February 7, 19

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

Mr. Oliver D. Kingsley, Jr.
Senior Vice President, Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. Kingsley:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. 76836, 76837 AND 76838) (TS 280)

The Commission has issued the enclosed Amendment Nos. 180, 190, and 152 to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3, respectively. These amendments are in response to your application dated May 18, 1990, as superseded by your letter of October 30, 1990, with regard to revised system operability requirements for cold shutdown conditions and to establish a more conservative maximum operating power level when the recirculation pump trip feature is inoperable.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

Thierry M. Ross

Thierry M. Ross, Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 180 to License No. DPR-33
2. Amendment No. 190 to License No. DPR-52
3. Amendment No. 152 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

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cP/ful

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AMENDMENT NO. 180 FOR BROWNS FERRY UNIT 1 - DOCKET NO. 50-259,
AMENDMENT NO. 190 FOR BROWNS FERRY UNIT 2 - DOCKET NO. 50-260, and
AMENDMENT NO. 152 FOR BROWNS FERRY UNIT 3 - DOCKET NO. 50-296
DATED: February 7, 1991

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 180
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated May 18, 1990, as superseded by your letter of October 30, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-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. 180 , 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 its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 7, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 180

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf* and spillover** pages are provided to maintain document completeness.

REMOVE	INSERT
3.2/4.2-14	3.2/4.2-14
3.2/4.2-15	3.2/4.2-15
3.2/4.2-23	3.2/4.2-23*
3.2/4.2-24	3.2/4.2-24
3.5/4.5-7	3.5/4.5-7
3.5/4.5-8	3.5/4.5-8*
3.5/4.5-12	3.5/4.5-12*
3.5/4.5-13	3.5/4.5-13
3.5/4.5-14	3.5/4.5-14
3.5/4.5-15	3.5/4.5-15*
3.5/4.5-16	3.5/4.5-16
3.5/4.5-17	3.5/4.5-17**
3.6/4.6-9	3.6/4.6-9*
3.6/4.6-10	3.6/4.6-10
3.6/4.6-30	3.6/4.6-30*
3.6/4.6-31	3.6/4.6-31
3.6/4.6-32	3.6/4.6-32**
3.6/4.6-33	3.6/4.6-33**

TABLE 3.2.B
INSTRUMENTATION THAT INITATES 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	$\geq$ 470" above vessel zero	A	1. Below trip setting initiates HPCI.
2	Instrument Channel - Reactor Low Water Level	$\geq$ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #1)	$\geq$ 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)	$\geq$ 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)	$\geq$ 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 and 62, SW #1)	$\geq$ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2(18)	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(18)	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(18)	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)(18)	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.

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS System is required by Section 3.5 to be OPERABLE, there shall be two OPERABLE trip systems except as noted. If a requirement of the first column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the first column reduced by more than one, action B shall be taken.

   Action:

   A. Repair in 24 hours. If the function is not OPERABLE in 24 hours, take action B.

   B. Declare the system or component inoperable.

   C. Immediately take action B until power is verified on the trip system.

   D. No action required; indicators are considered redundant.

2. In only one trip system.

3. Not considered in a trip system.

4. Requires one channel from each physical location (there are 4 locations) in the steam line space.

5. With diesel power, each RHRS pump is scheduled to start immediately and each CSS pump is sequenced to start about 7 sec. later.

6. With normal power, one CSS and one RHRS pump is scheduled to start instantaneously, one CSS and one RHRS pump is sequenced to start after about 7 sec. with similar pumps starting after about 14 sec. and 21 sec., at which time the full complement of CSS and RHRS pumps would be operating.

7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of water corresponds to at least 150 percent above maximum steady state steam flow to assure that spurious isolation does not occur while ensuring the initiation of isolation following a postulated steam line break. Similarly, the HPCIS setting of 90 psi corresponds to at least 150 percent above maximum steady state flow while also ensuring the initiation of isolation following a postulated break.

8. Note 1 does not apply to this item.

9. The head tank is designed to assure that the discharge piping from the CS and RHR pumps are full. The pressure shall be maintained at or above the values listed in 3.5.H, which ensures water in the discharge piping and up to the head tank.

NOTES FOR TABLE 3.2.B (Cont'd)

10. Only one trip system for each cooler fan.

11. In only two of the four 4160-V shutdown boards. See note 13.

12. In only one of the four 4160-V shutdown boards. See note 13.

13. An emergency 4160-V shutdown board is considered a trip system.

14. RHRSW pump would be inoperable. Refer to Section 4.5.C for the requirements of a RHRSW pump being inoperable.

15. The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level ($\geq$ 378" above vessel zero) originating in the core spray system trip system.

16. The ADS circuitry is capable of accomplishing its protective action with one OPERABLE trip system. Therefore, one trip system may be taken out of service for functional testing and calibration for a period not to exceed eight hours.

17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 30 percent within four hours.

18. Not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)	4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)
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 placed in the COLD SHUTDOWN CONDITION within 24 hours.	8. No additional surveillance required.
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.	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 per Specification 1.0.MM.
10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.	10. No additional surveillance required.
11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 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.)	11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

LIMITING CONDITIONS FOR OPERATION

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

12. If one RHR pump or associated heat exchanger located on the unit cross-connection in the adjacent unit is inoperable for any reason (including valve 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 STARTUP (or closed if permitted elsewhere in these specifications).

SURVEILLANCE REQUIREMENTS

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

12. No additional surveillance required.

13. No additional surveillance required.

14. All recirculation pump discharge valves shall be tested for OPERABILITY during any period of COLD SHUTDOWN CONDITION exceeding 48 hours, if OPERABILITY tests have not been performed during the preceding 31 days.

LIMITING CONDITIONS FOR OPERATION

3.5.C <u>RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)</u>

4. One of the D1 or D2 RHRSW pumps assigned to the RHR heat exchanger supplying the standby coolant supply connection may be inoperable for a period not to exceed 30 days provided the OPERABLE pump is aligned to supply the RHR heat exchanger header and the associated diesel generator and essential control valves are OPERABLE.

