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Am - 54 to DPR-52

February 25, 1980

Docket Nos. 50-259
and 50-260

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

Dear Mr. Parris:

The Commission has issued the enclosed Amendment Nos. 59 and 54 to Facility Licenses Nos. DPR-33, and DPR-52 for the Browns Ferry Nuclear Plant, Units Nos. 1 and 2. These amendments are in response to your letter of October 4, 1979 (TVA BFNP TS131) as supplemented by your letters dated January 15, 1980 and January 29, 1980.

These amendments change the Technical Specifications to: (1) incorporate the limiting conditions for operation of Browns Ferry Unit No. 1 in the fourth fuel cycle following the current refueling outage, (2) reflect the changes to the low pressure coolant injection (LPCI) system power supply and elimination of the LPCI loop selection logic as requested in our letter of May 11, 1979 authorizing these modifications and (3) clarify the surveillance requirements in Section 4.5.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

T A Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 59 to DPR-33
2. Amendment No. 54 to DPR-52
3. Safety Evaluation
4. Notice

cc w/encl:
See next page

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 54
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 4, 1979, as supplemented by submittals dated January 15, 1980 and January 29, 1980 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-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 54 , 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

T A Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 25, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 54

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

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

11/12	147/148	221/222
97/98	149/150	253/254
111/112	157/158	255/256
145/146	181/182	277/278

The underlined pages are those being changed; marginal lines on these pages indicate the area being revised. Overleaf pages are provided for convenience.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
1.1 <u>Fuel Cladding Integrity</u>	2.1 <u>Fuel Cladding Integrity</u>

I.	Core spray and LPCI actuation--reactor low water level	$\geq$ 378 in. above vessel zero
J.	HPCI and RCIC actuation--reactor low water level	$\geq$ 470 in. above vessel zero
K.	Main steam isolation valve closure--reactor low water level	$\geq$ 470 in. above vessel zero

TABLE 4.2.B (Continued)

Function	Functional Test	Calibration	Instrument Check
Instrument Channel Reactor Low Pressure (PS-68-93 & 94)	(1)	once/3 months	none
Core Spray Auto Sequencing Timers (Normal Power)	(4)	once/operating cycle	none
Core Spray Auto Sequencing Timers (Diesel Power)	(4)	once/operating cycle	none
LPCI Auto Sequencing Timers (Normal Power)	(4)	once/operating cycle	none
LPCI Auto Sequencing Timers (Diesel Power)	(4)	once/operating cycle	none
RHRSW A3, B1, C3, D1 Timers (Normal Power)	(4)	once/operating cycle	none
RHRSW A3, B1, C3, D1 Timers (Diesel Power)	(4)	once/operating cycle	none
ADS Timer	(4)	once/operating cycle	none

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 Storage Tank Low Level	(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

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 177.7" (538" above vessel zero) above the top of the active fuel closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Group 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel closes the Main Steam Line Isolation Valves and Main Steam, RCIC, and HPCI Drain Valves (Group 1 and 7). Details of valve grouping and required closing times are given in Specification 3.7. These trip settings are adequate to prevent core uncovery in the case of a break in the largest line assuming the maximum closing time.

The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel (Table 3.2.B) also initiate the RCIC and HPCI, and trips the recirculation pumps. The low reactor water level instrumentation that is set to trip when reactor water level is 17.7" (378" above vessel zero) above the top of the active fuel (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

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

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

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

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

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

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 and 3.9.B.3.

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 9,000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 15,000 gpm against an indicated system pressure of 200 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.

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

4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the cold shutdown condition within 24 hours.

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are operable.

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and all access paths of the RHRS (containment cooling mode)

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

4. No additional surveillance required

5. When it is determined that one RHR pump (containment cooling mode) or associated heat exchanger is inoperable at a time when operability is required, the remaining RHR pumps (containment cooling mode), the associated heat exchangers and diesel generators, and all active components in the access paths of the RHRS (containment cooling mode) shall be demonstrated to be operable immediately and weekly thereafter until the inoperable RHR pump (containment cooling mode) and associated heat exchanger is returned to normal service.

6. When it is determined that two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable at a time when operability is required, the remaining RHR pumps (containment cooling mode), the associated heat exchangers, and diesel generators, and all active components in the access paths of the RHRS (containment cooling

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

are operable.

