Docket

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

February 6, 1981

Docket Nos. 50-259
50-260
and 50-296

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

Dear Mr. Parris:

The Commission has issued the enclosed Amendment Nos. 66, 62 and 38 to Facility License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Unit Nos. 1, 2 and 3. These amendments which are in response to your applications dated June 13, 1980 (TVA BFNP TS 139) and October 16, 1980 (TVA BFNP TS 152), change the Technical Specifications to (1) revise the definitions for "Limiting Conditions for Operation" and "Operability" to be consistent with the definitions in the model technical specifications enclosed with our generic letter to you of April 10, 1980, and (2) revise the definition of "Cold Shutdown" to be consistent with the "Standard BWR Technical Specifications".

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

Sincerely,

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 66 to DPR-33
2. Amendment No. 62 to DPR-52
3. Amendment No. 38 to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures:
See next page

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 66
License No. DPR-33

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

   A. The applications for amendment by Tennessee Valley Authority (the licensee) dated June 13, 1980 and October 16, 1980, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility License No. DPR-33 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 66, 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

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 6, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 66

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

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

   1/2
   3/4
   15/16
   33/34
   61/62
   63/64
   77/78
   175
   187/188

   The underlined pages are the pages being changed; the marginal lines on these pages denote the area being changed. The overleaf page is provided for convenience.

2. Add the following new page:

   2a

INTRODUCTION

This document presents the technical specifications for the Browns Ferry Nuclear Plant Unit 1 only.

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit requires unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

B. Limiting Safety System Setting (LSSS) - The limiting safety system setting are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represent margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

C. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

1. In the event a Limiting Condition for Operation and/or associated requirements cannot be satisifed because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours unless corrective measures are completed that permit operation under the permissible discovery or until the reactor is placed in an operational condition in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications. This provides actions to be taken for circumstances not directly provided for in the specifications and where occurrence would violate the intent of the specification. For example, if a specification calls for two systems (or subsystems) to be operable and provides for explicit requirements if one system (or subsystem) is inoperable, then if both systems (or subsystems) are inoperable the unit is to be in at least Hot Standby in 6 hours and in Cold Shutdown within the following 30 hours if the inoperable condition is not corrected.

1.0 DEFINITIONS (continued)

2. When a system, subsystem, train, component or device is determined to be inoperable solely because its onsite power source is inoperable, or solely because its offsite power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition For Operation, provided:
(1) its corresponding offsite or diesel power source is operable; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are operable, or likewise satisfy these requirements. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least Hot Standby within 6 hours, and in at least Cold Shutdown within the following 30 hours. This is not applicable if the unit is already in Cold Shutdown or Refueling. This provision describes what additional conditions must be satisfied to permit operation to continue consistent with the specifications for power sources, when an offsite or onsite power source is not operable. It specifically prohibits operation when one division is inoperable because its offsite or diesel power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason. This provision permits the requirements associated with individual systems, subsystems, trains, components or devices to be consistent with the requirements of the associated electrical power source. It allows operation to be governed by the time limit of the requirements associated with the Limiting Condition For Operation for the offsite or diesel power source, not the individual requirements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its offsite or diesel power source.

D. DELETED

1.0 DEFINITIONS (cont'd)

E. Operable - Operability - A system, subsystem, train, component, or device shall be Operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

F. Operating - Operating means that a system or component is performing its intended functions in its required manner.

G. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

H. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup" or "Run" position with the reactor critical and above 1% rated power.

I. Hot Standby Condition - Hot standby condition means operation with coolant temperature greater than 212°F, system pressure less than 1055 psig, the main steam isolation valves closed and the mode switch in the Startup/Hot Standby position.

J. Cold Condition - Reactor coolant temperature equal to or less than 212°F.

K. Hot Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature greater than 212°F.

L. Cold Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature equal to or less than 212°F.

M. Mode of Operation - A reactor mode switch selects the proper interlocks for the operational status of the unit. The following are the modes and interlocks provided:

1. Startup/Hot Standby Mode - In this mode the reactor protection scram trips initiated by condenser low vacuum and main steam line isolation valve colsure, are bypassed when reactor pressure is less than 1055 psig, the reactor protection system is energized with IRM neutron monitoring system trip, the APRM 15% high flux trip, and control rod withdrawal interlocks in service. This is often referred to as just Startup Mode. This is intended to imply the startup/Hot Standby position of the mode switch.

2. Run Mode - In this mode the reactor system pressure is at or above 850 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RBM interlocks in service.

3. Shutdown Mode - Placing the mode switch to the shutdown position initiates a reactor scram and power to the control rod drives is removed. After a short time period (about 10 sec), the scram signal is removed allowing a scram reset and restoring the normal valve lineup in the control rod drive hydraulic system; also, the main steam line isolation scram and main condenser low vacuum scram are bypassed if reactor vessel pressure is below 1055 psig.

4. Refuel Mode - With the mode switch in the refuel position interlocks are established so that one control rod only may be withdrawn when the Source Range Monitor indicate at least 3 cps and the refueling crane is not over the reactor; also, the main steam line isolation scram and main condenser low vacuum scram are bypassed if reactor vessel pressure is below 1055 psig. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

N. Rated Power - Rated power refers to operation at a reactor power of 3,293 MWt; this is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power. Design power, the power to which the safety analysis applies, corresponds to 3440 MWt.

O. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All non-automatic containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed. These valves may be opened to perform necessary operational activities.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

P. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

2.1 BASES: FUEL CLADDING INTEGRITY SAFETY LIMIT

The fuel cladding represents one of the physical barriers which separate radioactive materials from environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system setpoints. While fission product migration from cladding performation is just as measurable as that from use-related cracking, the thermally-caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding safety limit is defined in terms of the reactor operating conditions which can result in cladding perforation.

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset transition boiling (MCPR of 1.0). This establishes a Safety Limit such that the minimum critical power ratio (MCPR) is no less than 1.07. MCPR >1.07 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possiblity of clad failure. Since boiling transition is not a directly observable parameter, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables, i.e., normal plant operation presented on Figure 2.1.1 by the nominal expected flow control line. The Safety Limit (MCPR of 1.07) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR > limits specified in specification 3.5.K) more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit 1.07 is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the safety limit are provided at the beginning of each fuel cycle.

2.1 BASES

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 = 1.07 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 occurence 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 of 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 = 1.07) operation is constrained to a maximum LHGR of 18.5 kw/ft for 7x7 fuel and 13.4 kw/ft for all 8x8 fuels. This limit is reached when the Core Maximum Fraction of Limiting Power Density equals 1.0 (CMFLPD = 1.0). For the case where Core Maximum Fraction of Limiting Power Density exceeds the Fraction of Rated Thermal Power, operation is permitted only at less than 100% of rated 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 flow will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 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 28×10^3 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 shut down, consideration must also be given to water level requirements due to the effect of decay heat. If water level should drop below the top of the fuel during this time, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation. As long as the fuel remains covered with water, sufficient cooling is available to prevent fuel clad perforation.