5. The standby coolant supply capability may be inoperable for a period not to exceed 10 days.

6. If Specifications 3.5.C.2 through 3.5.C.5 are not met, an orderly shutdown shall be initiated and the unit placed in the COLD SHUTDOWN CONDITION within 24 hours.

7. There shall be at least 2 RHRSW pumps, associated with the selected RHR pumps, aligned for RHR heat exchanger service for each reactor vessel containing irradiated fuel.

SURVEILLANCE REQUIREMENTS

4.5.C <u>RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)</u>

4. No additional surveillance is required.

LIMITING CONDITIONS FOR OPERATION

3.5.D Equipment Area Coolers

1. The equipment area cooler associated with each RHR pump and the equipment area cooler associated with each set of core spray pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE.

2. When an equipment area cooler is not OPERABLE, the pump(s) served by that cooler must be considered inoperable for Technical Specification purposes.

E. High Pressure Coolant Injection System (HPCIS)

1. The HPCI system shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in Specification 3.5.E.2. OPERABILITY shall be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION, or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

SURVEILLANCE REQUIREMENTS

4.5.D Equipment Area Coolers

1. Each equipment area cooler is operated in conjunction with the equipment served by that particular cooler; therefore, the equipment area coolers are tested at the same frequency as the pumps which they serve.

E. High Pressure Coolant Injection System (HPCIS)

1. HPCI Subsystem testing shall be performed as follows:

a.	Simulated Automatic Actuation Test	Once/18 months
b.	Pump OPERA-BILITY	Per Specification 1.0.MM
c.	Motor Oper-ated Valve OPERABILITY	Per Specification 1.0.MM
d.	Flow Rate at normal reactor vessel operating pressure	Once/3 months

LIMITING CONDITIONS FOR OPERATION

3.5.E <u>High Pressure Coolant Injection System (HPCIS)</u>

2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS (LPCI), and RCICS are OPERABLE.

3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.

F. <u>Reactor Core Isolation Cooling System (RCICS)</u>

1. The RCICS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

SURVEILLANCE REQUIREMENTS

4.5.E <u>High Pressure Coolant Injection System (HPCIS)</u>

4.5.E.1 (Cont'd)

e. Flow Rate at 150 psig — Once/18 months

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. — Once/Month

2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

F. <u>Reactor Core Isolation Cooling System (RCICS)</u>

1. RCIC Subsystem testing shall be performed as follows:

a. Simulated Automatic Actuation Test — Once/18 months

LIMITING CONDITIONS FOR OPERATION

3.5.F Reactor Core Isolation Cooling System (RCICS)

3.5.F.1 (Cont'd)

be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

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 depressurized to less than 150 psig within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5.F Reactor Core Isolation Cooling System (RCICS)

4.5.F.1 (Cont'd)

b. Pump OPERABILITY	Per Specification 1.0.MM
c. Motor-Operated Valve OPERABILITY	Per Specification 1.0.MM
d. Flow Rate at normal reactor vessel operating pressure	Once/3 months
e. Flow Rate at 150 psig	Once/18 months

The RCIC pump shall deliver at least 600 gpm during each flow test.

f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct* position.	Once/Month

2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its normal position when a signal is present may be in a position for another mode of operation.

LIMITING CONDITIONS FOR OPERATION

3.5.G Automatic Depressurization System (ADS)

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

   (1) PRIOR TO 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 in the COLD SHUTDOWN CONDITION or 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.

3. If Specifications 3.5.G.1 and 3.5.G.2 cannot be met, an orderly shutdown will be

SURVEILLANCE REQUIREMENTS

4.5.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. No additional surveillances are required.

LIMITING CONDITIONS FOR OPERATION

3.5.G Automatic Depressurization System (ADS)

3.5.G.3 (Cont'd)

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.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

SURVEILLANCE REQUIREMENTS

4.5.G Automatic Depressurization System (ADS)

H. Maintenance of Filled Discharge Pipe

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

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined.

2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.6.C. Coolant Leakage

1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

   b. Anytime the reactor is in RUN MODE, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN MODE except as defined in 3.6.C.1.c below.

   c. During the first 24 hours in the RUN MODE following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

4.6.C. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.

LIMITING CONDITIONS FOR OPERATION

3.6.C Coolant Leakage

2. Both the sump and air sampling systems shall be OPERABLE during REACTOR POWER OPERATION. From and after the date that one of these systems is made or found to be inoperable for any reason, REACTOR POWER OPERATION is permissible only during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

   The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

D. Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

SURVEILLANCE REQUIREMENTS

4.6.C Coolant Leakage

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

D. Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

3.6.C/4.6.C (Cont'd)

suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the unit should be shut down to allow further investigation and corrective action.

The two gpm limit for coolant leakage rate increase over any 24 hour period is a limit specified by the NRC (Reference 2). This limit applies only during the RUN mode to avoid being penalized for the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pump is 50 gpm and the capacity of the drywell equipment sump pump is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

REFERENCE

1. Nuclear System Leakage Rate Limits (BFNP FSAR Subsection 4.10)
2. Safety Evaluation Report (SER) on IE Bulletin 82-03

3.6.D/4.6.D Relief Valves

To meet the safety basis, 13 relief valves have been installed on the unit with a total capacity of 84.1 percent of nuclear boiler rated steam flow at a reference pressure of (1,105 + 1 percent) psig. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves operable, results in adequate margin to the code allowable overpressure limit of 1,375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1,375 psig.

Experience in relief valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief valves are benchtested every second operating cycle to ensure that their setpoints are within the ± 1 percent tolerance. The relief valves are tested in place in accordance with Specification 1.0.MM to establish that they will open and pass steam.

3.6/4.6 BASES

3.6.D/4.6.D (Cont'd)

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION. Overpressure protection is provided during hydrostatic tests by two of the relief valves whose relief setting has been established in conformance with ASME Section XI code requirements. The capacity of one relief valve exceeds the charging capacity of the pressurization source used during hydrostatic testing. Two relief valves are used to provide redundancy.

