understanding, omission, designation, grammar and spelling. In addition, the amendment addresses proposed modifications to certain valves, a part of the containment isolation system, which will be used to maintain a pressure differential between the drywell and torus atmospheres.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

OFFICE➤						
SURNAME➤						
DATE➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 3 TO DPR-68

DOCKET NO. 50-296

Replace the following pages of the Technical Specifications (Appendix A) with the attached sheets.

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uncertainties employed in deriving the safety limit are provided at the beginning of each fuel cycle.

The MCPR value used in the ECCS performance evaluation (1.18) is less limiting than the MCPR for operation (1.27).

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of MCPR = 1.05 would not produce boiling transition. Thus, although it is not required to establish the safety limit additional margin exists between the safety limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to BFNP operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operating (the limit applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

In addition to the boiling transition limit (MCPR = 1.05) operation is constrained to a maximum LHGR 13.4 Kw/ft. At 100% power this limit is reached with a maximum total peaking factor (MTPF) of 2.481. For the case of MTPF exceeding 2.481, operation is permitted only at less than 100% of rated thermal power and only with reduced APRM scram settings as required by specification 2.1.A.1.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of $28x10^3$ lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than $28x10^3$ lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

For the fuel in the core during periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If water level

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Min. No. of Operable Inst. Channels Per Trip System (1)	Trip Function	Trip Level Setting	Shut-down	Modes in Which Function Must Be Operable: Refuel (7)	Startup/Hot Standby	Run	Action (1)
4	Turbine Stop Valve Closure	≤ 10% Valve Closure		X(4)	X(4)	X(4)	1.A or 1.D
2	Turbine Control Valve - Loss of Control Pressure	≥ 550 psig		X(4)	X(4)	X(4)	1.A or 1.D
2	Turbine First Stage Pressure Permissive	≤ 154 psig		X(18)	X(18)	X(18)	(19)
2	Turbine Condenser Low Vacuum	≥ 23 In. Hg, Vacuum		X(3)	X(3)	X	1.A or 1.C
2	Main Steam Line High Radiation (14)	≤ 3X Normal Full Power Background (20)		X(9)	X(9)	X(9)	1.A or 1.C

Amendment No. 3

12. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.

13. Less than 14 operable LPRM's will cause a trip system trip.

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

15. The APRM 15% scram is bypassed in the Run Mode.

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system.

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

18. Operability is required when reactor thermal power is below 30% (high-pressure turbine first-stage pressure (≤ 154 psig).

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required.

20. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in the primary coolant.

closure, turbine stop valve closure and loss of condenser vacuum are discussed in Specification 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds three times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent release of radioactive material to the turbine. An alarm is initiated whenever the radiation level exceeds 1.5 times normal background to alert the operator to possible serious radioactivity spikes due to abnormal core behavior. The air ejector off-gas monitors serve to back up the main steam line monitors to provide further assurance against release of radioactive materials to site environs by isolating the main condenser off-gas line to the main stack.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. Section 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system (120/125 scram) in conjunction with the APRM system (15% scram) provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The discharge volume tank accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 50 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharge water and precludes the situation in

3.3 REACTIVITY CONTROL

2. Reactivity margin inoperable control rods

a. Control rod drives which cannot be moved with control drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to the Cold Shutdown condition within 24 hours and shall not be started unless (1) investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.2.c.

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically.

4.3 REACTIVITY CONTROL

2. Reactivity margin - inoperable control rods

a. Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week when operating above 30% power. In the event power operation is continuing with three or more inoperable control rods, this test shall be performed at least once each day, when operating above 30% power.

b. A second licensed operator shall verify the conformance to Specification 3.3.A.2.d before a rod may be bypassed in the Rod Sequence Control System.

3.3 REACTIVITY CONTROL

b. During the shutdown procedure no rod movement is permitted between the testing performed above 20% power and the reinstatement of the RSCS restraints at or above 20% power. Alignment of rod groups shall be accomplished prior to performing the tests.

c. Whenever the reactor is in the startup or run modes below 20% rated power the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.

4.3 REACTIVITY CONTROL

a. The capability of the RSCS to properly fulfill its function shall be verified by the following tests:

Sequence portion - Select a sequence and attempt to withdraw a rod in the remaining sequences. Move one rod in a sequence and select the remaining sequences and attempt to move a rod in each. Repeat for all sequences.

Group notch portion - For each of the six comparator circuits go through test initiate; comparator inhibit; verify; reset. On seventh attempt test is allowed to continue until completion is indicated by illumination of test complete light.

b. The capability of the Rod Worth Minimizer (RWM) shall be verified by the following checks:

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

G. Automatic Depressurization System (ADS)

1. Five of the six valves of the Automatic Depressurization System shall be operable:

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

(2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except as specified in 3.5.G.2 and 3.5.G.3 below.

2. If more than one ADS valve is known to be incapable of automatic operation, the reactor may remain in operation for a period not to exceed 7 days, provided the HPCI system is operable. (Note that the pressure relief function of these valves is assured by section 3.6.D of these specifications and that this specification only applies to the ADS function.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

G. Automatic Depressurization System (ADS)

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

a. A simulated automatic actuation test shall be performed prior to startup after each refueling outage. Manual surveillance of the relief valves is covered in 4.6.D.2.

2. When it is determined that more than one ADS valve is incapable of automatic operation, the HPCIS shall be demonstrated to be operable immediately and daily thereafter as long as Specification 3.5.G.2 applies.

taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS.

With one ADS valve known to be incapable of automatic operation, five valves remain operable to perform their ADS function. The ECCS loss-of-coolant accident analyses for small line breaks assumed that five of the six ADS valves were operable. Reactor operation with more than one ADS valve inoperable is only allowed to continue for seven days provided that the HPCI system is demonstrated to be operable immediately and daily thereafter.

H. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI, HPCIS, and RCICS are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in an operable condition. If a discharge pipe is not filled, the pumps that supply that line must be assumed to be inoperable for Technical Specification purposes.

The core spray and RHR system discharge piping high point vent is visually checked for water flow once a month prior to

3.6 PRIMARY SYSTEM BOUNDARY

H. Shock Suppressors (Snubbers)

1. During all modes of operation except Cold Shutdown and Refuel, all safety-related snubbers shall be operable except as noted in 3.6.H.2 through 3.6.H.5 below.