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	Modes in Which Function Must Be Operable: Shut-down	Refuel(7)	Startup/Hot Standby	Run	Action(1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤ 120/125 Indicated on scale	X(22)	X (22)	X	(5)	1.A
3	Inoperative			X	X	(5)	1.A
	APRM (16)						
2	High Flux	See Spec. 2.1.A.1				X	1.A or 1.B
2	High Flux	≤ 15% rated power		X(21)	X(17)	(15)	1.A or 1.B
2	Inoperative	(13)		X(21)	X(17)	X	1.A or 1.B
2	Downscale	≥ 3 Indicated on Scale		(11)	(11)	X(12)	1.A or 1.B
2	High Reactor Pressure	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14)	≤ 2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14)	≥ 538" above vessel zero		X	X	X	1.A
2	High Water Level in Scram Discharge Tank	≤ 50 Gallons	X	X(2)	X	X	1.A

TABLE 3.1.A (Continued)

Min. No. of Operable Inst. Channels Per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable: Refuel(7)	Startup/Hot Standby	Run	Action(1)
4	Main Steam Line Isolation Valve Closure	≤ 10% Valve Closure	X(3)(6)	X(3)(6)	X(6)	1.A or 1.C
2	Turbine Cont. Valve Fast Closure	Upon trip of the fast acting solenoid valves	X(4)	X(4)	X(4)	1.A or 1.D
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 Oil 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

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

7. A train is considered a trip system.

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

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

Amendment No. 40, 66

TABLE 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero.	A	1. Below trip setting initiated HPCI.
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #1)	≥ 378" above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #2)	≥ 378" above vessel zero.	A	1. Below trip settings in conjunction with drywell high pressure, low water level permissive, 120 sec. delay timer and CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184 & 185, SW #1)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LITS-3-52 & 62, SW #1)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.
2	Instrument Channel - Drywell High Pressure (PS-64-58 E-H)	1≤ P≤2.5 psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
1	Instrument Channel - Reactor Low Pressure (PS-68-93 & 94, SW #1)	100 psig $\pm$ 15	A	1. Below trip setting in conjunction with containment isolation signal and both suction valves open will close RHR (LPCI) admission valves.
2	Core Spray Auto Sequencing Timers (5)	$6 \leq t \leq 8$ secs.	B	1. With diesel power 2. One per motor
2	LPCI Auto Sequencing Timers (5)	$0 \leq t \leq 1$ sec.	B	1. With diesel power 2. One per motor
1	RHRSW A3, B1, C3, and D1 Timers	$13 \leq t \leq 15$ sec.	A	1. With diesel power 2. One per pump
2	Core Spray and LPCI Auto Sequencing Timers (6)	$0 \leq t \leq 1$ sec. $6 \leq t \leq 8$ sec. $12 \leq t \leq 16$ sec. $18 \leq t \leq 24$ sec.	B	1. With normal power 2. One per CSS motor 3. Two per RHR motor
1	RHRSW A3, B1, C3, and D1 Timers	$27 \leq t \leq 29$ sec.	A	1. With normal power 2. One per pump

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Drywell High Pressure (PS-64-58 A-D, SW #2)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal.(15)
2	Instrument Channel - Reactor Low Water Level (LS-3-56A, B, C, D)	≥ 470" above vessel zero	A	1. Below trip setting trips recirculation pumps
2	Instrument Channel Reactor High Pressure (PS-3-204 A, B, C, D)	≤ 1120 psig	A	1. Above trip setting trips recirculation pumps
2	Instrument Channel - Drywell High Pressure (PS-64-58A-D, SW #1)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
2(16)	Instrument Channel - Drywell High Pressure (PS-64-57A-D)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor water level, drywell high pressure, 120 sec. delay timer and CSS or RHR pump running, initiates ADS.
2	Instrument Channel - Reactor Low Pressure (PS-3-74 A & B, SW #2) (PS-68-95, SW #2) (PS-68-96, SW #2)	450 psig ± 15	A	1. Below trip setting permissive for opening CSS and LPCI admission valves.
2	Instrument Channel - Reactor Low Pressure (PS-3-74A & B, SW #1) (PS-68-95, SW #1) (PS-68-96, SW #1)	230 psig ± 15	A	1. Recirculation discharge valve actuation.

TABLE 3.2.E
INSTRUMENTATION THAT MONITORS LEAKAGE INTO DRYWELL

System (2)	Setpoints		Action	Remarks
Equipment Drain			(1)	1. Used to determine identifiable reactor coolant leakage. 2. Considered part of sump system.
Flow Integrator	N/A			
Sump Fill Rate Timer	≥20.1 min.			
Sump Pump Out Rate Timer	≤13.4 min.			
Floor Drain			(1)	1. Used to determine unidentifiable reactor coolant leakage. 2. Considered part of sump system.
Flow Integrator	N/A			
Sump Fill Rate Timer	≥80.4 min.			
Sump Pump Out Rate Timer	≤8.9 min.			
Drywell Air Sampling	Gas and Particulate	3 x Average Background	(3)	

NOTES:

(1) Whenever a system is required to be operable, there shall be one operable system either automatic or manual, or the action required in Section 3.6.C.2 shall be taken.

(2) An alternate system to determine the leakage flow is a manual system whereby the time between sump pump starts is monitored. The time interval will determine the leakage flow because the volume of the sump will be known.

(3) Upon receipt of alarm, immediate action will be taken to confirm the alarm and assess the possibility of increased leakage.

TABLE 3.2.F
SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	LI-3-46 A LI-3-46 B	Reactor Water Level	Indicator -107.5" to +107.5"	(1) (2) (3)
2	PI-3-54 PI-3-61	Reactor Pressure	Indicator 0-1200 psig	(1) (2) (3)
2	PR-64-50 PI-64-67	Drywell Pressure	Recorder 0-80 psia Indicator 0-80 psia	(1) (2) (3)
2	TI-64-52 TR-64-52	Drywell Temperature	Recorder, Indicator 0-400°F	(1) (2) (3)
1	TR-64-52	Suppression Chamber Air Temperature	Recorder 0-400°F	(1) (2) (3)
2	TI-64-55 TIS-64-55	Suppression Chamber Water Temperature	Indicator, 0-400°F	(1) (2) (3)
2	LI-64-54 A LI-64-66	Suppression Chamber Water Level	Indicator -25" to +25"	(1) (2) (3)
1	NA	Control Rod Position	6V Indicating Lights)	
1	NA	Neutron Monitoring	SRM, IRM, LPRM 0 to 100% power)	(1) (2) (3) (4)
1	PS-64-67	Drywell Pressure	Alarm at 35 psig)	
1	TR-64-52 and PS-64-58 B and IS-64-67	Drywell Temperature and Pressure and Timer	Alarm if temp. > 281°F and pressure > 2.5 psig after 30 minute delay)	(1) (2) (3) (4)
1	LI-84-2A	CAD tank "A" level	Indicator 0 to 100%	(1)
1	LI-84-13A	CAD tank "B" level	Indicator 0 to 100%	(1)

3.6.A Thermal and Pressurization Limitations

3. During heatup by non-nuclear means, except when the vessel is vented or as indicated in 3.6.A.4, cooldown following nuclear shutdown on low-level physics tests, the reactor vessel temperatures shall be at or above the temperatures of curve #2 of figure 3.6.1.

4. The reactor vessel shell temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on curve #1 of figure 3.6-1. The applicability of this curve to these tests is extended to non-nuclear heatup and ambient loss cooldown associated with these tests only if the heatup and cooldown rates do not exceed 15°F per hour.