REFERENCES

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)

2. Amendment 22 in response to AEC Question 4.2 of December 6, 1971.

3. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)

4. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilliland to F. E. Kruesi, August 29, 1973

5. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda

3.6.E/4.6.E Jet Pumps

Failure of a jet pump nozzle assembly holddown mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Also, failure of the diffuser would eliminate the capability to reflood the core to two-thirds height level following a recirculation line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5 percent, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10 percent, riser and nozzle assembly integrity has been verified.

3.6.E/4.6.E (Cont'd)

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 Recirculation Pump Operation

Steady-state operation without forced recirculation will not be permitted for more than 12 hours. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

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.

3.6.G/4.6.G (Cont'd)

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

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.

REFERENCES

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)

2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code

3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)

4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)

5. Mechanical Maintenance Instruction 46 (Mechanical Equipment, Concrete, and Structural Steel Cleaning Procedure for Residue From Plant Fire - Units 1 and 2)

6. Mechanical Maintenance Instruction 53 (Evaluation of Corrosion Damage of Piping Components Which Were Exposed to Residue From March 22, 1975 Fire)

7. Plant Safety Analysis (BFNP FSAR Subsection 4.12)

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 190
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 May 18, 1990, as superseded by your letter of October 30, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 190 , 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 its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: February 7, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 190

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf* and spillover** pages are provided to maintain document completeness.

REMOVE	INSERT
3.2/4.2-14	3.2/4.2-14
3.2/4.2-15	3.2/4.2-15
3.2/4.2-16	3.2/4.2-16*
3.2/4.2-17	3.2/4.2-17
3.2/4.2-23	3.2/4.2-23*
3.2/4.2-24	3.2/4.2-24
3.5/4.5-7	3.5/4.5-7
3.5/4.5-8	3.5/4.5-8*
3.5/4.5-12	3.5/4.5-12*
3.5/4.5-13	3.5/4.5-13
3.5/4.5-14	3.5/4.5-14
3.5/4.5-15	3.5/4.5-15*
3.5/4.5-16	3.5/4.5-16
3.5/4.5-17	3.5/4.5-17*
3.6/4.6-9	3.6/4.6-9*
3.6/4.6-10	3.6/4.6-10
3.6/4.6-30	3.6/4.6-30*
3.6/4.6-31	3.6/4.6-31
3.6/4.6-32	3.6/4.6-32**
3.6/4.6-33	3.6/4.6-33*

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 (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Below trip setting initiates HPCI.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LS-3-58A-D)	≥ 398" 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 (LS-3-58A-D)	≥ 398" above vessel zero.	A	1. Below trip settings, in conjunction with drywell high pressure, low water level permissive, ADS timer timed out and CSS or RHR pump running, initiates ADS. 2. Below trip settings, in conjunction with low reactor water level permissive, ADS timer timed out, ADS high drywell pressure bypass timer timed out, CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184, 185)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LIS-3-52 and LIS-3-62A)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2(18)	Instrument Channel - Drywell High Pressure (PIS-64-58 E-H)	1≤ p≤2.5 psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.
2(18)	Instrument Channel - Drywell High Pressure (PIS-64-58 A-D)	≤ 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(18)	Instrument Channel - Drywell High Pressure (PIS-64-58A-D)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
2(16)(18)	Instrument Channel - Drywell High Pressure (PIS-64-57A-D)	≤ 2.5 psig	A	1. Above trip setting, in conjunction with low reactor water level, low reactor water level permissive, ADS timer timed out, and CSS or RHR pump running, initiates ADS.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Pressure (PIS-3-74 A & B) (PIS-68-95, 96)	450 psig ± 15	A	1. Below trip setting permissive for opening CSS and LPCI admission valves.
2	Instrument Channel - Reactor Low Pressure (PS-3-74 A & B) (PS-68-95, 96)	230 psig ± 15	A	1. Recirculation discharge valve actuation.
2	Core Spray Auto Sequencing Timers (5)	$6 \le t \le 8$ sec.	B	1. With diesel power 2. One per motor
2	LPCI Auto Sequencing Timers (5)	$0 \le t \le 1$ sec.	B	1. With diesel power 2. One per motor
1	RHRSW A1, B3, C1, and D3 Timers	$13 \le t \le 15$ sec.	A	1. With diesel power 2. One per pump
2	Core Spray and LPCI Auto Sequencing Timers (6)	$0 \le t \le 1$ sec. $6 \le t \le 8$ sec. $12 \le t \le 16$ sec. $18 \le t \le 24$ sec.	B	1. With normal power 2. One per CSS motor 3. Two per RHR motor
1	RHRSW A1, B3, C1, and D3 Timers	$27 \le t \le 29$ 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 - RHR Discharge Pressure	100 ±10 psig	A	1. Below trip setting defers ADS actuation.
2	Instrument Channel CSS Pump Discharge Pressure	185 ±10 psig	A	1. Below trip setting defers ADS actuation.
1(3)	Core Spray Sparger to Reactor Pressure Vessel d/p	2 psid ±0.4	A	1. Alarm to detect core sparger pipe break.
1	RHR (LPCI) Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	Core Spray Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	ADS Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems and valves.

NOTES FOR TABLE 3.2.

1. Whenever any CSCS System is required by Section 3.5 to be OPERABLE, there shall be two OPERABLE trip systems except as noted. If a requirement of the first column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the first column reduced by more than one, action B shall be taken.

   Action:

   A. Repair in 24 hours. If the function is not OPERABLE in 24 hours, take action B.

   B. Declare the system or component inoperable.

   C. Immediately take action B until power is verified on the trip system.

   D. No action required; indicators are considered redundant.

   E. Within 24 hours restore the inoperable channel(s) to OPERABLE status or place the inoperable channel(s) in the tripped condition.

2. In only one trip system.

3. Not considered in a trip system.

4. Deleted.

5. With diesel power, each RHRS pump is scheduled to start immediately and each CSS pump is sequenced to start about 7 seconds later.