4.6 PRIMARY SYSTEM BOUNDARY

H. Shock Suppressors (Snubbers)

The following surveillance requirements apply to all hydraulic snubbers listed in 3.6.H.2.

1. All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections, and linkage connections to the piping and anchor to verify their operability in accordance with the following schedule:

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval	
0	Operating Cycle	±25%
1	12 months	±25%
2	6 months	±25%
3,4	124 months	±25%
5,6,7	62 days	±25%
≥8	31 days	±25%

The required inspection interval shall not be lengthened more than one step at a time.

3.6 PRIMARY SYSTEM BOUNDARY

2. The safety-related snubbers listed in Table 3.6.H are required to protect the primary coolant system or other safety related systems or components and are therefore subject to these specifications.

3. From and after the time that a safety-related snubber is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable or replaced.

4.6 PRIMARY SYSTEM BOUNDARY

Snubbers may be categorized in two groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

2. All hydraulic snubbers whose seal materials are other than ethylene propylene or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.

3. The initial inspection shall be performed within 6 months from the date of issuance of these specifications. For the purpose of entering the schedule in Specification 4.6.H.1, it shall be assumed that the facility had been on a 6 month inspection interval.

3.6 PRIMARY SYSTEM BOUNDARY

4. If the requirements of 3.6.H.1 and 3.6.H.3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 36 hours.

5. If a safety-related snubber is determined to be inoperable while the reactor is in the shutdown or refuel mode, the snubber shall be made operable prior to reactor startup.

6. Snubbers may be added to safety-related systems without prior license amendment to Table 3.6.H provided that a revision to Table 3.6.H is included with a subsequent license amendment request.

4.6 PRIMARY SYSTEM BOUNDARY

4. Once each refueling cycle, a representative sample of 10 snubbers or approximately 10% the snubbers, whichever is less, shall be functionally tested for operability including verification of proper piston movement, lock up and bleed. For each unit and subsequent unit found inoperable, an additional 10% or ten snubbers shall be so tested until no more failures are found or all units have been tested. Snubbers of rated capacity greater than 50,000 lb need not be functionally tested.

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SSA1(Z)	Main Steam A	585			X	
SSA2(X)	Main Steam A	585			X	
SSB1(Z)	Main Steam B	585			X	
SSB2(X)	Main Steam B	585			X	
SSB4(Z)	Main Steam B	585			X	
SSB5(Y)	Main Steam B	585			X	
SSB6(X)	Main Steam B	585			X	
SSC1(Z)	Main Steam C	585			X	
SSC2(X)	Main Steam C	585			X	
SSC4(Z)	Main Steam C	585			X	
SSC5(Y)	Main Steam C	585			X	
SSC6(X)	Main Steam C	585			X	
SSD1(Z)	Main Steam D	585			X	
SSD2(X)	Main Steam D	585			X	
SSA1(X)	Feedwater A	601			X	

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SSA2(Z)	Feedwater A	601			X	
SSA3(Y)	Feedwater A				X	
SSA4(Z)	Feedwater A				X	
SSA5(X)	Feedwater A				X	
SSA6(Z)	Feedwater A				X	
SSA7(Z)	Feedwater A				X	
SSA8(X)	Feedwater A				X	
SSA9(Z)	Feedwater A				X	
SSB1(X)	Feedwater B				X	
SSB2(Z)	Feedwater B				X	
SSB3(Y)	Feedwater B				X	
SSB4(Z)	Feedwater B				X	
SSB5(X)	Feedwater B				X	
SSB6(Z)	Feedwater B				X	
SSB7(Z)	Feedwater B				X	

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SSB8(X)	Feedwater B				X	
SSB9(Z)	Feedwater B				X	
R10	RHR	555				X
R12 upper	RHR	550				X
R12 lower	RHR	550				X
R19	RHR	555				X
R20 upper	RHR	549				X
R20 lower	RHR	549				X
R41 outside	RHR	555				X
R41 inside	RHR	555				X
R51 - north	RHR	536				X
R51 - south	RHR	536				X
R52 - west	RHR	559				X
R52 - east	RHR	559				X
R53 - north	RHR	540				X

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
R53 - east	RHR	540				X
R54	RHR	531				X
R55 - west	RHR	536				X
R55 - north	RHR	536				X
R56	RHR	535				X
R57 - west	RHR	536				X
R57 - east	RHR	536				X
R58 - north	RHR	576				X
R58 - south	RHR	576				X
R59	RHR	571				X
R60 - east	RHR	572				X
R60 - west	RHR	572				X
R61 upper	RHR	598				X
R61 lower	RHR	598				X
R62 - north	RHR	598				X

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
R62 - south	RHR	598				X
R62	RHR	598				X
R62 - east	RHR	572				X
R64 - west	RHR	572				X
R65	RHR	573				X
R67	RHR	581				X
R68	RHR	579				X
R69	RHR	575				X
R77	RHR	578				X
R72	RHR head spray	630			X	
R73	RHR head spray	636			X	
R73	RHR head spray	636			X	
R73	RHR head spray	636			X	
R73	RHR head spray	636			X	
R75	RHR head spray	648	X	X	X	

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
R1	Control Rod drive	612			X	
R2	Control rod drive	612			X	
R1	Core spray	606			X	
R2	Core spray	606			X	
R6 - North-west	Core spray	544				X
R6 - South-west	Core spray	544				X
R8	Core spray	609			X	
R9	Core spray	609			X	
R13 - North-east	Core spray	544				X
R13 - South-east	Core spray	544				X
R19	Standby liquid control	624			X	

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TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
R21	Standby liquid control	624				X
R6	HPCI	563			X	
R31A	HPCI	543				X
R31B	HPCI	543				X
R90	HPCI	540				X
R91 - north	HPCI	538				X
R91 - south	HPCI	538				X
R4 - north	RCIC	528				X
R4 - south	RCIC	528				X
R5 - east	RCIC	538		X		X
R5 - south	RCIC	538				X
R7 - east (upper)	RCIC	548				X
R7 - west (lower)	RCIC	548				X
R9 - north	RCIC	564			X	