5. The reactor vessel head bolting studs may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are above 70°F. Before loading the flanges any more, the vessel flange and head flange must be greater than 100°F, and must remain above 100°F while under full tension.

6. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

7. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

4.6.A Thermal and Pressurization Limitations

3. Test specimens representing the reactor vessel, base weld and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The number and type of specimens will be in accordance with GE report NEDO-10115. The specimens shall meet the intent of ASTM E 185-70. Samples shall be withdrawn at one-fourth and three-fourths service life.

4. Neutron flux wires shall be installed in the reactor vessel adjacent to the reactor vessel wall at the core midplane level. The wires shall be removed and tested during the first refueling outage to experimentally verify the calculated values of neutron fluence at one-fourth of the beltline shell thickness that are used to determine the NDTT shift from Figure 3.6-2.

5. When the reactor vessel head bolting studs are tensioned and the reactor is in a cold condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

6. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

7. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

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.

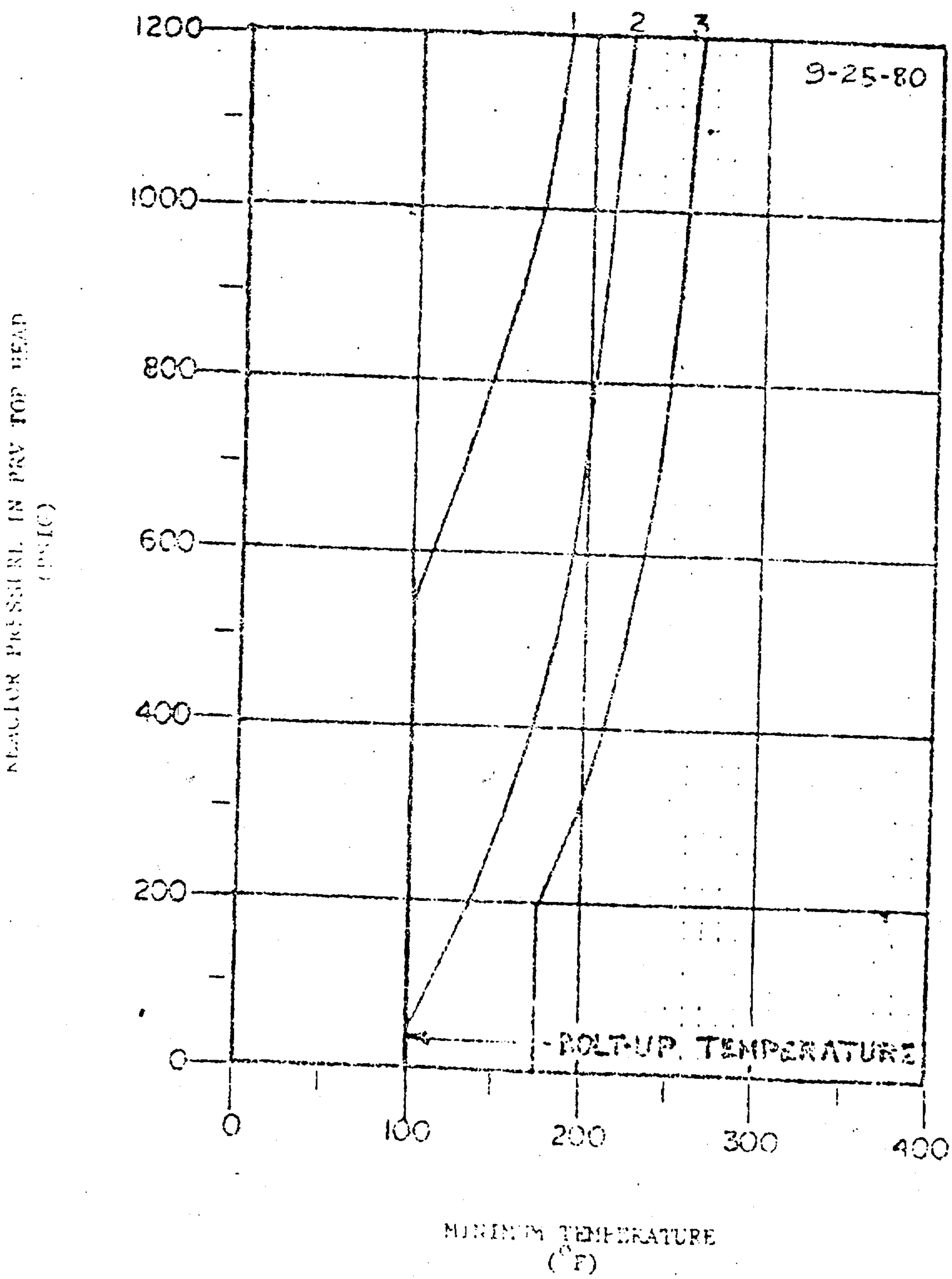
5. If a snubber is determined to be inoperable while the reactor is in the shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup.

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% of 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.

Figure 3.6-1



Curve #1
Minimum temperature for pressure tests such as required by Section XI.

Curve #2
Minimum temperature for mechanical heat up or cooldown following nuclear shutdown.

Curve #3
Minimum temperature for core operation (criticality) Includes additional margin required by 10CFR50 Appendix G, Par. IV A.2.C.

Notes
These curves are shifted 30°F to the right of the original set of curves to include a ΔRT_{NDT} of 30°F. This shift will allow these curves to used thru 4.0 EFPY.

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. 62
License No. DPR-52

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

   A. The applications for amendment by Tennessee Valley Authority (the licensee) dated June 13, 1980 and October 16, 1980, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, the provisions of the Act, and the 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. 62, 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

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 6, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 62

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:

1/2
3/4
33/34
61/62
63/64
77/78
175
187/188

The underlined pages are the pages being changed; the marginal lines on these pages denote the area being changed. The overleaf page is provided for convenience.

2. Add the following new page:

2a

INTRODUCTION

This document presents the technical specifications for the Browns Ferry Nuclear Plant Unit 2 only.

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit requires unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

B. Limiting Safety System Setting (LSSS) - The limiting safety system setting are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represent margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

C. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

1. In the event a Limiting Condition for Operation and/or associated requirements cannot be satisifed because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours unless corrective measures are completed that permit operation under the permissible discovery or until the reactor is placed in an operational condition in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications. This provides actions to be taken for circumstances not directly provided for in the specifications and where occurrence would violate the intent of the specification. For example, if a specification calls for two systems (or subsystems) to be operable and provides for explicit requirements if one system (or subsystem) is inoperable, then if both systems (or subsystems) are inoperable the unit is to be in at least Hot Standby in 6 hours and in Cold Shutdown within the following 30 hours if the inoperable condition is not corrected.