6. With normal power, one CSS and one RHRS pump is scheduled to start instantaneously, one CSS and one RHRS pump is sequenced to start after about 7 sec. with similar pumps starting after about 14 sec. and 21 sec., at which time the full complement of CSS and RHRS pumps would be operating.

7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of water corresponds to at least 150 percent above maximum steady state steam flow to assure that spurious isolation does not occur while ensuring the initiation of isolation following a postulated steam line break. Similarly, the HPCIS setting of 90 psi corresponds to at least 150 percent above maximum steady state flow while also ensuring the initiation of isolation following a postulated break.

8. Note 1 does not apply to this item.

9. The head tank is designed to assure that the discharge piping from the CS and RHR pumps are full. The pressure shall be maintained at or above the values listed in 3.5.H, which ensures water in the discharge piping and up to the head tank.

NOTES FOR TABLE 3.2.B (Cont'd)

10. Only one trip system for each cooler fan.

11. In only two of the four 4160-V shutdown boards. See note 13.

12. In only one of the four 4160-V shutdown boards. See note 13.

13. An emergency 4160-V shutdown board is considered a trip system.

14. RHRSW pump would be inoperable. Refer to Section 4.5.C for the requirements of a RHRSW pump being inoperable.

15. The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level ($\geq$ 398" above vessel zero) originating in the core spray system trip system.

16. The ADS circuitry is capable of accomplishing its protective action with one OPERABLE trip system. Therefore, one trip system may be taken out of service for functional testing and calibration for a period not to exceed eight hours.

17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 30 percent within four hours.

18. Not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)	4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)
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 placed in the COLD SHUTDOWN CONDITION within 24 hours.	8. No additional surveillance required.
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.	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 per Specification 1.0.MM.
10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.	10. No additional surveillance required.
11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 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.)	11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)	4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)
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 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.	12. No additional surveillance required.
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.	13. No additional surveillance required.
14. All recirculation pump discharge valves shall be OPERABLE PRIOR TO STARTUP (or closed if permitted elsewhere in these specifications).	14. All recirculation pump discharge valves shall be tested for OPERABILITY during any period of COLD SHUTDOWN CONDITION exceeding 48 hours, if OPERABILITY tests have not been performed during the preceding 31 days.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)	4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)
4. Three of the D1, D2, B1, B2 RHRSW pumps assigned to the RHR heat exchanger supplying the standby coolant supply connection may be inoperable for a period not to exceed 30 days provided the OPERABLE pump is aligned to supply the RHR heat exchanger header and the associated diesel generator and essential control valves are OPERABLE.	4. No additional surveillance is required.
5. The standby coolant supply capability may be inoperable for a period not to exceed 10 days.	
6. If Specifications 3.5.C.2 through 3.5.C.5 are not met, an orderly shutdown shall be initiated and the unit placed in the COLD SHUTDOWN CONDITION within 24 hours.	
7. There shall be at least 2 RHRSW pumps, associated with the selected RHR pumps, aligned for RHR heat exchanger service for each reactor vessel containing irradiated fuel.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.5.D Equipment Area Coolers

1. The equipment area cooler associated with each RHR pump and the equipment area cooler associated with each set of core spray pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE.

2. When an equipment area cooler is not OPERABLE, the pump(s) served by that cooler must be considered inoperable for Technical Specification purposes.

E. High Pressure Coolant Injection System (HPCIS)

1. The HPCI system shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in Specification 3.5.E.2. OPERABILITY shall be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION, or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

4.5.D Equipment Area Coolers

1. Each equipment area cooler is operated in conjunction with the equipment served by that particular cooler; therefore, the equipment area coolers are tested at the same frequency as the pumps which they serve.

E. High Pressure Coolant Injection System (HPCIS)

1. HPCI Subsystem testing shall be performed as follows:

a.	Simulated Automatic Actuation Test	Once/18 months
b.	Pump OPERABILITY	Per Specification 1.0.MM
c.	Motor Operated Valve OPERABILITY	Per Specification 1.0.MM
d.	Flow Rate at normal reactor vessel operating pressure	Once/3 months

LIMITING CONDITIONS FOR OPERATION

3.5.E High Pressure Coolant Injection System (HPCIS)

2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS(LPCI), and RCICS are OPERABLE.

3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.

F. Reactor Core Isolation Cooling System (RCICS)

1. The RCICS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

SURVEILLANCE REQUIREMENTS

4.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

e. Flow Rate at 150 psig	Once/18 months

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct* position.	Once/Month

2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

F. Reactor Core Isolation Cooling System (RCICS)

1. RCIC Subsystem testing shall be performed as follows:

a. Simulated Automatic Actuation Test	Once/18 months

LIMITING CONDITIONS FOR OPERATION

3.5.F. Reactor Core Isolation Cooling System (RCICS)

3.5.F.1 (Cont'd)

be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

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 depressurized to less than 150 psig within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5.F Reactor Core Isolation Cooling System (RCICS)

4.5.F.1 (Cont'd)

b.	Pump OPERABILITY	Per Specification 1.0.MM
c.	Motor-Operated Valve OPERABILITY	Per Specification 1.0.MM
d.	Flow Rate at normal reactor vessel operating pressure	Once/3 months
e.	Flow Rate at 150 psig	Once/18 months

The RCIC pump shall deliver at least 600 gpm during each flow test.

f.	Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct* position.	Once/Month

2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its normal position when a signal is present may be in a position for another mode of operation.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.G Automatic Depressurization System (ADS)	4.5.G Automatic Depressurization System (ADS)
1. Six valves of the Automatic Depressurization System shall be OPERABLE: (1) PRIOR TO 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 in the COLD SHUTDOWN CONDITION or as specified in 3.5.G.2 and 3.5.G.3 below.	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. With one of the above required ADS valves inoperable, provided the HPCI system, the core spray system, and the LPCI system are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least a HOT SHUTDOWN CONDITION within the next 12 hours and reduce reactor steam dome pressure to ≤105 psig within 24 hours.	2. No additional surveillances are required.
3. With two or more of the above required ADS valves inoperable, be in at least a HOT SHUTDOWN CONDITION within 12 hours and reduce reactor steam dome pressure to ≤105 psig within 24 hours.	