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
R9 - south	RCIC	564			X	
R1 upper	Condensate S&S (ring header)	548				X
R1 lower	Condensate S&S (ring header)	548				X
R2 - north	Condensate S&S (ring header)	548				X
R2 - west	Condensate S&S (ring header)	548				X
R3 - east	Condensate S&S (ring header)	548				X
R3 - west	Condensate S&S (ring header)	548				X
R4 - north	Condensate S&S (ring header)	548		X		X
R4 - east	Condensate S&S (ring header)	548		X		
R5 upper	Condensate S&S (ring header)	548		X		

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
R5 lower	Condensate S&S (ring header)	548				X
SSX-1	PSC (ring hdr)	525				X
SSZ-2	PSC (ring hdr)	525				X
SSX-3	PSC (ring hdr)	525				X
SSZ-4	PSC (ring hdr)	525				X
SSZ-5	PSC (ring hdr)	525				X
SSX-6	PSC (ring hdr)	525				X
SSX-7	PSC (ring hdr)	525				X
SSZ-8	PSC (ring hdr)	525				X
SSX-1A	PSC (ring hdr)	525				X
SSZ-2A	PSC (ring hdr)	525				X
SSX-3A	PSC (ring hdr)	525				X
SSZ-4A	PSC (ring hdr)	525				X
SSZ-5A	PSC (ring hdr)	525				X
SSX-6A	PSC (ring hdr)	525				X

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SSX-7A	PSC (ring hdr)	525				X
SSZ-8A	PSC (ring hdr)	525				X
R2 upper	Condensate bypass line	557				X
R2 lower	Condensate bypass line	557				X
R9	Condensate bypass line	557				X
R13 - east	Condensate bypass line	557				X
R13 - west	Condensate bypass line	557				X
R24	EECW	605			X	
SS1-A	Recirculation	556			X	
SS1-B	Recirculation	556			X	
SS2-A	Recirculation	558			X	
SS2-B	Recirculation	558			X	
SS3-A(295°)	Recirculation	564			X	

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SS3-A(335°)	Recirculation	564			X	
SS3-B(115°)	Recirculation	564			X	
SS3-B(154°)	Recirculation	564			X	
SS4-A	Recirculation	570			X	
SS4-B	Recirculation	570			X	
SS5-A(262°)	Recirculation	581			X	
SS5-A(325°)	Recirculation	581			X	
SS5-B(35°)	Recirculation	581			X	
SS5-B(98°)	Recirculation	581			X	
SS6-A	Recirculation	568			X	
SS6-B	Recirculation	568			X	
SS7	Recirculation	564			X	
SS8	Recirculation	564			X	

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3.6/4.6 BASES

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in Section 4.6.G.4 to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgment from actual plant observation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in three additional examinations above the requirements of Section XI of ASME Code.

REFERENCES

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)
2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code
3. ASME Boiler and Pressure Vessel Code, Section III (1968 edition)
4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 edition)

3.6.H/4.6.H Shock Suppressors (Snubbers)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all hydraulic snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during relatively low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the

3.6/4.6 BASES

These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Ten percent or ten snubbers, whichever is less, represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. Those snubbers designated in Table 3.6.H as being in high radiation areas or especially difficult to remove need not be selected for functional tests provided operability was previously verified.

Snubbers of rated capacity greater than 50,000 lb are exempt from the functional testing requirements because of the impracticability of testing such large units.

Amendment No. 3

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2.

   a. Minimum water volume - 123,000 ft^3

   b. Maximum water volume - 135,000 ft^3

Amendment No. 3

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

1. Pressure Suppression Chamber

   a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

3.7 CONTAINMENT SYSTEMS

c. With the suppression pool water temperature > 95°F initiate pool cooling and restore the temperature to ≤ 95°F within 24 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

d. With the suppression pool water temperature > 105°F during testing of ECCS or relief valves, stop all testing, initiate pool cooling and follow the action in specification 3.7.A.1.c above.

e. With the suppression pool water temperature > 120°F following reactor isolation, depressurize to < 200 psig at normal cool-down rates.

f. With the suppression pool water temperature > 110°F during startup or power operation the reactor shall be scrammed.

4.7 CONTAINMENT SYSTEMS

3.7 CONTAINMENT SYSTEMS

5. Primary containment pressure shall be limited to a maximum of 30 psig during repressurization following a loss of coolant accident.

4.7 CONTAINMENT SYSTEMS

TABLE 3.7.D
PRIMARY CONTAINMENT ISOLATION VALVES

Valves	Valve Identification	Test Medium	Test Method
64-(ck)	Suppression Chamber vacuum relief	Air (1)	Applied between 64-20 and 64-(ck)
64-21	Suppression Chamber vacuum relief	Air (1)	Applied between 64-21 and 64-(ck)
64-(ck)	Suppression Chamber vacuum relief	Air (1)	Applied between 64-21 and 64-(ck)
64-29	Drywell main exhaust	Air (1)	Applied between 64-29, 64-30, 64-32, 64-33, and 84-19.
64-30	Drywell main exhaust	Air (1)	Applied between 64-29, 64-30, 64-32, 64-33, and 84-19.
64-31	Drywell exhaust to Standby Gas Treatment	Air (1)	Applied between 64-31, 64-141, 84-20, and 64-140
64-32	Suppression Chamber Main Exhaust	Air (1)	Applied between 64-32, 64-33, 64-29, 64-30, and 84-19.
64-33	Suppression Chamber Main	Air ()	Applied between 64-32, 64-33, 64-29, 64-30, and 84-19.
64-34	Suppression Chamber to Standby Gas Treatment	Air (1)	Applied between 64-34, 64-141, and 64-139
69-1	RWCU Supply	Water (2)	Applied between 69-1, 69-500, and 10-505
69-2	RWCU Supply	Water (2)	Applied between 69-2, 69-500, and 10-505