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not operable, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains operable.

8. If specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be shutdown and placed in the cold condition within 24 hours.

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one RHR loop with two pumps or two loops with one pump per loop shall be operable. The pumps' associated diesel generators must also be operable.

10. If the conditions of specification 3.5.A.5 are met, LPCI and containment cooling are not required.

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

mode) shall be demonstrated to be operable immediately and daily thereafter until at least three RHR pumps (containment cooling mode) and associated heat exchangers are returned to normal service.

7. When it is determined that one or more access paths of the RHRS (containment cooling mode) are inoperable when access is required, all active components in the access paths of the RHRS (containment cooling mode) shall be demonstrated to be operable immediately and all active components in the access paths which are not backed by a second operable access path for the same phase of the mode (drywell sprays, suppression chamber sprays and suppression pool cooling) shall be demonstrated to be operable daily thereafter until the second path is returned to normal service.

8. No additional surveillance required.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be operable shall be demonstrated to be operable monthly.

10. No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION

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

11. When there is irradiated fuel in the reactor and the reactor vessel pressure is greater than atmospheric, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be operable and capable of supplying cross-connect capability except as specified in specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

12. If three RHR pumps or associated heat exchangers located on the unit cross-connection in the adjacent units are inoperable for any reason (including valve

SURVEILLANCE REQUIREMENTS

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

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be operable monthly when the cross-connect capability is required.

12. When it is determined that three RHR pumps or associated heat exchangers located on the unit cross-connection in the adjacent units are inoperable at

inoperability, pipe break, etc), the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pump and associated diesel generator are operable.

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

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

a time when operability is required, the remaining RHR pump and associated heat exchanger on the unit cross-connection and the associated diesel generator shall be demonstrated to be operable immediately and every 15 days thereafter until the inoperable pump and associated heat exchanger are returned to normal service.

13. No additional surveillance required.

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

3.5.F Reactor Core Isolation Cooling

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

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

G. Automatic Depressurization System (ADS)

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

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

   (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 three of the six ADS valves are 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.) If more than three of the six ADS valves are known to be incapable of automatic operation, an immediate orderly shutdown shall be initiated, with the reactor in a hot shutdown condition in 6 hours and in a cold shutdown condition in the following 18 hours.

4.5.F Reactor Core Isolation Cooling

2. When it is determined that the RCICS in inoperable, the HPCIS shall be demonstrated to be operable immediately and weekly thereafter.

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 two of the ADS valves are 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.

3.5.G Automatic Depressurization System (ADS)

3. If specifications 3.5.G.1 and 3.5.G.2 cannot be met, an orderly shutdown will be initiated and the reactor vessel pressure shall be reduced to 105 psig or less within 24 hours.

H. Maintenance of Filled Discharge Pipe

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

4.5.G Automatic Depressurization System (ADS)

H. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

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. Safety and Relief Valves

1. When more than one relief valve or one or more safety 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. Safety and Relief Valves

1. At least one safety valve and approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves (2 safety and 11 relief) 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 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.

3.6.E Jet Pumps

3.6.F Jet Pump Flow Mismatch

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a hot shutdown condition within 24 hours unless the loop is sooner returned to service.

2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

3. Steady state operation with both recirculation pumps out of service for up to 12 hrs is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hrs.

G. Structural Integrity

1. The structural integrity of the primary system shall be

4.6.E Jet Pumps

b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

2. Whenever there is recirculation flow with the reactor in the Startup or Run Mode and one recirculation pump is operating with the equalizer valve closed, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

F. Jet Pump Flow Mismatch

1. Recirculation pump speeds shall be checked and logged at least once per day.

G. Structural Integrity

1. Table 4.6.A together with supplementary notes, specifies the

3.6/4.6 BASES:

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10 percent or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the unit shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump diffuser body; however, the converse is not true. The lack of any substantial stress in the jet pump diffuser body makes failure impossible without an initial nozzle-riser system failure.

3.6.F/4.6.F <u>Jet Pump Flow Mismatch</u>

3.6/4.6 BASES:

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50% of its rated speed provides assurance when going from one to two pump operation that excessive vibration of the jet pump risers will not occur.