LIMITING CONDITIONS FOR OPERATION

3.5.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.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

Pl-75-20	48 psig
Pl-75-48	48 psig
Pl-74-51	48 psig
Pl-74-65	48 psig

SURVEILLANCE REQUIREMENTS

4.5.H. Maintenance of Filled Discharge Pipe

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

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined.

2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.C. Coolant Leakage	4.6.C. Coolant Leakage
1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.	1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.
b. Anytime the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below.	
c. During the first 24 hours in the RUN mode following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.	

LIMITING CONDITIONS FOR OPERATION

3.6.C Coolant Leakage

2. Both the sump and air sampling systems shall be OPERABLE during REACTOR POWER OPERATION. From and after the date that one of these systems is made or found to be inoperable for any reason, REACTOR POWER OPERATION is permissible only during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

   The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

D. Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

SURVEILLANCE REQUIREMENTS

4.6.C Coolant Leakage

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

D. Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

3.6.B/4.6.C (Cont'd)

five gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the unit should be shut down to allow further investigation and corrective action.

The 2 gpm limit for coolant leakage rate increases over any 24-hour period is a limit specified by the NRC (Reference 2). This limit applies only during the RUN mode to avoid being penalized for the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pump is 50 gpm and the capacity of the drywell equipment sump pump is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

REFERENCE

1. Nuclear System Leakage Rate Limits (BFNP FSAR Subsection 4.10)
2. Safety Evaluation Report (SER) on IE Bulletin 82-03

3.6.D/4.6.D Relief Valves

To meet the safety basis, 13 relief valves have been installed on the unit with a total capacity of 84.1 percent of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves OPERABLE, results in adequate margin to the code allowable overpressure limit of 1,375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1,375 psig.

Experience in relief valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief valves are benchtested every second operating cycle to ensure that their setpoints are within the ± 1 percent tolerance. The relief valves are tested in place in accordance with Specification 1.0.MM to establish that they will open and pass steam.

3.6.D/4.6.D (Cont'd)

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION. Overpressure protection is provided during hydrostatic tests by two of the relief valves whose relief setting has been established in conformance with ASME Section XI code requirements. The capacity of one relief valve exceeds the charging capacity of the pressurization source used during hydrostatic testing. Two relief valves are used to provide redundancy.

REFERENCES

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)

2. Amendment 22 in response to AEC Question 4.2 of December 6, 1971.

3. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)

4. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilleland to F. E. Kruesi, August 29, 1973

5. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda

3.6.E/4.6.E Jet Pumps

Failure of a jet pump nozzle assembly holddown mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Also, failure of the diffuser would eliminate the capability to reflood the core to two-thirds height level following a recirculation line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5 percent, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10 percent, riser and nozzle assembly integrity has been verified.

3.6.E/4.6.E (Cont'd)

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 Recirculation Pump Operation

Operation without forced recirculation is permitted for up to 12 hours when the reactor is not in the RUN mode. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring at least one recirculation pump to be operable while in the RUN mode provides protection against the potential occurrence of core thermal-hydraulic instabilities at low flow conditions.

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

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.

REFERENCES

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)

2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code

3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)

4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)

5. Mechanical Maintenance Instruction 46 (Mechanical Equipment, Concrete, and Structural Steel Cleaning Procedure for Residue From Plant Fire - Units 1 and 2)

6. Mechanical Maintenance Instruction 53 (Evaluation of Corrosion Damage of Piping Components Which Were Exposed to Residue From March 22, 1975 Fire)

7. Plant Safety Analysis (BFNP FSAR Subsection 4.12)

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 152
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated May 18, 1990, as superseded by your letter of October 30, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-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. 152 , 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 its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: February 7, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 152

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf* and spillover** pages are provided to maintain document completeness.

REMOVE	INSERT
3.2/4.2-14	3.2/4.2-14
3.2/4.2-15	3.2/4.2-15
3.2/4.2-22	3.2/4.2-22*
3.2/4.2-23	3.2/4.2-23
3.5/4.5-7	3.5/4.5-7
3.5/4.5-8	3.5/4.5-8*
3.5/4.5-12	3.5/4.5-12*
3.5/4.5-13	3.5/4.5-13
3.5/4.5-14	3.5/4.5-14
3.5/4.5-15	3.5/4.5-15*
3.5/4.5-16	3.5/4.5-16
3.5/4.5-17	3.5/4.5-17**
3.6/4.6-9	3.6/4.6-9*
3.6/4.6-10	3.6/4.6-10
3.6/4.6-30	3.6/4.6-30*
3.6/4.6-31	3.6/4.6-31
3.6/4.6-32	3.6/4.6-32**
3.6/4.6-33	3.6/4.6-33*

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 initiates 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 and 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.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2(18)	Instrument Channel - Drywell High Pressure (PS-64-58 E-H)	$1 \leq p \leq 2.5$ psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.
2(18)	Instrument Channel - Drywell High Pressure (PS-64-58 A-D, SW#2)	$\leq$ 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(18)	Instrument Channel - Drywell High Pressure (PS-64-58A-D, SW#1)	$\leq$ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
2(16)(18)	Instrument Channel - Drywell High Pressure (PS-64-57A-D)	$\leq$ 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.

NOTES FOR TABLE 3.2.:

1. Whenever any CSCS System is required by Section 3.5 to be OPERABLE, there shall be two OPERABLE trip systems except as noted. If a requirement of the first column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the first column reduced by more than one, action B shall be taken.

   Action:

   A. Repair in 24 hours. If the function is not OPERABLE in 24 hours, take action B.

   B. Declare the system or component inoperable.

   C. Immediately take action B until power is verified on the trip system.

   D. No action required; indicators are considered redundant.

2. In only one trip system.

3. Not considered in a trip system.

4. Requires one channel from each physical location (there are 4 locations) in the steam line space.

5. With diesel power, each RHRS pump is scheduled to start immediately and each CSS pump is sequenced to start about 7 seconds later.