TABLE 3.7.D
PRIMARY CONTAINMENT ISOLATION VALVES

Valves	Valve Identification	Test Medium	Test Method
76-239	Containment Atmospheric Monitor	Air	Applied between 76-239 and 76-240
76-242	Containment Atmospheric Monitor	Air	Applied between 76-242 and 76-244
76-243	Containment Atmospheric Monitor	Air	Applied between 76-243 and 76-244
76-248	Containment Atmospheric Monitor	Air	Applied between 76-248 and 76-251
76-250	Containment Atmospheric Monitor	Air	Applied between 76-250 and 76-251
76-253	Containment Atmospheric Monitor	Air	Applied between 76-253 and 76-255
76-254	Containment Atmospheric Monitor	Air	Applied between 76-254 and 76-255
84-20	Main Exhaust to Standby Gas Treatment	Air (1)	Applied between 84-20, 64-141, 64-140, and 64-31
84-600	Main Exhaust to Standby Gas Treatment	Nitrogen (1)	Applied between 84-8A and 84-600
84-601	Main Exhaust to Standby Gas Treatment	Nitrogen (1)	Applied between 84-8B and 84-601
84-602	Main Exhaust to Standby Gas Treatment	Nitrogen	Applied between 84-8C and 84-603

TABLE 3.7.D
PRIMARY CONTAINMENT ISOLATION VALVES

Valves	Valve Identification	Test Medium	Test Method
84-603	Main Exhaust to Standby Gas Treatment	Nitrogen	Applied between 84-8D and 84-602
64-141	Drywell pressurization, Compressor bypass	Air (1)	Applied between 64-141, 64-140, 64-31 and 84-20
64-140	Drywell pressurization, Compressor discharge	Air (1)	Applied between 64-141, 64-140, 64-31, and 84-20
64-139	Drywell pressurization, Compressor suction	Air (1)	Applied between 64-139, 64-141, and 64-34

(1) Air/nitrogen test to be displacement flow
(2) Water test to be injection loss or downstream collection.

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 of 170°F which is sufficient for complete condensation. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regine of potentially high suppression chamber loadings.

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 assures adequate margin for controlled blowdown anytime during RCIC operation and assures 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 from 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 pressure, 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.

The licensee will provide a plant unique analysis as discussed in SER Supplement No. 8 and the TVA letter of May 17, 1976. The results of that analysis will determine the need to continue operation with a differential pressure of 1.0 psid between the drywell and suppression chamber. In the interim, operation with this differential pressure provides a factor of safety of about 2 on structural loads.

Inerting

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent or so) reaction of the zirconium and steam during a loss-of-coolant accident could lead to the liberation of hydrogen combined with an air atmosphere to result in a flammable concentration in the containment. If a sufficient amount of hydrogen is generated and oxygen is available in stoichiometric quantities, the subsequent ignition of the hydrogen in rapid recombination rate could lead to failure of the containment to maintain low leakage integrity. The 4% oxygen concentration minimizes the possibility of hydrogen combustion following a loss-of-coolant accident.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

A. The plant superintendent has on-site responsibility for the safe operation of the facility and shall report to the Chief, Nuclear Generation Branch. In the absence of the plant superintendent, the assistant superintendnet will assume his responsibilities.

B. The portion of TVA management which relates to the operation of the plant is shown in Figure 6.1-1.

C. The functional organization for the operation of the station shall be as shown in Figure 6.1-2.

D. Shift manning requirements shall, as a minimum, be as described in section 6.8.

E. Qualifications of the Browns Ferry Nuclear Plant management and operating staff shall meet the minimum acceptable levels as described in ANSI - N18.1, Selection and Training of Nuclear Power Plant Personnel, dated March 8, 1971.

F. Retraining and replacement training of station personnel shall be in accordance with ANSI - N18.1, Selection and Training of Nuclear Power Plant Personnel, dated March 8, 1971. The minimum frequency of the retraining program shall be every two years.

G. An Industrial Security Program shall be maintained for the life of the plant.

H. Responsibilities of a post-fire overall restoration coordinator will consist of duties as described in section 6.9.

I. The Safety Engineer shall have the following qualifications:

   a. Must have a sound understanding and thorough technical knowledge of safety and fire protection practices, procedures, standards and other codes relating to electrical utility operations. Must be able to read and understand engineering drawings.

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Must possess an analytical ability for problem solving and data analysis. Must be able to communicate well both orally and in writing and must be able to write investigative reports and prepare written procedures. Must have the ability to secure the cooperation of management, employees and groups in the implementation of safety programs. Must be able to conduct safety presentations for supervisors and employees.

b. Should have experience in safety engineering work at this level or have 3 years experience in safety and/or fire protection engineering. It is desirable that the incumbent be a graduate of an accredited college or university with a degree in industrial, mechanical, electrical or safety engineering or fire protection engineering.

6.2 Review and Audit

The Manager of Power is responsible for the safe operation of all TVA power plants, including the Browns Ferry Nuclear Plant. The functional organization for Review and Audit is shown in Figure 6.2-1.

Organizational units for the review of facility operation shall be constituted and have the responsibilities and authorities listed below.

A. Nuclear Safety Review Board (NSRB)

6.0 ADMINISTRATIVE CONTROLS

1. Membership

The NSRB shall consist of a chairman and at least five other members appointed or approved by the Manager of Power. A majority of the members shall be independent of the Division of Power Production. The qualifications of members shall meet the requirements of ANSI Standard N18.7-1972. Membership shall include at least one outside consultant and representatives of the following TVA organizations: Office of Engineering Design and Construction; Division of Environmental Planning; Division of Power Production; Division of Power Resource Planning. An alternate chairman may be designated by the chairman or, in his absence or incapacity, may be selected by the NSRB. The NSRB chairman shall appoint a secretary.

2. Minimum Meeting Frequency

The NSRB shall meet at least quarterly and at more frequent intervals at the call of the chairman, as required.

3. Quorum

A quorum shall consist of four members, a minority of which shall be from the Division of Power Production.