1.0 DEFINITIONS (continued)

2. When a system, subsystem, train, component or device is determined to be inoperable solely because its onsite power source is inoperable, or solely because its offsite power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition For Operation, provided:
(1) its corresponding offsite or diesel power source is operable; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are operable, or likewise satisfy these requirements. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least Hot Standby within 6 hours, and in at least Cold Shutdown within the following 30 hours. This is not applicable if the unit is already in Cold Shutdown or Refueling. This provision describes what additional conditions must be satisfied to permit operation to continue consistent with the specifications for power sources, when an offsite or onsite power source is not operable. It specifically prohibits operation when one division is inoperable because its offsite or diesel power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason. This provision permits the requirements associated with individual systems, subsystems, trains, components or devices to be consistent with the requirements of the associated electrical power source. It allows operation to be governed by the time limit of the requirements associated with the Limiting Condition For Operation for the offsite or diesel power source, not the individual requirements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its offsite or diesel power source.

D. DELETED

1.0 DEFINITIONS (cont'd)

E. Operable - Operability - A system, subsystem, train, component, or device shall be Operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

F. Operating - Operating means that a system or component is performing its intended functions in its required manner.

G. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

H. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup" or "Run" position with the reactor critical and above 1% rated power.

I. Hot Standby Condition - Hot standby condition means operation with coolant temperature greater than 212°F, system pressure less than 1055 psig, the main steam isolation valves closed and the mode switch in the Startup/Hot Standby position.

J. Cold Condition - Reactor coolant temperature equal to or less than 212°F.

K. Hot Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature greater than 212°F.

L. Cold Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature equal to or less than 212°F.

M. Mode of Operation - A reactor mode switch selects the proper interlocks for the operational status of the unit. The following are the modes and interlocks provided:

1. Startup/Hot Standby Mode - In this mode the reactor protection scram trips initiated by condenser low vacuum and main steam line isolation valve colsure, are bypassed when reactor pressure is less than 1055 psig, the reactor protection system is energized with IRM neutron monitoring system trip, the APRM 15% high flux trip, and control rod withdrawal interlocks in service. This is often referred to as just Startup Mode. This is intended to imply the startup/Hot Standby position of the mode switch.

2. Run Mode - In this mode the reactor system pressure is at or above 850 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RBM interlocks in service.

3. Shutdown Mode - Placing the mode switch to the shutdown position initiates a reactor scram and power to the control rod drives is removed. After a short time period (about 10 sec), the scram signal is removed allowing a scram reset and restoring the normal valve lineup in the control rod drive hydraulic system; also, the main steam line isolation scram and main condenser low vacuum scram are bypassed if reactor vessel pressure is below 1055 psig.

4. Refuel Mode - With the mode switch in the refuel position interlocks are established so that one control rod only may be withdrawn when the Source Range Monitor indicate at least 3 cps and the refueling crane is not over the reactor; also, the main steam line isolation scram and main condenser low vacuum scram are bypassed if reactor vessel pressure is below 1055 psig. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

N. Rated Power - Rated power refers to operation at a reactor power of 3,293 MWt; this is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power. Design power, the power to which the safety analysis applies, corresponds to 3440 MWt.

O. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All non-automatic containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed. These valves may be opened to perform necessary operational activities.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

P. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

Amendment No. 46, 62

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	Modes in Which Function Must Be Operable: Shut-down	Refuel(7)	Startup/Hot Standby	Run	Action(1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤ 120/125 Indicated on scale	X(22)	X (22)	X	(5)	1.A
3	Inoperative			X	X	(5)	1.A
	APRM (16)						
2	High Flux	See Spec. 2.1.A.1				X	1.A or 1.B
2	High Flux	≤ 15% rated power (13)		X(21)	X(17)	(15)	1.A or 1.B
2	Inoperative			X(21)	X(17)	X	1.A or 1.B
2	Downscale	≥ 3 Indicated on Scale		(11)	(11)	X(12)	1.A or 1.B
2	High Reactor Pressure	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14)	≤ 2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14)	≥ 538" above vessel zero		X	X	X	1.A
2	High Water Level in Scram Discharge Tank	≤ 50 Gallons	X	X(2)	X	X	1.A

TABLE 3.1.A (Continued)

Min. No. of Operable Inst. Channels Per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable			Action(1)
			Refuel(7)	Startup/Hot Standby	Run	
4	Main Steam Line Isolation Valve Closure	$\leq$ 10% Valve Closure	X(3)(6)	X(3)(6)	X(6)	1.A or 1.C
2	Turbine Cont. Valve Fast Closure	Upon trip of the fast acting solenoid valves	X(4)	X(4)	X(4)	1.A or 1.D
4	Turbine Stop Valve Closure	$\leq$ 10% Valve Closure	X(4)	X(4)	X(4)	1.A or 1.D
2	Turbine Control Valve - Loss of Control Oil Pressure	$\geq$ 550 psig	X(4)	X(4)	X(4)	1.A or 1.D
2	Turbine First Stage Pressure Permissive	$\leq$ 154 psig	X(18)	X(18)	X(18)	(19)
2	Turbine Condenser Low Vacuum	$\geq$ 23 In. Hg, Vacuum	X(3)	X(3)	X	1.A or 1.C
2	Main Steam Line High Radiation (14)	$\leq$ 3X Normal Full Power Background (20)	X(9)	X(9)	X(9)	1.A or 1.C

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

7. A train is considered a trip system.

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

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

TABLE 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero.	A	1. Below trip setting initiated HPCI.
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #1)	≥ 378" above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #2)	≥ 378" above vessel zero.	A	1. Below trip settings in conjunction with drywell high pressure, low water level permissive, 120 sec. delay timer and CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184 & 185, SW #1)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LITS-3-52 & 62, SW #1)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.
2	Instrument Channel - Drywell High Pressure (PS-64-58 E-H)	1≤ P≤2.5 psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Drywell High Pressure (PS-64-58 A-D, SW #2)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal.(15)
2	Instrument Channel - Reactor Low Water Level (LS-3-56A, B, C, D)	≥ 470" above vessel zero	A	1. Below trip setting trips recirculation pumps
2	Instrument Channel Reactor High Pressure (PS-3-204 A, B, C, D)	≤ 1120 psig	A	1. Above trip setting trips recirculation pumps
2	Instrument Channel - Drywell High Pressure (PS-64-58A-D, SW #1)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
2(16)	Instrument Channel - Drywell High Pressure (PS-64-57A-D)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor water level, drywell high pressure, 120 sec. delay timer and CSS or RHR pump running, initiates ADS.
2	Instrument Channel - Reactor Low Pressure (PS-3-74 A & B, SW #2) (PS-68-95, SW #2) (PS-68-96, SW #2)	450 psig ± 15	A	1. Below trip setting permissive for opening CSS and LPCI admission valves.
2	Instrument Channel - Reactor Low Pressure (PS-3-74A & B, SW #1) (PS-68-95, SW #1) (PS-68-96, SW #1)	230 psig ± 15	A	1. Recirculation discharge valve actuation.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
1	Instrument Channel - Reactor Low Pressure (PS-68-93 & 94, SW #1)	100 psig ± 15	A	1. Below trip setting in conjunction with containment isolation signal and both suction valves open will close RHR (LPCI) admission valves.
2	Core Spray Auto Sequencing Timers (5)	6 ≤ t ≤ 8 secs.	B	1. With diesel power 2. One per motor
2	LPCI Auto Sequencing Timers (5)	0 ≤ t ≤ 1 sec.	B	1. With diesel power 2. One per motor
1	RHRSW A1, B1, C3, and D1 Timers	13 ≤ t ≤ 15 sec.	A	1. With diesel power 2. One per pump
2	Core Spray and LPCI Auto Sequencing Timers (6)	0 ≤ t ≤ 1 sec. 6 ≤ t ≤ 8 sec. 12 ≤ t ≤ 16 sec. 18 ≤ t ≤ [illegible] sec.	B	1. With normal power 2. One per CSS motor 3. Two per RHR motor
1	RHRSW A3, B1, C3, and D1 Timers	27 ≤ t ≤ 29 sec.	A	1. With normal power 2. One per pump