6. With normal power, one CSS and one RHRS pump is scheduled to start instantaneously, one CSS and one RHRS pump is sequenced to start after about 7 seconds with similar pumps starting after about 14 seconds and 21 seconds, at which time the full complement of CSS and RHRS pumps would be operating.

7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of water corresponds to at least 150 percent above maximum steady state steam flow to assure that spurious isolation does not occur while ensuring the initiation of isolation following a postulated steam line break. Similarly, the HPCIS setting of 90 psi corresponds to at least 150 percent above maximum steady state flow while also ensuring the initiation of isolation following a postulated break.

8. Note 1 does not apply to this item.

9. The head tank is designed to assure that the discharge piping from the CS and RHR pumps are full. The pressure shall be maintained at or above the values listed in 3.5.H, which ensures water in the discharge piping and up to the head tank.

NOTES FOR TABLE 3 B (Continued)

10. Only one trip system for each cooler fan.

11. In only two of the four 4160-V shutdown boards. See note 13.

12. In only one of the four 4160-V shutdown boards. See note 13.

13. An emergency 4160-V shutdown board is considered a trip system.

14. RHRSW pump would be inoperable. Refer to Section 4.5.C for the requirements of a RHRSW pump being inoperable.

15. The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level ($\geq$ 378" above vessel zero) originating in the core spray system trip system.

16. The ADS circuitry is capable of accomplishing its protective action with one OPERABLE trip system. Therefore, one trip system may be taken out of service for functional testing and calibration for a period not to exceed eight hours.

17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 30 percent within four hours.

18. Not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

LIMITING CONDITIONS FOR OPERATION

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

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 placed in the COLD SHUTDOWN 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.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 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.)

SURVEILLANCE REQUIREMENTS

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

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 per Specification 1.0.MM.

10. No additional surveillance required.

11. The B and D RHR pumps on unit 2 which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)	4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)
12. If one RHR pump or associated heat exchanger located on the unit cross-connection in unit 2 is inoperable for any reason (including valve 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.	12. No additional surveillance required.
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.	13. No additional surveillance required.
14. All recirculation pump discharge valves shall be OPERABLE PRIOR TO STARTUP (or closed if permitted elsewhere in these specifications).	14. All recirculation pump discharge valves shall be tested for OPERABILITY during any period of COLD SHUTDOWN CONDITION exceeding 48 hours, if OPERABILITY tests have not been performed during the preceding 31 days.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.C <u>RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)</u>	4.5.C <u>RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)</u>
4. One of the B1 or B2 RHRSW pumps assigned to the RHR heat exchanger supplying the standby coolant supply connection may be inoperable for a period not to exceed 30 days provided the OPERABLE pump is aligned to supply the RHR heat exchanger header and the associated diesel generator and essential control valves are OPERABLE.	4. No additional surveillance is required.
5. The standby coolant supply capability may be inoperable for a period not to exceed 10 days.	
6. If Specifications 3.5.C.2 through 3.5.C.5 are not met, an orderly shutdown shall be initiated and the unit placed in the COLD SHUTDOWN CONDITION within 24 hours.	
7. There shall be at least 2 RHRSW pumps, associated with the selected RHR pumps, aligned for RHR heat exchanger service for each reactor vessel containing irradiated fuel.	

LIMITING CONDITIONS FOR OPERATION

3.5.D Equipment Area Coolers

1. The equipment area cooler associated with each RHR pump and the equipment area cooler associated with each set of core spray pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE.

2. When an equipment area cooler is not OPERABLE, the pump(s) served by that cooler must be considered inoperable for Technical Specification purposes.

E. High Pressure Coolant Injection System (HPCIS)

1. The HPCI system shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in Specification 3.5.E.2. OPERABILITY shall be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION, or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

SURVEILLANCE REQUIREMENTS

4.5.D Equipment Area Coolers

1. Each equipment area cooler is operated in conjunction with the equipment served by that particular cooler; therefore, the equipment area coolers are tested at the same frequency as the pumps which they serve.

E. High Pressure Coolant Injection System (HPCIS)

1. HPCI Subsystem testing shall be performed as follows:

a.	Simulated Automatic Actuation Test	Once/18 months
b.	Pump OPERA-BILITY	Per Specification 1.0.MM
c.	Motor Operated Valve OPERABILITY	Per Specification 1.0.MM
d.	Flow Rate at normal reactor vessel operating pressure	Once/3 months

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.E <u>High Pressure Coolant Injection System (HPCIS)</u>	4.5.E <u>High Pressure Coolant Injection System (HPCIS)</u>
	4.5.E.1 (Cont'd)
	e. Flow Rate at 150 psig — Once/18 months The HPCI pump shall deliver at least 5000 gpm during each flow rate test.
	f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct* position. — Once/Month
2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS (LPCI), and RCICS are OPERABLE.	2. No additional surveillances are required.
3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.	* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.
F. <u>Reactor Core Isolation Cooling System (RCICS)</u>	F. <u>Reactor Core Isolation Cooling System (RCICS)</u>
1. The RCICS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall	1. RCIC Subsystem testing shall be performed as follows: a. Simulated Automatic Actuation Test — Once/18 months

LIMITING CONDITIONS FOR OPERATION

3.5.F Reactor Core Isolation Cooling System (RCICS)

3.5.F.1 (Cont'd)

be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

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 depressurized to less than 150 psig within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5.F Reactor Core Isolation Cooling System (RCICS)

4.5.F.1 (Cont'd)

b.	Pump OPERABILITY	Per Specification 1.0.MM
c.	Motor-Operated Valve OPERABILITY	Per Specification 1.0.MM
d.	Flow Rate at normal reactor vessel operating pressure	Once/3 months
e.	Flow Rate at 150 psig	Once/18 months

The RCIC pump shall deliver at least 600 gpm during each flow test.

f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct* position. — Once/Month

2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its normal position when a signal is present may be in a position for another mode of operation.

LIMITING CONDITIONS FOR OPERATION

3.5.G **Automatic Depressurization System (ADS)**

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

   (1) PRIOR TO 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 in the COLD SHUTDOWN CONDITION or 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.