4. Responsibilities

a. Review proposed tests and experiments, and their results, when such tests or experiments may constitute an unreviewed safety question as defined in Section 50.59, Part 50, Title 10, Code of Federal Regulations.

b. Review proposed changes to equipment, systems or procedures, which are described in the Final Safety Analysis Report or which may involve an unreviewed safety question, as defined in Section 50.59, Part 50, Title 10, Code of Federal Regulations, or which are referred by the operating organization.

c. Review proposed changes to Technical Specifications or licenses.

d. Review violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having safety significance.

e. Review signficant operating abnormalities or deviations from normal and expected performance of plant equipment.

f. Review reportable occurrences, as defined in the Technical Specifications.

g. Review information received indicating that there may be an unanticipated deficiency in some aspect of design or operation of safety-related systems or components.

h. Review the reports of annual audits of plant operation to verify that operation complies with the terms, conditions and intent of any license, permit, or other applicable regulations.

i. Review the minutes of Plant Operations Review Committee meetings to determine if matters considered by that committee involve unreviewed or unresolved safety questions.

5. Authority

The Nuclear Safety Review Board shall be advisory to the Manager of Power in matters relating to nuclear plant safety.

The Nuclear Safety Review Board shall have access to all TVA nuclear facilities, as well as design, construction, and operating records as necessary to perform it assigned functions.

Members have access to advice and services of technical specialists within their respective organizations and outside consulting services are available as required through contractual arrangements.

6. Records

The chairman shall prepare a final copy of the minutes and forward them to the Manager of Power.

4. Duties and Responsibilities

The PORC serves in an advisory capacity to the plant superintendent and as an investigating and reporting body to the Nuclear Safety Review Board in matters related to safety in plant operations. The plant superintendent has the final responsibility in determining the matters that should be referred to the Nuclear Safety Review Board.

The responsibility of the committee will include:

a. Review all standard and emergency operating and maintenance instructions and any proposed revisions thereto, with principal attention to provisions for safe operation.

b. Review porposed changes to the Technical Specifications.

c. Review proposed changes to equipment or systems having safety significance, or which may constitute "an unreviewed safety question," pursuant to 10 CFR 50.59.

d. Investigate reported or suspected incidents involving safety questions, violations of the Technical Specifications, and violations of plant instructions pertinent to nuclear safety.

e. Review reportable occurrences, unusual events, operating anomalies and abnormal performance of plant equipment.

f. Maintain a general surveillance of plant activities to identify possible safety hazards.

g. Review plans for special fuel handling, plant maintenance, operations, and tests or experiements which may involve special safety considerations, and the results thereof, where applicable.

h. Review adequacy of quality assurance program and recommend any appropriate changes.

i. Review implementating procedures of the Radiological Emergency Plan and the Industrial Security Program on an annual basis.

j. Review adequacy of employee training programs and recommend change.

5. Authority

The PORC shall be advisory to the plant superintendent.

6. Records

Minutes shall be kept for all PORC meetings with copies sent to Director, Power Production; Chief, Nuclear Generation Branch; Chairman, NSRB.

7. Procedures

Written administrative procedures for committee operation shall be prepared and maintained describing the method for submission and content of presentations to the committee, review and approval by members of committee actions, dissemination of minutes, agenda and scheduling of meetings.

C. Quality Assurance and Audit Staff

The Office of Power Quality Assurance and Audit Staff (QA&AS) shall formally audit operation of the nuclear plant. Audits of selected aspects of plant operations shall be conducted on a frequency commensurate with their safety significance and in such a manner as to assure that an audit of safety-related activities is completed within a period of two years.

The audits shall be performed in accordance with appropriate written instructions or procedures and should include verification of compliance with internal rules, procedures (for example, normal off/normal, emergency, operating, maintenance, surveillance, test, security, and radiation control procedures and the emergency plan), regulations, and license provisions; training, qualification, and performance of operating staff; and corrective actions following reportable occurrences.

6.4 Actions to be Taken in the Event of a Reportable Occurrence in Plant Operation (Ref. Section 6.7)

A. Any reportable occurrence shall be promptly reported to the Chief, Nuclear Generation Branch and shall be promptly reviewed by PORC. This committee shall prepare a separate report for each reportable occurrence. This report shall include an evaluation of the cause of the occurrence and recommendations for appropriate action to prevent or reduce the probability of a repetition of the occurrence.

B. Copies of all such reports shall be submitted to the Chief, Nuclear Generation Branch, the Manager of Power, the Division of Power Resource Planning, and the Chairman of the NSRB for their review.

C. The plant superintendent shall notify the NRC as specified in Specification 6.7 of the circumstances of any reportable occurrence.

6.5 Action to be Taken in the Event a Safety Limit is Exceeded

If a safety limit is exceeded, the reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC. A prompt report shall be made to the Chief, Nuclear Generation Branch and the Chairman of the NSRB. A complete analysis of the circumstances leading up to and resulting from the situation, together with recommendations to prevent a recurrence, shall be prepared by the PORC. This report shall be submitted to the Chief, Nuclear Generation Branch, the Manager of Power, the Division of Power Resource Planning, and the NSRB. Notification of such occurrences will be made to the NRC by the plant superintendent within 24 hours.

6.6 Station Operating Records

A. Records and/or logs shall be kept in a manner convenient for review as indicated below:

1. All normal plant operation including such items as power level, fuel exposure, and shutdowns

2. Principal maintenance activities

3. Reportable occurrences

6.7 Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.

1. Routine Reports

a. Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commerical power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

b. Annual Operating Report.[1] Routine operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following intial criticality.

6.0 ADMINISTRATIVE CONTROLS

The annual operating reports made by licensees shall provide a comprehensive summary of the operating experience gained during the year, even though some repetition of previously reported information may be involved. References in the annual operating report to previously submitted reports shall be clear.

Each annual operating report shall include:

(1) A narrative summary of operating experience during the report period relating to safe operation of the facility, including safety-related maintanance not covered in item 1.b.(2)(e) below.

(2) For each outage or forced reduction in power[2] of over twenty percent of design power level where the reduction extends for greater than four hours:

(a) the proximate cause and the system and major component involved (if the outage or forced reduction in power involved equipment malfunction);

(b) A brief discussion of (or reference to reports of) any reportable occurrences pertaining to the outage of power reduction;

(c) corrective action taken to reduce the probability of recurrence, if appropriate;

(d) operating time lost as a result of the outage or power reduction (for scheduled or forced outages,[3] use the generator off-line hours; for forced reductions in power, use the approximate duration of operation at reduced power);

(e) a description of major safety-related corrective maintenance performed during the outage or power reduction, including the outage or power reduction, including the system and component involved and identification of the critical path activity dictating the length of the outage or power reduction; and

(f) a report of any single release of radioactivity or radiation exposure specifically associated with the outage which accounts for more than 10% of the allowable annual values.