TABLE 3.2.E
INSTRUMENTATION THAT MONITORS LEAKAGE INTO DRYWELL

System (2)	Setpoints		Action	Remarks
Equipment Drain			(1)	1. Used to determine identifiable reactor coolant leakage. 2. Considered part of sump system.
Flow Integrator	N/A			
Sump Fill Rate Timer	≥20.1 min.			
Sump Pump Out Rate Timer	≤13.4 min.			
Floor Drain			(1)	1. Used to determine unidentifiable reactor coolant leakage. 2. Considered part of sump system.
Flow Integrator	N/A			
Sump Fill Rate Timer	≥80.4 min.			
Sump Pump Out Rate Timer	≤8.9 min.			
Drywell Air Sampling	Gas and Particulate	3 x Average Background	(3)	

NOTES:

(1) Whenever a system is required to be operable, there shall be one operable system either automatic or manual, or the action required in Section 3.6.C.2 shall be taken.

(2) An alternate system to determine the leakage flow is a manual system whereby the time between sump pump starts is monitored. The time interval will determine the leakage flow because the volume of the sump will be known.

(3) Upon receipt of alarm, immediate action will be taken to confirm the alarm and assess the possibility of increased leakage.

TABLE 3.2.F
SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	LI-3-46 A LI-3-46 B	Reactor Water Level	Indicator -107.5" to +107.5"	(1) (2) (3)
2	PI-3-54 PI-3-61	Reactor Pressure	Indicator 0-1200 psig	(1) (2) (3)
2	PR-64-50 PI-64-67	Drywell Pressure	Recorder 0-80 psia Indicator 0-80 psia	(1) (2) (3)
2	TI-64-52 TR-64-52	Drywell Temperature	Recorder, Indicator 0-400°F	(1) (2) (3)
1	TR-64-52	Suppression Chamber Air Temperature	Recorder 0-400°F	(1) (2) (3)
2	TI-64-55 TIS-64-55	Suppression Chamber Water Temperature	Indicator, 0-400°F	(1) (2) (3)
2	LI-64-54 A LI-64-66	Suppression Chamber Water Level	Indicator -25" to +25"	(1) (2) (3)
1	NA	Control Rod Position	6V Indicating Lights)	
1	NA	Neutron Monitoring	SRM, IRM, LPRM 0 to 100% power)	(1) (2) (3) (4)
1	PS-64-67	Drywell Pressure	Alarm at 35 psig)	
1	TR-64-52 and PS-64-58 B and IS-64-67	Drywell Temperature and Pressure and Timer	Alarm if temp. > 281°F and pressure > 2.5 psig after 30 minute delay)	(1) (2) (3) (4)
1	LI-84-2A	CAD tank "A" level	Indicator 0 to 100%	(1)
1	LI-84-13A	CAD tank "B" level	Indicator 0 to 100%	(1)

3.6.A Thermal and Pressurization Limitations

3. During heatup by non-nuclear means, except when the vessel is vented or as indicated in 3.6.A.4, cooldown following nuclear shutdown on low-level physics tests, the reactor vessel temperatures shall be at or above the temperatures of curve #2 of figure 3.6.1.

4. The reactor vessel shell temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on curve #1 of figure 3.6-1. The applicability of this curve to these tests is extended to non-nuclear heatup and ambient loss cooldown associated with these tests only if the heatup and cooldown rates do not exceed 15°F per hour.

5. The reactor vessel head bolting studs may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are above 70°F. Before loading the flanges any more, the vessel flange and head flange must be greater than 100°F, and must remain above 100°F while under full tension.

6. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

7. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

4.6.A Thermal and Pressurization Limitations

3. Test specimens representing the reactor vessel, base weld and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The number and type of specimens will be in accordance with GE report NEDO-10115. The specimens shall meet the intent of ASTM E 185-70. Samples shall be withdrawn at one-fourth and three-fourths service life.

4. Neutron flux wires shall be installed in the reactor vessel adjacent to the reactor vessel wall at the core midplane level. The wires shall be removed and tested during the first refueling outage to experimentally verify the calculated values of neutron fluence at one-fourth of the beltline shell thickness that are used to determine the NDTT shift from Figure 3.6-2.

5. When the reactor vessel head bolting studs are tensioned and the reactor is in a cold condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

6. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

7. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

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.

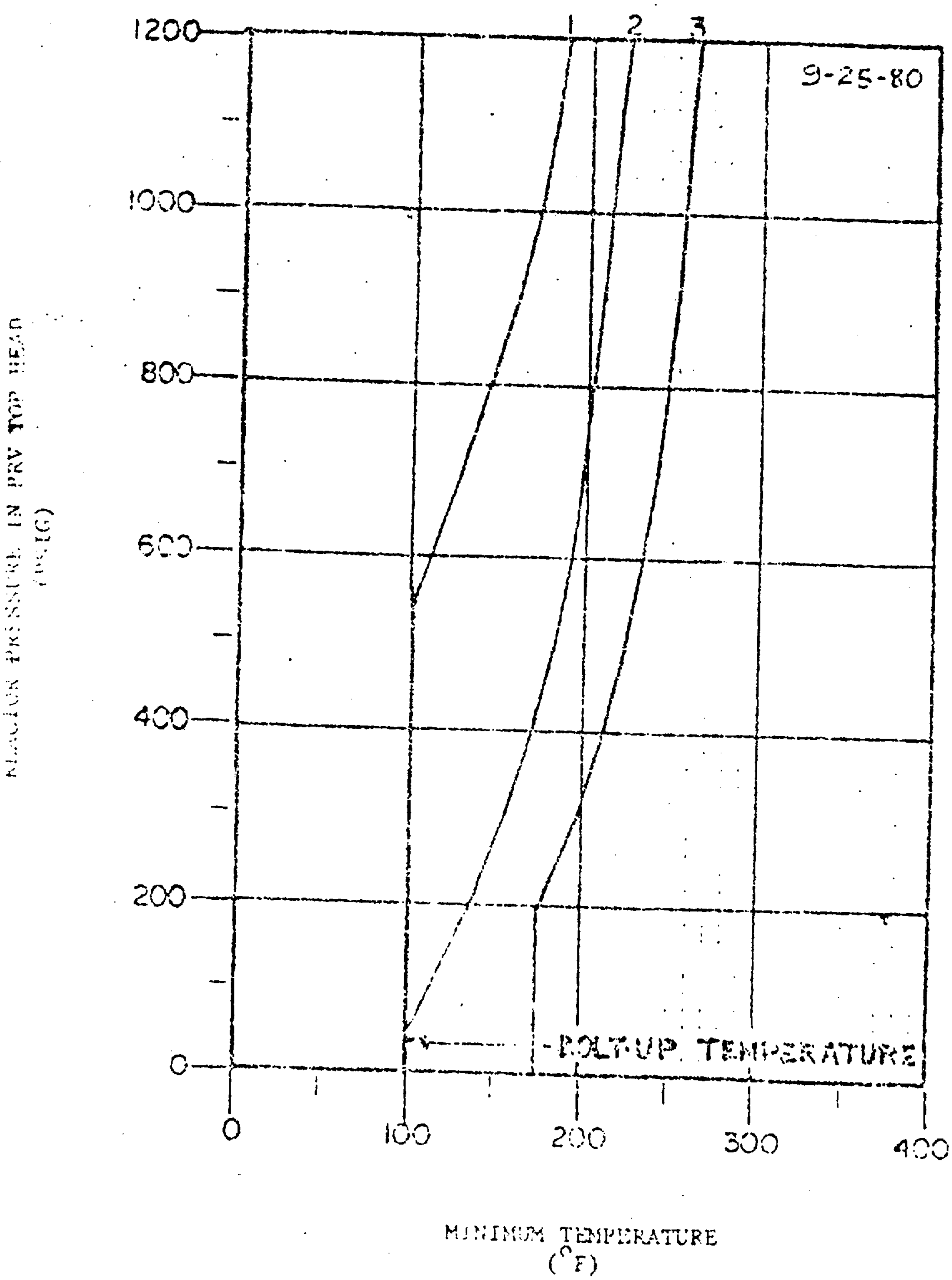
5. If a snubber is determined to be inoperable while the reactor is in the shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup.