3.6.G/4.6.G Structural Integrity

The requirements for the reactor coolant systems inservice inspection program have been identified by evaluating the need for a sampling examination of areas of high stress and highest probability of failure in the system and the need to meet as closely as possible the requirements of Section XI, of the ASME Boiler and Pressure Vessel Code.

The program reflects the built-in limitations of access to the reactor coolant systems.

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 judgement from actual plant obsrevation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in there additional examinations above the requirements of Section XI of ASME Code.

An augmented inservice surveillance program is required to determine whether any stress corrosion has occurred in any stainless steel piping, stainless components, and highly stressed alloy steel such as hanger springs, as a result of environmental conditions associated with the March 22, 1975 fire.

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
	Standby liquid control system check valves CV 63-526 & 525	1	1	NA	C	Process
	Feedwater check valves CV-3-558, 572, 554, & 568	2	2	NA	0	Process
	Control rod hydraulic return check valves CV-85-576 & 573	1	1	NA	0	Process
	RHRS - LPCI to reactor check valves CV-74-54 & 68	2		NA	C	Process

NOTES FOR TABLE 3.7.A

Key: O = Open

C = Closed

SC = Stays Closed

GC = Goes Closed

Note: Isolation groupings are as follows:

Group 1: The valves in Group 1 are actuated by any one of the following conditions:

1. Reactor Vessel Low Water Level (470")
2. Main Steamline High Radiation
3. Main Steamline High Flow
4. Main Steamline Space High Temperature
5. Main Steamline Low Pressure

Group 2: The valves in Group 2 are actuated by any of the following conditions:

1. Reactor Vessel Low Water Level (538")
2. High Drywell Pressure

Group 3: The valves in Group 3 are actuated by any of the following conditions:

1. Reactor Low Water Level (538")
2. Reactor Water Cleanup System High Temperature
3. Reactor Water Cleanup System High Drain Temperature

Group 4: The valves in Group 4 are actuated by any of the following conditions:

1. HPCI Steamline Space High Temperature
2. HPCI Steamline High Flow
3. HPCI Steamline Low Pressure

Group 5: The valves in Group 5 are actuated by any of the following condition:

1. RCIC Steamline Space High Temperature
2. RCIC Steamline High Flow
3. RCIC Steamline Low Pressure

Group 6: The valves in Group 6 are actuated by any of the following conditions:

1. Reactor Vessel Low Water Level (538")
2. High Drywell Pressure
3. Reactor Building Ventilation High Radiation

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

1. Reactor vessel low water level (470")

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

2. High Drywell pressure

TABLE 3.7.B
TESTABLE PENETRATIONS WITH DOUBLE O-RING SEALS

X-1A	Equipment Hatch
X-1B	" "
X-4	DW Head Access Hatch
X-6	CRD Removal Hatch
X-35A	T.I.P. Drives
X-35B	" "
X-35C	" "
X-35D	" "
X-35E	" "
X-35F	" "
X-35G	" "
X-47	Power Operation Test
X-200A	Supp. Chamber Access Hatch
X-200B	" " " "
X-213A	Suppression Chamber Drain
	DW Flange-Top Head
	Shear Lug Inspection Cover #1
	" " " Hatch #2
	" " " " #3
	" " " " #4
	" " " " #5
	" " " " #6
	" " " " #7
	" " " " #8

BASES

Group 1 - process lines are isolated by reactor vessel low water level (490") in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in group 1 are also closed when process instrumentation detects excessive main steam line flow, high radiation, low pressure, or main steam space high temperature.

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

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

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

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

Group 7 - process lines are closed only on reactor low water level (470"). These close on the same signal that initiates HPCIS and RCICS to ensure that the valves are not open when HPCIS or RCICS action is required.

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

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

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

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

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

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

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

3.7.E/4.7.E Control Room Emergency Ventilation

The control room emergency ventilation system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room emergency ventilation system is designed to automatically start upon control room isolation and to maintain the control room pressure to the design positive pressure so that all leakage should be out leakage.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If the system is found to be inoperable, there is not immediate threat to the control room and reactor operation or refueling operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within seven days, the reactor is shutdown and brought to cold shutdown within 24 hours or refueling operations are terminated.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 59 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 54 TO FACILITY OPERATING LICENSE NO. DPR-52

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS NOS. 1 AND 2

DOCKET NOS. 50-259 AND 50-260

1.0 Introduction

By letter dated October 4, 1979(1) (TVA BFNP TS 131), as supplemented by letters dated January 15, 1980 and January 29, 1980, the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License Nos. DPR-33 and DPR-52 for the Browns Ferry Nuclear Plant, Unit Nos. 1 and 2.

The proposed amendments and revised Technical Specifications would: (1) incorporate the limiting conditions for operation of Browns Ferry Unit No. 1 in the fourth fuel cycle following the current refueling outage, (2) reflect the changes to the low pressure coolant injection (LPCI) system power supply and elimination of the LPCI loop selection logic as requested in our letter of May 11, 1979 authorizing these modifications and (3) clarify the surveillance requirements in Section 4.5.

2.0 Discussion

Browns Ferry Unit No. 1 (BF-1) shutdown for its third refueling on January 3, 1980. BF-1 was initially fueled with 764 of the General Electric Co. (GE) 7 x 7 fuel assemblies containing 49 fuel rods each. During the first refueling, 166 of the 7 x 7 fuel assemblies were replaced with a like number of one water rod 8 x 8 fuel assemblies containing 63 fuel rods each. During the second refueling, an additional 156 of the original fuel assemblies were replaced with two water rod retrofit 8 x 8R fuel bundles containing 62 fuel rods each. During the current refueling outage, an additional 232 of the 7 x 7 fuel bundles will be replaced with P 8 x 8 fuel assemblies, each containing 62 fuel rods. The prepressurized fuel assemblies (P 8 X 8R) are essentially identical from a core physics standpoint to the two water rod fuel assemblies (8 X 8R) except that they are prepressurized with about three rather than one atmospheres of helium to minimize fuel clad interaction. Our evaluation of the P 8 X 8R fuel is discussed in the safety evaluation attached to our letter of April 16, 1979 to General Electric approving the use of this fuel in BWR reload licensing applications. The larger inventory of helium gas improves the gap conductance between fuel pellets and cladding resulting in reductions in fuel temperatures, thermal expansion and fission gas release. The pressurized rods operate at effectively lower linear heat generation rates and are therefore expected to yield performance benefits in terms of fuel reliability. The increased prepressurization also results in improved margin to MAPLHGR limits by reducing stored energy, although TVA is not proposing to take any credit for these beneficial effects in the subject reload application (i.e., they are not proposing any changes in the existing MAPLHGR vs. Exposure limits in the existing Technical Specifications). In support of this reload application for BF-1, TVA submitted a reload licensing document prepared by GE[2] and proposed changes to the Technical Specifications.[1] The first use of P8 x 8R fuel in a Browns Ferry Unit was approved for the last reload of Unit No. 3 (Amendment No. 28 to Facility Operating License No. DPR-58 dated November 30, 1979).

With this refueling, Browns Ferry Unit 1 will be on an 18 month refueling cycle. Units Nos. 2 and 3 are also on 18 month refueling cycles.

As noted above, this reload involves loading of prepressurized GE 8 x 8 retrofit (P8 x 8R) fuel. The description of the nuclear and mechanical designs of P8 x 8 fuel is contained in Reference 3. The use and safety implications of prepressurized fuel are presented in Reference 3 and have been found acceptable per Reference 4 (enclosed in Appendix C of Reference 3).

Values for plant-specific data such as steady state operating pressure, core flow, safety and safety/relief valve setpoints, rated thermal power, rated steam flow, and other design parameters are provided in Reference 3. Additional plant and cycle dependent information is provided in the reload application (Reference 2) which closely follows the outline of Appendix A of Reference 3. Reference 4 includes a description of the staff's review, approval, and conditions of approval for the plant-specific data. The above-mentioned plant-specific data have been used in the transient and accident analysis provided with the reload application in compliance with Reference 4.

Our safety evaluation of the GE generic reload licensing topical report has also concluded that the nuclear, and mechanical design of the 8 x 8R and P8 x 8R fuels, and GE's analytical methods for nuclear and thermal-hydraulic calculations as applied to mixed cores containing 8 x 8, 8 x 8R and P8 x 8R fuels, are acceptable. Approval of the application of the analytical methods did not include plants incorporating a prompt recirculation pump trip (RPT) or Thermal Power Monitor (TPM).

Because of our review of a large number of generic considerations related to use of 8 x 8R and P8 x 8R fuels in mixed loadings, and on the basis of the evaluations which have been presented in Reference 3, only a limited number of additional areas of review have been included in this safety evaluation report. For evaluations of areas not specifically addressed in this safety evaluation report, the reader is referred to Reference 3.

3.0 Evaluation

3.1 Core Reload

3.1.1 Nuclear Characteristics

For cycle 4 operation, 232 fresh P8 x 8R fuel bundles of type P8DRB284 will be loaded into the core (Reference 2). The remainder of the 764 fuel bundles in the core will be previously irradiated bundles as indicated in Reference 2. Based on the data provided in Reference 2 both the control rod system and the standby liquid control system will have acceptable shutdown capability during cycle 4.

3.1.2 Thermal Hydraulics

3.1.2.1 Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 3, for BWR cores which reload with GE's retrofit 8 x 8 fuel, the safety limit minimum critical power ratio (SLMCPR) resulting from either core-wide or localized abnormal operational transients is equal to 1.07. When meeting this SLMCPR during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition. The 1.07 SLMCPR is unchanged from the SLMCPR previously approved. The basis for this safety limit is addressed in Reference 3.

3.1.2.2 Operating Limit MCPR

Various transient events can reduce the MCPR from its normal operating level. To assure that the fuel cladding integrity SLMCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed for this reload by the licensee, in order to determine which event results in the largest reduction in the minimum critical power ratio. Addition of the largest reductions in critical power ratio to the SLMCPR establishes the operating limits for each fuel type.

3.1.2.2.1 Transient Analysis Methods

The generic methods used for these calculations, including cycle-independent initial conditions and transient input parameters, are described in Reference 3. The staff evaluation, included as Appendix C of Reference 3, contains our acceptance of the cycle-independent values. Additionally, Appendix C contains our evaluation of the transient analysis methods, together with a description and summary of the outstanding issues associated with these methods. Supplementary cycle-independent initial conditions and transient input parameters used in the transient analyses appear in the tables in Sections 6 and 7 of Reference 2. Our evaluation of the methods used to develop these supplementary input values is also included in Appendix C of Reference 3.

3.1.2.2.2 Transient Analysis Results

The transients evaluated were the limiting pressure and power increase transients generator load rejection without bypass and the feedwater controller failure (loss of 100°F feedwater heating), and the control rod withdrawal error. Initial conditions and transient input parameters as specified in Sections 6 and 7 of Reference 2 were assumed.

The results of these analyses are outlined in Reference 2 sections 9 and 10. On this topic, it is acceptable if fuel specific operating limits are established for prepressurized fuel (Appendix C, Reference 3). On this basis, the transient analysis results are acceptable for use in the evaluation of the operating limit MCPR. Based on this, the proposed Technical Specification modifications to operating limit MCPR are acceptable.

3.1.3 Accident Analyses

3.1.3.1 ECCS Appendix K Analysis

In our safety evaluation of Reference 3, we concluded that "the continued application of the present GE ECCS-LOCA ("Appendix K") models to the 8 x 8 retrofit reload fuel is generically acceptable and in our Reference 4 evaluation we extended that conclusion to prepressurized fuel. On these bases, the proposed MAPLHGR limits for the new prepressurized fuel are acceptable.

3.1.3.2 Control Rod Drop Accident

The scram reactivity shape function (cold) does not satisfy the requirements for the bounding analyses described in Reference 3. Therefore, it was necessary for the licensee to perform a plant and cycle specific analysis for the control rod drop accident. The results of this analysis are well below the acceptance criterion of 280 calories per gram.

3.1.3.3 Fuel Loading Error

The GE method for analysis of misoriented and misloaded bundles has been reviewed and approved by the staff and is part of the Reference 3 methodology. Potential fuel loading errors involving misoriented bundles and bundles loaded into incorrect positions have been analyzed by this methodology and the results are acceptable.

3.1.3.4 Overpressure Analysis

The overpressure analysis for the MSIV closure with high flux scram, which is the limiting overpressure event, has been performed in accordance with the requirements of Reference 3. As specified in the staff evaluation included in Reference 3, the sensitivity of peak vessel pressure to failure of one safety valve has also been evaluated. We agree that there is sufficient margin between the peak calculated vessel pressure and the design limit pressure. Therefore, the limiting overpressure event as analyzed by the licensee is considered acceptable.

3.1.4 Thermal Hydraulic Stability

The results of the thernal hydraulic stability analysis (Reference 3) show that the channel hydrodynamic and reactor core decay ratios at the natural circulation - 105% rod line intersection (which is the least stable physically attainable point of operation) are below the stability limit. Because operation in the natural circulation mode will be restricted by Technical Specifications, there will be added margin to the stability limit and this is acceptable.

3.1.5 Startup Test Program

The licensee has not changed his startup test program from that approved for the previous cycle. This program therefore remains acceptable.

3.2 Other Changes to Technical Specifications

3.2.1 Reactor Low Water Level

On August 2, 1978, we issued Amendments Nos. 40, 38 and 14 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. These amendments changed the Technical Specifications to lower the reactor low water level setpoint from 490 inches to 470 inches above vessel zero. The low water level setpoint, which is commonly called the L_2 setpoint, is that reactor water level below which the main steamline isolation valves close, HPCI and RCIC flows are initiated, and the recirculation pumps trip. We evaluated the ECCS performance with the L_2 setpoint at 470 inches and the effect of reduction in L_2 on results of anticipated transients and found that these were acceptable. The Amendments changed 4 pages of the Technical Specifications for each unit to reflect the approved value of 470 inches for the L_2 setpoint. Subsequently, the licensee found 4 additional pages in the Technical Specifications for Units 1 and 2 (pages 11, 254, 255 and 277) where the 490" was referenced with respect to valve closures. The proposed changes to the Technical Specifications are to correct this error. (This is an error toward the conservative.) We conclude that the proposed changes to rectify this oversight are acceptable.

3.2.2 Surveillance Requirements in Section 4.5.B

In Section 4.5.B of the present Technical Specification on the Residual Heat Removal System (RHRS) (LPCI and Containment Cooling), the surveillance requirements in several items do not track the correspondingly numbered limiting condition for operation (LCO) in Section 3.5.B. For example, surveillance requirement 4.5.B.10 is the surveillance requirement for LCO 3.5.B.11 and surveillance requirement 4.5.B.12 is the requirement for LCO 3.5.B.14. The licensee proposes to change this for clarity by having the numbers for the surveillance requirements correspond to the numbered LCOs. Where no surveillance is indicated, the surveillance requirement will state "No additional surveillance required." As part of their review of this section of the Technical Specifications, the licensee has proposed to increase the surveillance requirements on low pressure ECCS systems when one RHR pump is inoperable. The present Technical Specifications (Section 4.5.B.3, p. 145) require that "When it is determined that one RHR pump (LPCI mode) is inoperable at a time when operability is required, ... the operable RHRS pumps (LPCI mode) shall be demonstrated to be operable 10 days thereafter until the inoperable pump is returned to normal service." The licensee has proposed to change this to require that "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." While we have not concluded that this increased conservatism is necessary, we do find the increased surveillance is acceptable. Another change in the surveillance requirements (item 4.5.B.12, p. 149) is to correct a typographical error in the present Technical Specifications.

3.2.3 LPCI Modifications

By letter dated May 11, 1979, we issued Amendments Nos. 51, 45 and 23 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. The Amendments added a condition to the license for each facility authorizing TVA to perform certain modifications (as described in TVA's submittals and the Safety Evaluation related to these Amendments) to change the power supply for certain LPCI valves for Units Nos. 1, 2 and 3 and to eliminate the loop selection logic for Unit No. 3. Our letter of May 11, 1979 noted that TVA had committed to submit proposed Technical Specification changes with the reload amendment request for each unit to reflect these modifications. (The changes for Unit No. 3 were submitted with TVA's reload amendment request of August 6, 1979 and approved by Amendment No. 28 to License No. DPR-68 which we issued on November 30, 1979.) The modifications to BF-1 are described in detail in the safety evaluation accompanying our letter of May 11, 1979. In summary, the overall modifications encompassed:

a. Elimination of the Low Pressure Coolant Injection (LPCI) system's recirculation loop selection logic, revision of the logic and closure of the Residual Heat Removal (RHR) cross-tie valve and a recirculation equalizer valve; and

b. Changing the power supply to the reactor MOV boards that feed the motor operators of the LPCI injection valves, the recirculation pump discharge valves, and the RHR pump minimum flow bypass valves. The change involves the use of Class 1E motor-generator (M-G) sets as isolation devices between the auto-transfer feature of the 480V reactor MOV boards and the divisional 480V shutdown boards. The auto-transfer feature will be eliminated from all 480V reactor MOV boards not protected by M-G sets.

The proposed changes on pages 97, 111, 112, 182 and 221 reflect the above modification. Each proposed change is discussed in detail below.

a. The change to Table 4.2.B, p. 97 (Surveillance Requirements for Instrumentation that Initiate or Control the CSCS) removes the surveillance requirements on four reactor pressure sensors (PS-3-186A&B, and PS-3-187A&B) whose sole function was that of a permissive in the LPCI recirculation loop selection logic. Since the logic no longer exists, the sensors have been removed and deleting them from the instruments to be surveillance tested is appropriate. We find the proposed change acceptable.

b. The proposed change in Section 3.2 "BASES" at the bottom of page 111 and top of page 112 is to delete the words "provides input to the LPCI loop selection logic." This sentence discussed the bases for the reactor pressure sensors in "a" above. The change is to remove the low reactor water level instrumentation as the source of a LPCI loop selection logic initiation signal, since the latter function no longer exists. We find the proposed change acceptable.

c. The present Technical Specifications (Section 3.6.F.1 and 3.6.F.2, p. 182) require that the speeds of the two recirculation pumps be maintained within 122% and 135% of one another when the core power is above 80% or below 80% of rated power, respectively. As explained in the bases for these requirements (p. 221, "Jet Pump Flow Mismatch), this was necessary when there was automatic loop selection logic. The purpose of this limitation was to prevent the LPCI loop selection logic from selecting the wrong loop for injection which was possible for certain low probability accidents with the recirculation loop operating at large speed differences. Since the LPCI loop selection logic has been removed from the Browns Ferry Nuclear Plant, Unit Nos. 1 and 2, there is no longer the need for surveillance requirements relating to this logic nor the need to limit the variation of recirculation pump speed for purposes associated with this logic. We find the proposed changes to the Technical Specifications to be acceptable.

3.2.4 Single Loop Operation

On September 19, 1978 and September 29, 1978, we issued Amendments Nos. 41 and 43, respectively, to Facility License No. DPR-33 which authorized operation of BF-1 with one recirculation loop for the duration of cycle 2.

Cycle 2 for BF-1 ended in November 1978. During the period of single loop operation, there was a reduced limit on core maximum fraction of limiting power density (Section 2.1.B, page 10) that applied only during this period. The proposed change on page 10 is to remove the limit for one recirculation loop operation since it is no longer applicable. We find the proposed change to be desirable and acceptable.

4.0 Environmental Considerations

We have determined that the amendments do 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 amendments involve 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 the amendments.

5.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: February 25, 1980

References

1. Letter, L. M. Mills (TVA) to H. R. Denton (NRC), dated October 4, 1979.

2. "Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Plant Unit 1 Reload No. 3," NEDO-24209, August 1979.

3. "General Electric Boiling Water Reactor Generic Reload Application," NEDE-24011-P-A, August 1979.

4. Letter, T. A. Ippolito (USNRC) to R. Gridley (GE), April 16, 1979 and enclosed SER.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-259 AND 50-260

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 59 to Facility Operating License No. DPR-33 and Amendment No. 54 to Facility Operating License No. DPR-52 issued to Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Units Nos. 1 and 2, located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

These amendments change the Technical Specifications to: (1) incorporate the limiting conditions for operation of Browns Ferry Unit No. 1 in the fourth fuel cycle following the current refueling outage, (2) reflect the changes to the low pressure coolant injection (LPCI) system power supply and elimination of the LPCI loop selection logic as requested in our letter of May 11, 1979 authorizing these modifications and (3) clarify the surveillance requirements in Section 4.5.

The application for the amendments 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 amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendments dated October 4, 1979, as supplemented by submittals dated January 15, 1980 and January 29, 1980, (2) Amendment No. 59 to License No. DPR-33 and Amendment No. 54 to License No. DPR-52 and (3) the Commission's related Safety Evaluation. All of 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 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 25th day of February 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
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