(3) A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,* e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

(4) Indications of failed fuel resulting from irradiated fuel examinations, including eddy current tests, ultrasonic tests, or visual examinations completed during the report period.

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate Regional Office, to be submitted no later than the tenth of each month following the calendar month covered by the report.

2. Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

(9) Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less than conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

Note: This item is intended to provide for reporting of potentially generic problems.

b. Thirty-Day Written Reports. The reportable occurrences discussed below shall be the subject of written reports to the Director of the appropriate Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in items 2.b.(1) and 2.b.(2) need not be reported except where test results themselves reveal a degraded mode as described above.

(3) Observed inadequacies in the implementation of administrative or procedural controls which

6.0 ADMINISTRATIVE CONTROLS

6.8 Minimum Plant Staffing

The minimum plant staffing for monitoring and conduct of operations is as follows.

1. A licensed senior operator shall be present at the site at all times when there is fuel in the reactor.

2. A licensed operator shall be in the control room whenever there is fuel in the reactor.

3. A licensed senior operator shall be in direct charge of a reactor refueling operation; i.e., able to devote full time to the refueling operation.

4. A health physics technician shall be present at the facility at all times there is fuel in the reactor.

5. Two licensed operators shall be in the control room during any cold startups, while shutting down the reactor, and during recovery from unit trip.

6. Either the plant superintendent or the assistant plant superintendent shall have acquired the experience and training normally required for examination by the NRC for a Senior Reactor Operator's License, whether or not the examination is taken. In addition, either the operations supervisor or the assistant operations supervisor shall have an SRO license.

6.9 Overall Restoration Coordinator

An overall restoration coordinator has been appointed and designated the responsibility of overseeing the entire restoration activity of units 1 and 2. The restoration activity is a result of the cable fire which took place on March 22, 1975.

Responsibilities of the restoration coordinator for the overall Browns Ferry Nuclear Plant include the following:

1. Principal coordinator for all design, construction, and operational activities relative to restoration of units 1 and 2 and fire related improvements to the plant.

2. Review and approval of all documents submitted by TVA to NRC in connection with restoration return to service of units 1 and 2 and fire protection and prevention improvements to the plant.

3. Responsible for overall planning, establishment, and maintenance of critical path schedule for restoration of units 1 and 2 and fire protection and prevention improvements to the plant.

4. Coordination and approval of TVA's overall efforts in fire protection and prevention improvements, including design and installation of new systems and changes necessary in fire fighting methods and techniques.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 3 TO FACILITY LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT UNIT 3

DOCKET NO. 50-296

FEBRUARY 15, 1977

1.0 INTRODUCTION

The radiological safety Technical Specifications for the Browns Ferry Nuclear Plant Unit 3 were initially issued as Appendix A to the facility operating license DPR-68 on July 2, 1976. Amendment 2 to the facility operating license deleted all of the temporary restrictions on activities that were previously in effect and which were contained on page 1a of these Technical Specifications. At this time there are no temporary restrictions in effect, and the body of the Technical Specifications has remained unchanged since its initial issue date.

On August 20, 1976, the NRC authorized a return to power of the Browns Ferry Nuclear Plant Units 1 and 2 after restoration and modifications resulting from the fire of March 22, 1975. The license amendments authorizing that return to power included extensive revisions to the Appendix A Technical Specifications attached to the respective Unit 1 and 2 facility operating license. That action identified several items in the Unit 3 Technical Specifications which are different in statement from those adopted in the Unit 1 and 2 Technical Specifications. The differences related to minor additions to the plant staffing and procedures for fire protection, and to a uniformity of statement for conditions as adopted for other operating power reactors.

The purpose of Amendment 3 to Facility Operating License DPR-68, and this Safety Evaluation, is to update the Appendix A Technical Specifications for Unit 3 to provide a uniformity of statement with the Technical Specifications in effect for the two other units of the plant, and to correct identified errors of understanding, omission, designation, grammar and spelling.

In addition, the licensee has requested a modification to certain valves, a part of the containment isolation system, which will be used to maintain a pressure differential between the drywell and torus atmospheres. This amendment also addresses this modification and the associated changes to the Technical Specifications.

The items changed by this amendment, and their respective Technical Specification sections are:

(1) Main steam line radiation alarm (Section 3.1)

(2) Automatic Depressurization System (Sections 3.5G and 4.5G)

(3) Hydraulic snubbers (Sections 3.6H and 4.6H)

(4) Suppression pool temperature limits (Sections 3.7A and 4.7A)

(5) Modifications to primary containment isolation valves (Section 3.7.D)

(6) Containment Atmosphere Dilution (CAD) System pressure limits (Section 3.7.G)

(7) Fire protection staffing and procedures (Sections 6.1 and 6.9)

(8) Miscellaneous corrections of errors of omission, designation, grammar or spelling (All Sections)

2.0 EVALUATION OF REVISIONS TO THE TECHNICAL SPECIFICATIONS

2.1 Main Steam Line High Radiation Alarm

The Appendix A Technical Specifications for Browns Ferry Nuclear Plant Units 1 and 2 included a requirement for the reactor protection instrumentation to provide an alarm in the control room when the main steam line high radiation monitors detect a radiation level greater than 1.5 times the normal full power background. The purpose of this alarm is to alert the operator to possible serious radioactivity spikes due to abnormal core behavior. This alarm is in addition to the reactor scram trip and main steam line isolation trip initiated by these monitors upon detection of a radiation level equal or greater than 3 times normal background. The air ejector off-gas monitors serve to back up the main steam line monitors to provide further assurance against the release of radioactive materials to site environs by isolating the main condenser off-gas line to the main stack.