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% of 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.

Figure 3.6-1



Curve #1
Minimum temperature for pressure tests such as required by Section XI.

Curve #2
Minimum temperature for mechanical heat up or cooldown following nuclear shutdown.

Curve #3
Minimum temperature for core operation (criticality) Includes additional margin required by 10CFR50 Appendix G, Par. IV A.2.C.

Notes
These curves are shifted 30°F to the right of the original set of curves to include a ΔRT_{NDT} of 30°F. This shift will allow these curves to used thru 4.0 EFPY.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
License No. DPR-68

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

   A. The applications for amendment by Tennessee Valley Authority (the licensee) dated June 13, 1980 and October 16, 1980, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, the provisions of the Act, and the 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-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. 38, 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

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 6, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 38

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

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

2
3
32
65
81
185
186
201
219
354

2. Add the following new page:

2a

The marginal lines on the above pages indicate the area being changed.

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Safety Limit The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit requires unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

B. Limiting Safety System Setting (LSSS)- The limiting safety system setting are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represent margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

C. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

1. In the event a Limiting Condition and/or associated requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours unless corrective measures are completed that permit operation under the permissible discovery or until the reactor is placed in an operational condition in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications. This provides action to be taken for circumstances not directly provided for in the specifications and whose occurrence would violate the intent of the specification. For example, if a specification calls for two systems (or subsystems) to be operable and provides for explicit requirements if one system (or sub-systems) is inoperable, then if both systems (or subsystems) are inoperable, the unit is to be in at least Hot Standby in 6 hours and in Cold Shutdown within the following 30 hours if the operable condition is not corrected.

2. When a system, subsystem, train, component or device is determined to be inoperable solely because its onsite power source is inoperable, or solely because its offsite power source is inoperable, it may be considered opererable for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding offsite or diesel power source is operable and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are operable, or likewise satisfy these requirements. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least Hot Standby within 6 hours, and in at least Cold Shutdown within the following 30 hours. This is not applicable if the unit is already in Cold Shutdown or Refueling. This provision describes what additional conditions must be satisfied to permit operation to continue consistent with the specifications for power sources, when offsite or onsite power sources are not operable. It specifically prohibits operation when one division is inoperable because its offsite or diesel power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason. This provision permits the requirements associated with individual systems, subsystems, trains, components or devices to be consistent with the requirements of the associated electrical power source. It allows operation to be governed by the time limits of the requirements associated with the Limiting Condition for Operation for the offsite or diesel power source, not the individual requirements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its offsite or diesel power source.

D. DELETED

E. Operable - Operability - A system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

F. Operating - Operating means that a system or component is performing its intended functions in its required manner.

G. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and and importance of the required action.

H. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup" or "Run" position with the reactor critical and above 1% rated power.

I. Hot Standby Condition - Hot standby condition means operation with coolant temperature greater than 212°F, system pressure less than 1055 psig, the main steam isolation valves closed and the mode switch in the Startup/Hot Standby position.

J. Cold Condition - Reactor coolant temperature equal to or less than 212°F.

K. Hot Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature greater than 212°F.

L. Cold Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature equal to or less than 212°F.

M. Mode of Operation - A reactor mode switch selects the proper interlocks for the operational status of the unit. The following are the modes and interlocks provided:

1. Startup/Hot Standby Mode - In this mode the reactor protection scram trips initiated by condenser low vacuum and main steam line isolation valve closure, are bypassed when reactor pressure is less than 1055 psig, the reactor protection system is energized with IRM neutron monitoring system trip, the APRM 15% high flux trip, and control rod withdrawal interlocks in service. This is often referred to as just Startup Mode. This is intended to imply the Startup/Hot Standby position of the mode switch.

2. Run Mode - In this mode the reactor system pressure is at or above 850 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RBM interlocks in service.

3. Shutdown Mode - Placing the mode switch to the shutdown position initiates a reactor scram and power to the control rod drives is removed. After a short time period (about 10 sec), the scram signal is removed allowing a scram reset and restoring the normal valve lineup in the control rod drive hydraulic system; also, the main steam line isolation scram and main condenser low vacuum scram are bypassed if reactor vessel pressure is below 1055 psig.

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	Modes in Which Function Must Be Operable: Shut-down	Refuel (7)	Startup/Hot Standby	Run	Action(1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤ 120/125 Indicated on scale	X(22)	X (22)	X	(5)	1.A
3	Inoperative			X	X	(5)	1.A
	APRM (16)						
2	High Flux	See Spec. 2.1.A.1				X	1.A or 1.B
2	High Flux	≤ 15% rated power (13)		X (21)	X(17)	(15)	1.A or 1.B
2	Inoperative			X (21)	X(17)	X	1.A or 1.B
2	Downscale	≥ 3 Indicated on Scale		(11)	(11)	X(12)	1.A or 1.B
2	High Reactor Pressure	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14)	≤ 2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14)	≥ 538" above vessel zero		X	X	X	1.A
2	High Water Level in Scram Discharge Tank	≤ 50 Gallons	X	X(2)	X	X	1.A
4	Main Steam Line Isolation Valve Closure	≤ 10% Valve Closure		X(3) (6)	X(3) (6)	X(6)	1.A or 1.C
2	Turbine Cont. Valve Fast Closure	Upon trip of the fast acting solenoid valves		X(4)	X(4)	X(4)	1.A or 1.D

Table 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Drywell High Pressure (PS-64-58 E-H)	1≤p≤ 2.5 psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.
2	Instrument Channel - Drywell High Pressure (PS-64-58 A-D, SW #2)	≤2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal. (15)
2	Instrument Channel - Reactor Low Water Level (LS-3-56A, B, C, D)	≥ 470" above vessel zero	A	1. Below trip setting trips recirculation pumps
2	Instrument Channel Reactor High Pressure (PS-3-204 A, B, C, D)	≤1120 psig	A	1. Above trip setting trips recirculation pumps
2	Instrument Channel - Drywell High Pressure (PS-64-58A-D, SW #1)	≤2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
2(16)	Instrument Channel - Drywell High Pressure (PS-64-57A-D)	≤2.5 psig	A	1. Above trip setting in conjunction with low reactor water level, drywell high pressure, 120 sec, delay timer and CSS or RHR pump running, initiates ADS.