3. If Specifications 3.5.G.1 and 3.5.G.2 cannot be met, an orderly shutdown will be initiated and the reactor

SURVEILLANCE REQUIREMENTS

4.5.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. No additional surveillances are required.

LIMITING CONDITIONS FOR OPERATION

3.5.G Automatic Depressurization System (ADS)

3.5.G.3 (Cont'd)

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.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

SURVEILLANCE REQUIREMENTS

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:

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow determined.
2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.
3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.
4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

LIMITING CONDITIONS FOR OPERATION

3.6.C <u>Coolant Leakage</u>

1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

   b. Anytime the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below.

   c. During the first 24 hours in the RUN mode following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

SURVEILLANCE REQUIREMENTS

4.6.C <u>Coolant Leakage</u>

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.C Coolant Leakage	4.6.C Coolant Leakage
2. Both the sump and air sampling systems shall be OPERABLE during REACTOR POWER OPERATION. From and after the date that one of these systems is made or found to be inoperable for any reason, REACTOR POWER OPERATION is permissible only during the succeeding 24 hours for the sump system or 72 hours for the air sampling system. The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.	2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.
3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.	
D. Relief Valves	D. Relief Valves
1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.	1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle. 2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

3.6.C/4.6.C (Cont'd)

suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the unit should be shut down to allow further investigation and corrective action.

The two gpm limit for coolant leakage rate increase over any 24 hour period is a limit specified by the NRC (Reference 2). This limit applies only during the RUN mode to avoid being penalized for the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pump is 50 gpm and the capacity of the drywell equipment sump pump is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

References

1. Nuclear System Leakage Rate Limits (BFNP FSAR Subsection 4.10)
2. Safety Evaluation Report (SER) on IE Bulletin 82-03

3.6.D/4.6.D Relief Valves

To meet the safety basis, 13 relief valves have been installed on the unit with a total capacity of 83.77 percent of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves OPERABLE, results in adequate margin to the code allowable overpressure limit of 1,375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1,375 psig.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief and safety valves are benchtested every second operating cycle to ensure that their setpoints are within the ± 1 percent tolerance. The relief valves are tested in place in accordance with Specification 1.0.MM to establish that they will open and pass steam.

3.6.D/4.6.D (Cont'd)

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION. Overpressure protection is provided during hydrostatic tests by two of the relief valves whose relief setting has been established in conformance with ASME Section XI code requirements. The capacity of one relief valve exceeds the charging capacity of the pressurization source used during hydrostatic testing. Two relief valves are used to provide redundancy.

References

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)

2. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)

3. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilliland to F. E. Kruesi, August 29, 1973

3.6.E/4.6.E Jet Pumps

Failure of a jet pump nozzle assembly holddown mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Also, failure of the diffuser would eliminate the capability to reflood the core to two-thirds height level following a recirculation line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5 percent, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10 percent, riser and nozzle assembly integrity has been verified.

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

3.6.E/4.6.E (Cont'd)

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 Recirculation Pump Operation

Steady-state operation without forced recirculation will not be permitted for more than 12 hours. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50 percent 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.

3.6.G/4.6.G (Cont'd)

Only proven nondestructive testing techniques will be used.

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

References

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)

2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code

3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)

4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)

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

ENCLOSURE 4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 152 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3

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

1.0 INTRODUCTION

By letter dated May 18, 1990, as superseded by October 30, 1990, the Tennessee Valley Authority (TVA, the licensee) proposed changes to the Technical Specifications (TS) for the Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3. The proposed TS amendments would revise: (1) Table 3.2.B and Limiting Conditions for Operations (LCO) 3.5.B.11, 3.5.E.1, 3.5.F.1, 3.5.G.1, 3.6.D.1, including applicable Bases, to correct the equipment operability requirements for certain systems when the reactor is in the COLD SHUTDOWN CONDITION in order to accommodate required routine testing evaluations (e.g., hydrostatic and integrated leak rate testing), (2) Table 3.2.B to decrease the maximum operating power level allowed, with an inoperable Recirculation Pump Trip (RPT) system(s), from 85 percent to 30 percent power, and (3) Table 3.2.B to correct two typographical errors.

2.0 EVALUATION

TVA is required by TS Section 4.7.A to conduct a primary containment integrated leakrate test (ILRT) at certain frequencies. The ILRT is performed with the plant in the cold shutdown condition by pressurizing the primary containment (drywell and torus) to design basis accident pressure (49.6 psig) and monitoring pressure and temperature for a prescribed period of time. Drywell high pressure instrumentation listed in Table 3.2.B, "Instrumentation, That Initiates Or Controls The Core And Containment Cooling Systems," are required to be operable by Note 1 of the table. Note 1 states, "Whenever any CSCS System is required by Section 3.5 to be operable, there shall be two operable trip systems except as noted..." Drywell high pressure instruments have a trip level setting between 1 and 2.5 psig. To perform this test the high drywell pressure instruments would have to be inhibited to prevent unnecessary Emergency Core Cooling System (ECCS) actuation. These instruments, along with the low reactor pressure instruments provide indication of a steam leak. With the reactor in the cold shutdown condition (Reactor Coolant System (RCS) temperature less than 212°F and the reactor mode switch in the shutdown or refuel position), no steam is present to be detected, therefore it is acceptable for the drywell pressure instruments to be inoperable. The licensee proposed adding a Note 18 to the list of notes at the end of Table 3.2.B that states, "Not required to be OPERABLE in the COLD SHUTDOWN CONDITION." TVA's TS change to allow these particular instruments to be inoperable whenever the plant is in a cold shutdown condition is considered acceptable by the staff, and is also consistent with the General Electric Standard Technical Specifications (GE STS).

Once per operating cycle, the plant is required to perform an inservice hydrostatic pressure test on the reactor vessel and attached piping out to and including the first isolation valve to ensure the integrity of the RCS pressure boundary. The test is performed (1096 to 1150 psig in the dome) in exce normal operating pressure (approximately 1020 psig). An inservice leakag is required whenever the RCS pressure boundary is breached. This test is similar to the hydrostatic test, but is performed at normal operating pressur These tests are performed in the cold shutdown condition at the end of the refueling outage with fuel loaded and the reactor pressure vessel head installed.

TS LCO 3.6.D.1, "Relief Valves," states that "when more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours." This is interpreted to mean that all the relief valves are required to be operable when the reactor pressure is above 105 psig. When the reactor vessel hydrostatic test is performed, the reactor vessel pressure is increased to 1150 psig. In this condition, the current TS LCO requires operability of all relief valves since the pressure is above 105 psig. TVA proposed to add a note to state that relief valves are not required to be operable during cold shutdown conditions due to low reactor coolant enthalpy and availability of Residual Heat Removal (RHR) and Core Spray (CS) systems. TVA's TS amendment to revise relief valve operability requirements for cold shutdown conditions is considered acceptable by the staff, and is also consistent with GE STS.

During a hydrostatic test of the RCS, 11 of 13 relief valves will be disabled. The remaining 2 relief valves will be reset at higher pressures to provide an alternate means of overpressure protection. Since hydrostatic tests or inservice leakage tests are performed at cold shutdown conditions, only 2 relief valves are required to be operable. Hence, the revised TS Bases proposed by TVA which states that overpressure protection is provided during hydrostatic tests by two of the relief valves is considered acceptable by the staff.

TSs currently require the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems to be operable when RCS pressure is greater than 150 psig. This requirement was originally made because it was believed that the only time RCS pressure would be 150 psig or greater was during startup and power operation. Hydrostatic and inservice leakage tests are conducted while in cold shutdown conditions with RCS pressures greater than 150 psig for which existing TS require the HPCI and RCIC systems to be operable. If called upon to operate, these systems would not be able to perform their intended functions because they are steam-driven systems and there is no steam supply available while the plant is in a cold shutdown condition. Therefore, adding the statement "...except for COLD SHUTDOWN CONDITION..." in LCO 3.5.E.1 and 3.5.F.1 to allow these systems to be inoperable during cold shutdown conditions is considered acceptable by the staff, and is also consistent with GE STS.

The RHR crosstie is currently required to be operable when RCS pressure is greater than atmospheric (LCO 3.5.B.11). This requirement was originally intended to assure RHR crosstie operability during startup and power operation to maintain the capability for long-term reactor core and primary containment cooling independent of the RHR system operability for a given unit. This is provided in case the torus is breached and causes flooding of the RHR pumps of the affected unit. However, with the reactor in the cold shutdown condition there is no high energy potential to breach the torus. Consequently, the RHR crosstie need not be operable during cold shutdown conditions. Therefore, adding the statement, "and the reactor is not in the COLD SHUTDOWN CONDITION...," to TS LCO 3.5.B.11 as regards RHR crosstie operability is considered acceptable by the staff.

The safety function of the Automatic Depressurization System (ADS) is to reduce RCS pressure, in the event of a Loss of Coolant Accident (LCCA), to a lower pressure point where the low pressure ECCS pumps can inject water and make up for the inventory loss from the LOCA. During cold shutdown operations the RCS temperature is less than 212°F. Below this temperature there is no steam pressure to relieve, and in the event of a pressure boundary breach while performing a hydrostatic test the decrease in vessel water level would correspond to a decrease in pressure. Pressure would eventually decrease to the point where low pressure pumps could inject and provide a make up supply of water. This accomplishes the same function of the ADS, and as such the ADS is not needed to be operable during cold shutdown operations. TVA's proposed change to TS 3.5.G.1.(2), by adding the phrase "...except in the COLD SHUTDOWN CONDITION," is considered acceptable by the staff, and is also consistent with the GE STS.

The staff has reviewed the licensee's proposed changes to the plant TS LCO requirements for cold shutdown operations (See item (1) of Introduction to this safety evaluation (SE)). Based on the above review, the staff concludes that the licensee's justification to support the proposed changes to the above-mentioned sections is acceptable. These changes revise specific TS operability requirements from being pressure dependent to mode dependent, which will now accommodate ILRT and inservice hydrostatic and leakage rate testing.

The RPT provides automatic trip of both recirculation pumps after a turbine trip or generator load rejection, if reactor power is above approximately 30 percent of rated full load. The purpose of this trip is to reduce the peak reactor pressure and peak heat flux resulting from transients in which it is postulated that there is a coincident failure of the turbine bypass system. The RPT signal is initiated by either turbine control valve fast closure or turbine stop valve closure. An automatic reactor scram is also initiated by these signals. The very rapid reduction in core flow following a recirculation pump trip, early in the transient, reduces the severity of these events because an immediate increase in core voids provides negative reactivity to supplement the negative reactivity insertion from a control rod scram. The proposed TS amendment corrects the maximum operating power level allowed with an inoperable RPT system(s) from 85 percent to 30 percent Core Thermal Power (CTP). Thirty percent CTP is used in the RPT analysis (NEDO-24119, "Basis for Installation of Recirculation Pump Trip System for Browns Ferry," April 1978, and BFN Updated Final Safety Analysis Report (UFSAR) Section 7.9.4.5), and is conservatively determined to be the maximum power level at which fuel cladding integrity can be assumed during an end of cycle limiting overpressurization even without RPT protection.

The proposed TS change (See Item (2) of the Introduction to this SE) to Table 3.2.B reduces the maximum allowed operating reactor power level, with the RPT system inoperable, from 85 percent to 30 percent to make it consistent with assumptions in the current licensing basis analysis. This change is considered acceptable by the staff.

Furthermore, with regards to editorial changes (correction of typographical errors) made to TS Table 3.2.B (See Item (3) of the Introduction to this SE), the staff considers them acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

The amendments involve a change to a requirement with respect to use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration (published in the Federal Register on November 28, 1990) and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (55 FR 26295) on June 27, 1990, and (55 FR 49461) on November 28, 1990 and consulted with the State of Alabama. No public comments were received and the State of Alabama did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributors: G. Thomas and J. Harold

Dated: February 7, 1991