Consistent with the objective of achieving uniformity in the Technical Specifications for the three units, this revision provides a requirement for the same alarm in the reactor protection instrumentation to the Unit 3 Technical Specifications. The inclusion of this alarm provides further assurance that the operator is made cognizant of abnormal core behavior should the condition develop. The addition in no way lessens or reduces the requirements for protective instrumentation previously required by the Technical Specifications. This revision is indicated on pages 33, 35 and 42 in Section 3.1 of the Appendix A Technical Specifications.

2.2 Automatic Depressurization System (ADS)

By letter dated October 12, 1976, the licensee requested an amendment to the Appendix A Technical Specifications to reduce the number of Automatic Depressurization System (ADS) valves which are required to be operable. These proposed changes are the result of information from the General Electric Company that five of the six installed ADS valves were assumed to be operational in the ECCS Appendix K analysis of the small break loss-of-coolant analysis. The initial issue of the Appendix A Technical Specifications was based on the understanding that the ECCS Appendix K analysis was performed assuming all six of the installed ADS valves operated. Since that understanding was not correct and the licensee has now shown that the ADS valves satisfy the ECCS Appendix K requirements with one valve failed, we conclude that extended reactor operation with one ADS valve inoperable is acceptable. Operation for a limited time (seven days) with more than one ADS valve inoperable is similarly acceptable on the basis that the ADS is redundant to the HPCI system. We conclude that this change to the Technical Specifications is appropriate as it corrects a misunderstanding and does not in any way result in a reduction in the safety margins previously believed to exist. This revision is indicated on pages 161 and 175 in Section 3.5 and 4.5 of the Technical Specifications.

2.3 Shock Suppressors (Snubbers)

This revision to Section 3.6B and 4.6B of the Unit 3 Appendix A Technical Specifications makes five changes for the purpose of achieving uniformity in the Technical Specifications for the three units of the Browns Ferry Nuclear Plant. These changes are:

(1) Adoption of the more general title "shock suppressors" to include snubbers other than hydraulic snubbers.

(2) Base the schedule for the next required inspection interval on a specified number of snubbers found inoperable, rather than a specified percent of snubbers found inoperable.

(3) Identify ethylene propylene as a seal material known to be compatible with the operating environment.

(4) Increase the number of snubbers to be functionally tested once each refueling cycle, and excuse snubbers or rated capacity greater than 50,000 pounds from this functional test.

(5) Clarify our intent to permit the addition of snubbers to safety-related systems prior to a license amendment to Table 3.6.H.

The above changes provide uniform surveillance requirements among the three units for plant hydraulic snubbers. The above changes do not reduce, but rather increase, the surveillance requirements for hydraulic snubbers installed in Unit 3. Items (2) through (4) are also consistent with the surveillance requirements for hydraulic snubbers given in the Standard Technical Specifications for General Electric Boiling Water Reactors, Revision of August 15, 1975, NUREG-0123. The change in Item (5) deletes for clarification a portion of the limiting condition for operation 3.6.H.6. The intent of 3.6.H.6 was to permit the addition of snubbers to safety-related systems prior to a license amendment; but, to require that these additions be supported by safety evaluations, documentation and reporting requirements in a subsequent license amendment request conforming to the requirements of 10 CFR 50.59 for amendment of the Technical Specifications. The intent of 3.6.H.6 remains unchanged.

This revision is indicated on pages iii, vii, 198-200, 209-219, 228 and 230 of the Technical Specifications.

2.4 Suppression Pool Temperature Limits

The Technical Specifications as initially issued specified as limiting conditions for operation the suppression pool temperature limits needed to preclude the steam quenching phenomenon in the event of an extended relief valve operation. This revision is made to clarify the required course of action in the event these temperature limits are exceeded. Each of these limiting conditions for operation are restated to provide a definite action on the part of the control operator and the times within which the temperature limits must be recovered or the reactor/containment combination placed in a condition which precludes the steam quenching phenomenon. The revision adopts the related limiting conditions for operation and surveillance requirements as stated in the Technical Specifications for Browns Ferry Nuclear Plant, Units 1 and 2.

This revision also includes the "Bases" for the above suppression pool temperature limits related to precluding the steam quenching phenomenon inadvertently omitted in the initial issue of the Technical Specifications.

The revision is indicated on pages 231, 232, 286 and 286A of the Technical Specifications.

2.5 Modifications to the Primary Containment Isolation Valves

Background

Our evaluation of the suppression pool hydrodynamic loads during a lost-of-coolant accident in Section 5.1 of Supplement No. 9 to the Safety Evaluation Report for Browns Ferry Nuclear Plant Units 1, 2 and 3 reported that the Tennessee Valley Authority will operate with a differential pressure between the drywell and the torus atmospheres and has begun procurement of the equipment required to install a nitrogen pressurization system. The piping modifications associated with the inclusion of the differential pressure control system result in the addition of three primary containment isolation valves.

Description of Proposed Modifications

The licensee has subsequently requested approval of certain changes of the Technical Specifications related to a proposed design modification to the smaller-sized containment purge lines, i.e., those lines used for a controlled purge of the drywell and torus. The modification provides for the removal of one existing valve (FSV 64-145) and the addition of three valves (FSV 64-139, FSV 64-140 and FSV 64-141) and associated piping.[(1)(2)] These modifications were requested to facilitate the maintenance of the pressure differential between the drywell and torus atmospheres. The service air system will be used to establish the required pressure differential for the present, and air from that system will be routed through one of the proposed valves. The modifications also provide for the installation of several valves and connections which will eventually be used in conjunction with a nitrogen recirculation compressor as part of a nitrogen pressurization system.

These valves also serve as outboard containment isolation barriers and provide the proper flow routing for establishing the required pressure differential.

Evaluation and Conclusions

The three valves to be added will serve as the redundant containment isolation valves and as such are designed to seismic Category I and Safety Class 2 criteria. Automatic isolation occurs upon the receipt of a reactor vessel low water level, high drywell pressure, or high reactor building exhaust radiation signal. These valves, controls, actuation logic and installation meet all the requirements of previously accepted criteria for Browns Ferry containment isolation valves. The valves will fail in the closed position upon a loss of power; valve position is indicated in the control room. Provisions have been made in the piping modifications to permit local leak testing of the isolation valves in accordance with Appendix J to 10 CFR 50.

The modifications to the containment isolation valves are designed such that this change will not interfere with the safety related features incorporated in the existing plant design. In addition, the proposed design change is in conformance with the applicable regulations, regulatory guides, and staff positions. Therefore, we find the proposed modifications acceptable.

This revision to the Technical Specifications revises Table 3.7.D to delete the periodic leakage testing requirements for the valve to be removed, and to add the periodic leak testing requirements for the three added valves. The changes to this table are indicated on pages 272, 277 and 278 of the Technical Specifications, and are acceptable.

References

(1) Tennessee Valley Authority letter (from H. Parris to B. Rusche) requesting a change to Table 3.7.D of the Technical Specifications, August 9, 1976.

(2) Tennessee Valley Authority letter (from J. Gilleland to B. Rusche) providing additional information in support of the requested change in the Technical Specifications, September 3, 1976.

2.6 Containment Atmosphere Dilution (CAD) System Pressure Limit

The Technical Specifications for Browns Ferry Units 1 and 2 issued on August 20, 1976, added a limiting condition for operation restricting the primary containment pressure to a maximum of 30 psig during repressurization following a loss-of-coolant accident.

This matter was previously reviewed and approved in Supplement No. 1 to the Safety Evaluation Report of the Browns Ferry Nuclear Plant Units 1, 2 and 3 issued in December 1972. Inclusion of this item in the Technical Specification represents a formalization of operating procedures previously reviewed and mutually agreed to by the licensee and the Commission's staff.

Consistent with the objective of providing uniformity in the Technical Specifications, this revision adds the same limiting condition for operation to the Unit 3 Technical Specifications. The revision is indicated on page 260A of the Technical Specifications.

2.7 Fire Protection Staff Qualifications and Responsibilities

The Technical Specifications for Browns Ferry Units 1 and 2 issued on August 20, 1976, contained a requirement for the Safety Engineer to have certain qualifications for the job. This individual and his duties are shown in Figure 6.3-1 of the Technical Specifications for Units 1 and 2 and for Unit 3. This revision to the Unit 3 Technical Specifications adds the same statement of qualifications for the Safety Engineer as given in the Unit 1 and 2 Technical Specifications. The change is indicated on pages 362 and 362A of the Technical Specifications.

The Unit 3 Technical Specifications issued on July 2, 1976, contained a Section 6.9 defining the responsibilities of the overall restoration coordinator. The section identifies these responsibilities as they relate to Units 1 and 2, but makes no mention of Unit 3. The omission was not intentional; the term restoration was taken to include improvements to the station fire protection and prevention features. This revision is made to clarify the intent of this section as it relates to Unit 3. The changes are indicated on pages 388 and 389 of the Technical Specifications.

2.8 Miscellaneous Corrections

In the course of our review of the Technical Specifications for the Browns Ferry Nuclear Plant, we have found a number of errors. These consist of errors of omission in the Bases, errors of incorrect designation of events and reports, errors of grammar, and errors of spelling. None of these are of any significance relative to requirements on the licensee or the plant operation. These changes are made to correct errors. The changes are indicated on pages 16, 119, 124, 363, 364, 366, 367, 376, 379, 380, 381 and 384 of the Technical Specifications.

3.0 ENVIRONMENTAL IMPACT

We have determined that this 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.

4.0 CONCLUSIONS

For the matters described and evaluated in Section 2.0 of this Safety Evaluation we have concluded that (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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 the health and safety of the public.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-296

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 3 to Facility Operating License No. DPR-68 issued to Tennessee Valley Authority which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit 3, located in Limestone County, Alabama. The amendment is effective as of its date of issuance.

This amendment revises the Technical Specifications (Appendix A) of the Browns Ferry Nuclear Plant, Unit 3 to provide a uniformity of statement with the Technical Specifications in effect for Units 1 and 2 of the Browns Ferry Nuclear Plant and to correct identified errors of understanding, omission, designation, grammar and spelling. In addition, the amendment addresses proposed modifications to certain valves, a part of the containment isolation system, which will be used to maintain a pressure differential between the drywell and torus atmospheres.

The amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) Amendment No. 3 to License No. DPR-68, and (2) the Commission's related Safety Evaluation dated February 15, 1977. These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Athens Public Library, South and Forrest, Athens, Alabama.

A copy of items (1) and (2) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Project Management.

Dated at Bethesda, Maryland, this 15th day of February, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Light Water Reactors
Branch No. 1
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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-296

TENNESSEE VALLEY AUTHORITY

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This amendment revises the Technical Specifications (Appendix A) of the Browns Ferry Nuclear Plant, Unit 3 to provide a uniformity of statement with the Technical Specifications in effect for Units 1 and 2 of the Browns Ferry Nuclear Plant and to correct identified errors of understanding, omission, designation, grammar and spelling. In addition, the amendment addresses proposed modifications to certain valves, a part of the containment isolation system, which will be used to maintain a pressure differential between the drywell and torus atmospheres.

The amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

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For further details with respect to this action, see (1) Amendment No. 3 to License No. DPR-68, and (2) the Commission's related Safety Evaluation dated February 15, 1977. These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Athens Public Library, South and Forrest, Athens, Alabama.

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Dated at Bethesda, Maryland, this 15th day of February, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by
John F. Stolz

John F. Stolz, Chief
Light Water Reactors
Branch No. 1
Division of Project Management

OFFICE ➤	DPM:LWR #2	ELD	DPM:LWR #1			
SURNAME ➤	S.B.Burwell SBBurwell:mt	AMitchell	JStolz			
DATE ➤	2/7/77	2/14/77	2/7/77			

February 14, 1977

Note to S. B. Burwell

AMENDMENT NO. 3, BROWNS FERRY OL

Standard Findings 1(A) and (B) have been omitted from this amendment. I assume this was done because those findings refer to an "application for amendment" which may not have been submitted covering all aspects of this amendment. In such case, the following finding should be inserted in the amendment:

1.A. The facility will operate in conformance with the provisions of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations set forth in 10 CFR Chapter I;

The Federal Register notice recites (third paragraph) that the application complies with the Act. You may want to delete the reference to the application and just state that the amendment meets the Act's requirements.

Auburn L. Mitchell

Auburn L. Mitchell, Attorney
Office of the Executive Legal
Director