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TABLE 3.2.F
SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	LI-3-46 A LI-3-46 B	Reactor Water Level	Indicator -107.5" to +107.5"	(1) (2) (3)
2	PI-3-54 PI-3-61	Reactor Pressure	Indicator 0-1200 psig	(1) (2) (3)
2	PR-64-50 PI-64-67	Drywell Pressure	Recorder 0-80 psia Indicator 0-80 psia	(1) (2) (3)
2	TI-64-52 TR-64-52	Drywell Temperature	Recorder, Indicator 0-400°F	(1) (2) (3)
1	TR-64-52	Suppression Chamber Air Temperature	Recorder 0-400°F	(1) (2) (3)
2	TI-64-55 TIS-64-55	Suppression Chamber Water Temperature	Indicator, 0-400°F	(1) (2) (3)
2	LI-64-54 A LI-64-66	Suppression Chamber Water Level	Indicator -25" to +25"	(1) (2) (3)
1	N/A	Control Rod Position	6V Indicating Lights SRM, IRM, LPRM 0 to 100% power	(1) (2) (3) (4)
1	N/A	Neutron Monitoring		
1	PS-64-67	Drywell Pressure	Alarm at 35 psig Alarm if temp. > 281°F and pressure >2.5 psig after 30 minute delay	(1) (2) (3) (4)
1	TR-64-52 and PS-64-58 B and IS-64-67	Drywell Temperature and Pressure and Timer		
1	LI-84-2A	CAD Tank "A" Level	Indicator 0 to 100%	(1)
1	LI-84-13A	CAD Tank "B" Level	Indicator 0 to 100%	(1)

3.6 PRIMARY SYSTEM BOUNDARY

2. During all operations with a critical core, other than for low level physics tests, the reactor vessel shell and fluid temperatures shall be at or above the temperature of curve Number 3 of figure 3.6-1.

3. During heatup by non-nuclear means, except when the vessel is vented or as indicated in 3.6.A.4, cooldown following nuclear shutdown on low-level physics tests, the reactor vessel temperatures shall be at or above the temperatures of curve Number 2 of figure 3.6-1.

4.6 PRIMARY SYSTEM BOUNDARY

d. Reactor vessel bottom head temperature

e. Reactor vessel shell adjacent to shell flange

2. Reactor vessel metal temperature at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to shell flange, shall be recorded at least every 15 minutes during inservice hydrostatic or leak testing when the vessl pressure is > 312 psig.

3. Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The number and type of specimens will be in accordance with GE report NEDO-10115. The specimens shall meet the intent of ASTM E 185-70. Samples shall be withdrawn at one-fourth and three-fourths service life.

3.6 PRIMARY SYSTEM BOUNDARY

4. The reactor vessel shell temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on curve Number 1 of figure 3.6-1. The applicability of this curve to these tests is extended to non-nuclear heatup and ambient loss cool-down associated with these tests only if the heatup and cooldown rates do not exceed 15°F per hour.

5. The reactor vessel head bolting studs may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are above 70°F. Before loading the flanges any more, the vessel flange and head flange must be greater than 100°F, and must remain above 100°F while under full tension.

6. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

7. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and bottom head drain are within 145°F.

4.6 PRIMARY SYSTEM BOUNDARY

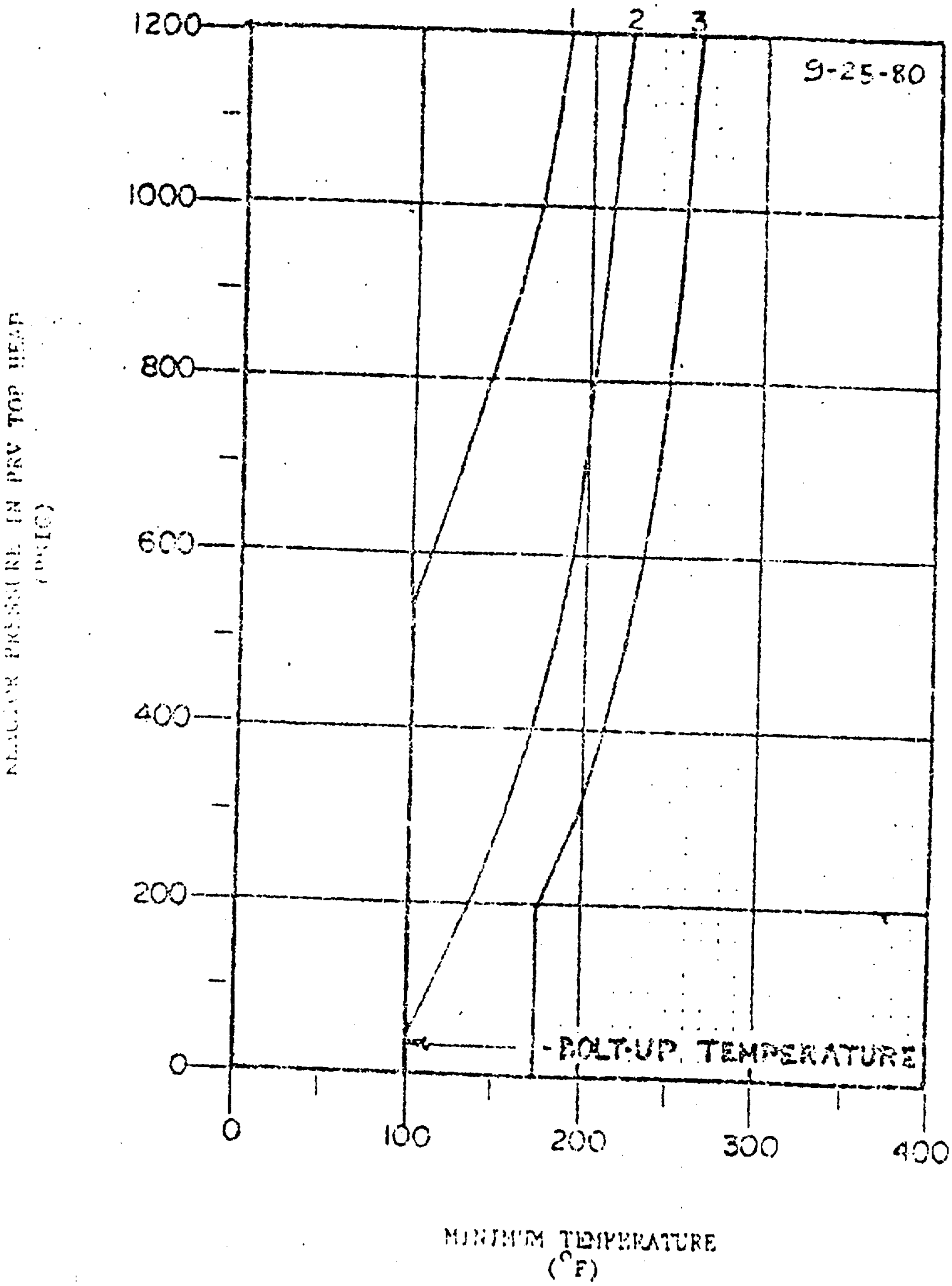
4. Neutron flux wires shall be installed in the reactor vessel adjacent to the reactor vessel wall at the core midplane level. The wires shall be removed and tested during the first refueling outage to experimentally verify the calculated values of integrated neutron fluence of one-fourth of the belt line shell thickness that are used to determine the NDTT shift from Figure 3.6-2.

5. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

6. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

7. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

Figure 3.6-1



Curve #1
Minimum temperature for pressure tests such as required by Section XI.

Curve #2
Minimum temperature for mechanical heat up or cooldown following nuclear shutdown.

Curve #3
Minimum temperature for core operation (criticality) Includes additional margin required by 10CFR50 Appendix G, Par. IV A.2.C.

Notes
These curves are shifted 30°F to the right of the original set of curves to include a ΔRT_{NDT} of 30°F. This shift will allow these curves to used thru 4.0 EFPY.

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	
R-64	RHRSW	582				X
R-62	RHRSW	582				X

3.11 FIRE PROTECTION SYSTEMS

E. Fire Protection System Inspection

1. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified TVA personnel or an outside fire protection firm.

2. An inspection and audit by an outside qualified fire consultant will be performed at intervals no greater than 3 years. (The first inspection and audit will be during the period of June - September 1977).

F. If it becomes necessary to breach a fire stop, an attendant shall be posted on each aide of the open penetration until work is completed and the penetration is resealed.

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

4.11 FIRE PROTECTION SYSTEMS

E. Fire Protection Systems Inspection

Any inspection or audit will review and evaluate the effectiveness of fire prevention and protection by physical inspection of plant facilities, systems, and equipment as related to fire safety. Evaluations will be made of, but not necessarily limited to, the following:

Administrative control documentation, maintenance of fire related records, physical plant inspection, related historical research and application, and management interviews.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 66 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 62 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

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

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

1.0 Introduction

By letters dated June 13, 1980 (TVA BFNP TS 139) and October 16, 1980 (TVA BFNP TS 152), the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Unit Nos. 1, 2 and 3.

The letter of June 13, 1980 was in response to our generic letter of April 10, 1980 to "all power reactor licenses" requesting that they submit proposed changes to their technical specifications that incorporate the requirements of the Model Technical Specifications (which were enclosed with our generic letter) to employ an explicit definition of the term OPERABLE for all components of safety related systems. The letter of October 16, 1980 was a request to change the definition of "Cold Shutdown" to make it consistent with the "BWR Standard Technical Specification." Thus, the proposed amendments and revised Technical Specifications would make the Browns Ferry Units 1, 2 and 3 Technical Specifications consistent with the NRC Standard Technical Specifications in defining operability of safety related systems and cold shutdown.

2.0 Evaluation

Our generic letter of April 10, 1980 discusses the basis for our request that all licensees review and, as necessary, revise the technical specifications for their facilities. Basically, the thrust of the model technical specifications is to insure that there are not only limiting conditions of operation (LCOs) that require all redundant components of safety related systems to be operable but that the LCOs address multiple outages of redundant components and the effects of outages of any support systems - such as electrical power or cooling water - that are relied upon to maintain the OPERABILITY of the particular system. The changes to the definitions of "Limiting Conditions for Operation" and "Operability" proposed in TVA's submittal of June 13, 1980 are consistent with the definitions in the Model Technical Specifications enclosed with our generic letter of April 10, 1980. The proposed changes are acceptable.

In the submittal of June 13, 1980, TVA also proposed several additional administrative type changes to the Technical Specifications for Units 1, 2 and 3 to correct errors or omissions. Each of the proposed changes are discussed below.

On page 16 of the Unit 1 Technical Specifications, the value for the safety limit minimum critical power ratio (SLMCPR) shown in parenthesis is the old value of 1.06. This limit only applied during the first fuel cycle. Once 8x8 fuel was added to the core, the SLMCPR became 1.07; a SLMCPR of 1.07 is the correct and accepted value for all three units. The proposed change corrects an administrative error and is acceptable.

By letter dated November 9, 1979, we issued Amendment Nos. 53, 49 and 26 to Facility License Nos. DPR-33, DPR-52 and DPR-68 in response to TVA's application of August 27, 1979. The amendments chan[illegible]e Technical Specifications to increase the high drywell pressure [illegible] level setpoint from 2.0 psig to 2.5 psig. Besides the changes in T[illegible]plication of August 27, 1979, there were additional pressure swit[illegible] for which the ranges shown in the Technical Specifications should have been changed by the amendments but which TVA failed to include in their application. The proposed changes on pages 33, 62, 63 and 78 of the Technical Specifications for Units 1 and 2 and the proposed changes on pages 32, 65 and 81 of the Technical Specifications for Unit 3 correct these omissions. The changes are acceptable.

The proposed change on page 219 of the Unit 3 Technical Specifications is to add two new snubbers to the list of snubbers to be inspected. The two new snubbers are for the residual heat removal service water system. The proposed additions are acceptable.

The proposed changes on page 188 of the Unit 1 and 2 Technical Specifications and on page 201 of the Unit 3 Technical Specifications is to substitute a revised Nil Ductility Temperature (NDT) operating curve. These curves specify the minimum temperature that the reactor vessel must be at for the range of primary coolant pressures. The upper portion of the curves provides an additional 20°F shift from the original curves for protection because of uncertainty of radiation damage. The lower portion of curves 2 and 3 reflect the limiting conditions for protection of the feedwater nozzles from degradation. This lower portion includes the 40°F conservatism for nuclear heatup. These proposed curves are more conservative than those in the present technical specifications and have been administratively imposed. The proposed changes are acceptable.

The change on p. 354 of the Unit 3 Fire Protection Inspection program is to change the time frame for the audit to make it consistent with the time frame specified for Units 1 and 2. The proposed change is acceptable. The audit by an outside consultant for the first 3 year period has been completed.

3.0 Environmental Considerations

We have determined that these 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 these amendments involve an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 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 these amendments.

4.0 Conclusion

We have concluded 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 6, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 66 to Facility Operating License No. DPR-33, Amendment No. 62 to Facility Operating License No. DPR-52 and Amendment No. 38 to Facility Operating License No. DPR-68 issued to Tennessee Valley Authority (the licensee), for operation of the Browns Ferry Nuclear Plant, Unit Nos. 1, 2 and 3, located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

The amendments change the Technical Specifications to: (1) revise the definitions for "Limiting Conditions for Operation" and "Operability" to be consistent with the definitions in the model technical specifications enclosed with the Commission's generic letter of April 10, 1980 to "All Power Reactor Licensees" and (2) revise the definition of "Cold Shutdown" to be consistent with the "Standard BWR Technical Specifications."

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the applications for amendments dated June 13, 1980 and October 16, 1980, (2) Amendment No. 66 to License No. DPR-33, Amendment No. 62 to License No. DPR-52, and Amendment No. 38 to License No. DPR-68, 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, NW., 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 Licensing.

Dated at Bethesda, Maryland, this 6th day of February 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing