November 2, 1995

Mr. Oliver D. Kingsley, Jr.
President, TVA Nuclear and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS FOR THE BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3 (TAC NOS. M92499, M92500, AND M92501) (TS 361 AND TS 371)

Dear Mr. Kingsley:

The Commission has issued the enclosed Amendment Nos.225 , 240, and 199 to Facility Operating License Nos. DPR-33, DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3, respectively. These amendments are in response to your application dated June 2, 1995. The amendments revise the operability definition for residual heat removal service water components for use as a standby coolant supply. The amendments also incorporate related changes to the technical specification Bases which were submitted on October 2, 1995.

A copy of the NRC's Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by

Joseph F. Williams, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 225 to License No. DPR-33
2. Amendment No. 240 to License No. DPR-52
3. Amendment No. 199 to License No. DPR-68
4. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 225
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 June 2, 1995, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 225, 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: November 2, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 225

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
3.5/4.5-9	3.5/4.5-9*
3.5/4.5-10	3.5/4.5-10
3.5/4.5-26	3.5/4.5-26*
3.5/4.5-27	3.5/4.5-27
3.5/4.5-28	3.5/4.5-28
3.5/4.5-29	3.5/4.5-29
3.5/4.5-30	3.5/4.5-30
3.5/4.5-31	3.5/4.5-31**
3.5/4.5-32	3.5/4.5-32**
3.5/4.5-33	3.5/4.5-33**
3.5/4.5-34	3.5/4.5-34**
3.5/4.5-35	3.5/4.5-35**
---	3.5/4.5-36**
---	3.5/4.5-37

LIMITING CONDITIONS FOR OPERATION

3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)

1. PRIOR TO STARTUP from a COLD SHUTDOWN CONDITION, the RHRSW pumps, including pump D1 or D2, shall be OPERABLE and assigned to service as indicated in Table 3.5-1.

SURVEILLANCE REQUIREMENTS

4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)

1. a. Each of the RHRSW pumps normally assigned to automatic service on the EECW headers will be tested automatically each time the diesel generators are tested. Each of the RHRSW pumps and all associated essential control valves for the EECW headers and RHR heat exchanger headers shall be demonstrated to be OPERABLE in accordance with Specification 1.0.MM.

   b. Annually each RHRSW pump shall be flow-rate tested. To be considered OPERABLE, each pump shall pump at least 4500 gpm through its normally assigned flow path.

   c. Monthly verify that each valve (manual, power-operated, or automatic) in the flowpath servicing safety-related equipment in the affected unit that is not locked, sealed, or otherwise secured in position, is in its correct position.

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)
2. During REACTOR POWER OPERATION, RHRSW pumps must be OPERABLE and assigned to service as indicated in Table 3.5-1 for the specified time limits.	2. No additional surveillance is required.
3. During Unit 1 REACTOR POWER OPERATION, both RHRSW pumps D1 and D2 and associated valves normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection must be OPERABLE except as specified in 3.5.C.4 and 3.5.C.5 below. (Note: Because standby coolant supply capability is not a short-term requirement, a component is not considered inoperable if standby coolant supply capability can be restored to service within 5 hours.)	3. Routine surveillance for these pumps is specified in 4.5.C.1.

AMENDMENT No. 225

3.5 BASES

3.5.A. Core Spray System (CSS) and 3.5.B Residual Heat Removal System (RHRS)

Analyses presented in the FSAR* and analyses presented in conformance with 10 CFR 50, Appendix K, demonstrated that the core spray system in conjunction with two LPCI pumps provides adequate cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel clad temperature to below 2,200°F which assures that core geometry remains intact and to limit the core average clad metal-water reaction to less than 1 percent. Core spray distribution has been shown in tests of systems similar to design to BFNP to exceed the minimum requirements. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel.

The RHRS (LPCI mode) is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is completely independent of the core spray system; however, it does function in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI mode of the RHRS and the core spray system provide adequate cooling for break areas of approximately 0.2 square feet up to and including the double-ended recirculation line break without assistance from the high-pressure emergency core cooling subsystems.

The intent of the CSS and RHRS specifications is to not allow startup from the cold condition without all associated equipment being OPERABLE. However, during operation, certain components may be out of service for the specified allowable repair times. The allowable repair times have been selected using engineering judgment based on experiences and supported by availability analysis.

Should one core spray loop become inoperable, the remaining core spray loop, the RHR System, and the diesel generators are required to be OPERABLE should the need for core cooling arise. These provide extensive margin over the OPERABLE equipment needed for adequate core cooling. With due regard for this margin, the allowable repair time of seven days was chosen.

Should one RHR pump (LPCI mode) become inoperable, three RHR pumps (LPCI mode) and the core spray system are available. Since adequate core cooling is assured with this complement of ECCS, a seven day repair period is justified.

Should two RHR pumps (LPCI mode) become inoperable, there remains no reserve (redundant) capacity within the RHRS (LPCI mode). Therefore, the affected unit shall be placed in cold shutdown within 24 hours.

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*A detailed functional analysis is given in Section 6 of the BFNP FSAR.

Should one RHR pump (containment cooling mode) become inoperable, a complement of three full capacity containment heat removal systems is still available. Any two of the remaining pumps/heat exchanger combinations would provide more than adequate containment cooling for any abnormal or postaccident situation. Because of the availability of equipment in excess of normal redundancy requirements, a 30-day repair period is justified.

Should two RHR pumps (containment cooling mode) become inoperable, a full heat removal system is still available. The remaining pump/heat exchanger combinations would provide adequate containment cooling for any abnormal postaccident situation. Because of the availability of a full complement of heat removal equipment, a 7-day repair period is justified.

Observation of the stated requirements for the containment cooling mode assures that the suppression pool and the drywell will be sufficiently cooled, following a loss-of-coolant accident, to prevent primary containment overpressurization. The containment cooling function of the RHRS is permitted only after the core has reflooded to the two-thirds core height level. This prevents inadvertently diverting water needed for core flooding to the less urgent task of containment cooling. The two-thirds core height level interlock may be manually bypassed by a keylock switch.

Since the RHRS is filled with low quality water during power operation, it is planned that the system be filled with demineralized (condensate) water before using the shutdown cooling function of the RHR System. Since it is desirable to have the RHRS in service if a "pipe-break" type of accident should occur, it is permitted to be out of operation for only a restricted amount of time and when the system pressure is low. At least one-half of the containment cooling function must remain OPERABLE during this time period. Requiring two OPERABLE CSS pumps during cooldown allows for flushing the RHRS even if the shutdown were caused by inability to meet the CSS specifications (3.5.A) on a number of OPERABLE pumps.

When the reactor vessel pressure is atmospheric, the limiting conditions for operation are less restrictive. At atmospheric pressure, the minimum requirement is for one supply of makeup water to the core. Requiring two OPERABLE RHR pumps and one CSS pump provides redundancy to ensure makeup water availability.

Verification that the LPCI subsystem cross-tie valve is closed and power to its operator is disconnected ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other LPCI subsystem.

Since the RHR system cross-connect capability provides added long term redundancy to the other emergency and containment cooling systems, a

5-hour time to establish flow path availability is allowed. This time limit does not reduce the other requirements associated with RHR system pump operability.

Should one RHR pump or associated heat exchanger located on the unit cross-connection in the adjacent unit become inoperable, an equal capability for long-term fluid makeup to the reactor and for cooling of the containment remains OPERABLE. Because of the availability of an equal makeup and cooling capability, a 30-day repair period is justified.

The suppression chamber can be drained when the reactor vessel pressure is atmospheric, irradiated fuel is in the reactor vessel, and work is not in progress which has the potential to drain the vessel. By requiring the fuel pool gate to be open with the vessel head removed, the combined water inventory in the fuel pool, the reactor cavity, and the separator/dryer pool, between the fuel pool low level alarm and the reactor vessel flange, is about 65,800 cubic feet (492,000 gallons). This will provide adequate low-pressure cooling in lieu of CSS and RHR (LPCI and containment cooling mode) as currently required in Specifications 3.5.A.4 and 3.5.B.9. The additional requirements for providing standby coolant supply available will ensure a redundant supply of coolant supply. Control rod drive maintenance may continue during this period provided no more than one drive is removed at a time unless blind flanges are installed during the period of time CRDs are not in place.

Should the capability for providing flow through the cross-connect lines be lost, a 10-day repair time is allowed before shutdown is required. This repair time is justified based on the very small probability for ever needing RHR pumps and heat exchangers to supply an adjacent unit.

REFERENCES

1. Residual Heat Removal System (BFNP FSAR subsection 4.8)

2. Core Standby Cooling Systems (BFNP FSAR Section 6)

3.5.C. RHR Service Water System and Emergency Equipment Cooling Water System (EECWS)

The EECW has two completely redundant and independent headers (north and south) in a loop arrangement inside and outside the Reactor Building. Four RHRSW pumps, two per header, (A3, B3, C3 and D3) are dedicated to automatically supplying the EECW system needs. Four additional pumps (A1, B1, C1 and D1) can serve as RHRSW system pumps or be manually valved into the EECW system headers and serve as backup for the RHRSW pumps deciated to supplying the EECW system. Those components requiring EECW, except the control air compressors which

are not safety related, are able to be fed from both headers thus assuring continuity of operation if either header becomes inoperable. The control air compressors only use the EECW north header as an emergency backup supply.

There are four RHR heat exchanger headers (A, B, C, & D) with one RHR heat exchanger from each unit on each header. There are two RHRSW pumps on each header; one normally assigned to each header (A2, B2, C2, or D2) and one on alternate assignment (A1, B1, C1, or D1). One RHR heat exchanger header can adequately deliver the flow supplied by both RHRSW pumps to any two of the three RHR heat exchangers on the header. One RHRSW pump can supply the full flow requirement of one RHR heat exchanger. Two RHR heat exchangers can more than adequately handle the cooling requirements of one unit in any abnormal or postaccident situation.

The RHR Service Water System was designed as a shared system for three units. The specification, as written, is conservative when consideration is given to particular pumps being out of service and to possible valving arrangements. If unusual operating conditions arise such that more pumps are out of service than allowed by this specification, a special case request may be made to the NRC to allow continued operation if the actual system cooling requirements can be assured.

Since the standby coolant supply capability provides added long term redundancy to the other emergency and containment cooling systems, a 5-hour time to establish flow path availability is allowed. This time limit does not reduce the other requirements associated with RHRSW/EECW system pump operability.

Should one of the two RHRSW pumps normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection become inoperable, an equal capability for long-term fluid makeup to the unit reactor and for cooling of the unit containment remains OPERABLE. Because of the availability of an equal makeup and cooling capability, a 30-day repair period is justified. Should the capability to provide standby coolant supply be lost, a 10-day repair time is justified based on the low probability for ever needing the standby coolant supply.

With only one unit fueled, four RHRSW pumps are required to be OPERABLE for indefinite operation to meet the requirements of Specification 3.5.B.1 (RHR system). If only three RHRSW pumps are OPERABLE, a 30-day LCO exists because of the requirement of Specification 3.5.B.5 (RHR system).

3.5.D Equipment Area Coolers

There is an equipment area cooler for each RHR pump and an equipment area cooler for each set (two pumps, either the A and C or B and D pumps) of core spray pumps. The equipment area coolers take suction

near the cooling air discharge of the motor of the pump(s) served and discharge air near the cooling air suction of the motor of the pump(s) served. This ensures that cool air is supplied for cooling the pump motors.

The equipment area coolers also remove the pump, and equipment waste heat from the basement rooms housing the engineered safeguard equipment. The various conditions under which the operation of the equipment air coolers is required have been identified by evaluating the normal and abnormal operating transients and accidents over the full range of planned operations. The surveillance and testing of the equipment area coolers in each of their various modes is accomplished during the testing of the equipment served by these coolers. This testing is adequate to assure the OPERABILITY of the equipment area coolers.

REFERENCES

1. Residual Heat Removal System (BFN FSAR Section 4.8)

2. Core Standby Cooling System (BFN FSAR subsection 6.7)

3.5.E. High Pressure Coolant Injection System (HPCIS)

The HPCIS is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCIS continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or Core Spray system operation maintains core cooling. The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to pump 5000 gpm at reactor pressures between 1120 and 150 psig. The HPCIS is not required to be OPERABLE below 150 psig since this is well within the range of the low pressure cooling systems and below the pressure of any events for which HPCI is required to provide core cooling.

The minimum required NPSH for HPCI is 21 feet. There is adequate elevation head between the suppression pool and the HPCI pump, such that the required NPSH is available with a suppression pool temperature up to 140°F with no containment back pressure.

The HPCIS is not designed to operate at full capacity until reactor pressure exceeds 150 psig and the steam supply to the HPCI turbine is automatically isolated before reactor pressure decreases below 100 psig. The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the HPCI turbine for OPERABILITY testing yet, still below the shutoff head of the CSS and RHRS pumps so they will inject water into

3.5 BASES (Cont'd)

the vessel if required. The ADS provides additional backup to reduce pressure to the range where the CSS and RHRS will inject into the vessel if necessary. Considering the low reactor pressure, the redundancy and availability of CSS, RHRS, and ADS during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate HPCI OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate HPCI OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the HPCIS inoperable, a seven-day period to return the system to service is justified based on the availability of the ADS, CSS, RHRS (LPCI) and the RCICS. The availability of these redundant and diversified systems provides adequate assurance of core cooling while HPCIS is out of service.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the HPCIS will be OPERABLE when required.

3.5.F Reactor Core Isolation Cooling System (RCICS)

The RCICS functions to provide makeup water to the reactor vessel during shutdown and isolation from the main heat sink to supplement or replace the normal makeup sources. The RCICS provides its design flow between 150 psig and 1120 psig reactor pressure. Below 150 psig, RCICS is not required to be OPERABLE since this pressure is substantially below that for any events in which RCICS is needed to maintain sufficient coolant to the reactor vessel. RCICS will continue to operate below 150 psig at reduced flow until it automatically isolates at greater than or equal to 50 psig reactor steam pressure. 150 psig is also below the shutoff head of the CSS and RHRS, thus, considerable overlap exists with the cooling systems that provide core cooling at low reactor pressure. The minimum required NPSH for RCIC is 20 feet. There is adequate elevation head between the suppression pool and the RCIC pump, such that the required NPSH is available with a suppression pool temperature up to 140°F with no containment back pressure.

The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the RCIC turbine for OPERABILITY testing, yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. Considering the low reactor pressure and the availability of the low pressure coolant systems during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate RCIC OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate RCIC OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the RCICS inoperable, a seven-day period to return the system to service is justified based on the availability of the HPCIS to cool the core and upon consideration that the average risk associated with failure of the RCICS to cool the core when required is not increased.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the RCICS will be OPERABLE when required.

3.5.G Automatic Depressurization System (ADS)

The ADS consists of six of the thirteen relief valves. It is designed to provide depressurization of the reactor coolant system during a small break loss of coolant accident (LOCA) if HPCI fails or is unable to maintain the required water level in the reactor vessel. ADS operation reduces the reactor vessel pressure to within the operating pressure range of the low pressure emergency core cooling systems (core spray and LPCI) so that they can operate to protect the fuel barrier. Specification 3.5.G applies only to the automatic feature of the pressure relief system.

Specification 3.6.D specifies the requirements for the pressure relief function of the valves. It is possible for any number of the valves assigned to the ADS to be incapable of performing their ADS functions because of instrumentation failures, yet be fully capable of performing their pressure relief function.

The emergency core cooling system LOCA analyses for small line breaks assumed that four of the six ADS valves were OPERABLE. By requiring six valves to be OPERABLE, additional conservatism is provided to account for the possibility of a single failure in the ADS system.

Reactor operation with one of the six ADS valves inoperable is allowed to continue for fourteen days provided the HPCI, core spray, and LPCI systems are OPERABLE. Operation with more than one ADS valve inoperable is not acceptable.

With one ADS valve known to be incapable of automatic operation, five valves remain OPERABLE to perform the ADS function. This condition is within the analyses for a small break LOCA and the peak clad temperature is well below the 10 CFR 50.46 limit. Analysis has shown that four valves are capable of depressurizing the reactor rapidly enough to maintain peak clad temperature within acceptable limits.

H. Maintenance of Filled Discharge Pipe

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

The core spray and RHR system discharge piping high point vent is visually checked for water flow once a month and prior to testing to ensure that the lines are filled. The visual checking will avoid starting the core spray or RHR system with a discharge line not filled. In addition to the visual observation and to ensure a filled discharge line other than prior to testing, a pressure suppression chamber head tank is located approximately 20 feet above the discharge line high point to supply makeup water for these systems. The condensate head tank located approximately 100 feet above the discharge high point serves as a backup charging system when the pressure suppression chamber head tank is not in service. System discharge pressure indicators are used to determine the water level above the discharge line high point. The indicators will reflect approximately 30 psig for a water level at the high point and 45 psig for a water level in the pressure suppression chamber head tank and are monitored daily to ensure that the discharge lines are filled.

When in their normal standby condition, the suction for the HPCI and RCIC pumps are aligned to the condensate storage tank, which is physically at a higher elevation than the HPCIS and RCICS piping. This assures that the HPCI and RCIC discharge piping remains filled. Further assurance is provided by observing water flow from these systems' high points monthly.

3.5.I. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit.

3.5.J. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated.

The LHGR shall be checked daily during reactor operation at ≥ 25 percent power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 percent of rated thermal power, the largest total peaking would have to be greater than approximately 9.7 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

3.5.L. APRM Setpoints

The fuel cladding integrity safety limits of Section 2.1 were based on a total peaking factor within design limits (FRP/CMFLPD ≥ 1.0). The APRM instruments must be adjusted to ensure that the core thermal limits are not exceeded in a degraded situation when entry conditions are less conservative than design assumptions.

3.5.M. Core Thermal-Hydraulic Stability

The minimum margin to the onset of thermal-hydraulic instability occurs in Region I of Figure 3.5.M-1. A manually initiated scram upon entry into this region is sufficient to preclude core oscillations which could challenge the MCPR safety limit.

Because the probability of thermal-hydraulic oscillations is lower and the margin to the MCPR safety limit is greater in Region II than in Region I of Figure 3.5.M-1, an immediate scram upon entry into the region is not necessary. However, in order to minimize the probability of core instability following entry into Region II, the operator will take immediate action to exit the region. Although formal surveillances are not performed while exiting Region II (delaying exit for surveillances is undesirable), an immediate manual scram will be initiated if evidence of thermal-hydraulic instability is observed.

3.5 BASES (Cont'd)

Clear indications of thermal-hydraulic instability are APRM oscillations which exceed 10 percent peak-to-peak or LPRM oscillations which exceed 30 percent peak-to-peak (approximately equivalent to APRM oscillations of 10 percent during regional oscillations). Periodic LPRM upscale or downscale alarms may also be indicators of thermal hydraulic instability and will be immediately investigated.

Periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

Normal operation of the reactor is restricted to thermal power and core flow conditions (i.e., outside Regions I and II) where thermal-hydraulic instabilities are very unlikely to occur.

3.5.N. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEIM-10735, August 1973.

2. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 14, 1974 (USA Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

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

5. Letter from R. H. Buchholz (GE) to P. S. Check (NRC), "Response to NRC Request For Information On ODYN Computer Model," September 5, 1980.

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by OPERABILITY of the remaining redundant equipment.

Whenever a CSCS system or loop is made inoperable, the other CSCS systems or loops that are required to be OPERABLE shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Average Planar LHGR, LHGR, and MCPR

The APLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 240
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 June 2, 1995, 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. 240 , 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: November 2, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 240

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
3.5/4.5-9	3.5/4.5-9*
3.5/4.5-10	3.5/4.5-10
3.5/4.5-24	3.5/4.5-24*
3.5/4.5-25	3.5/4.5-25
3.5/4.5-26	3.5/4.5-26
3.5/4.5-27	3.5/4.5-27
3.5/4.5-28	3.5/4.5-28**
3.5/4.5-29	3.5/4.5-29**
3.5/4.5-30	3.5/4.5-30**
3.5/4.5-31	3.5/4.5-31**
3.5/4.5-32	3.5/4.5-32**
3.5/4.5-33	3.5/4.5-33**
---	3.5/4.5-34**
---	3.5/4.5-35

LIMITING CONDITIONS FOR OPERATION

3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)

1. PRIOR TO STARTUP from a COLD SHUTDOWN CONDITION, the RHRSW pumps, including one of pumps D1, D2, B2 or B1, shall be OPERABLE and assigned to service as indicated in Table 3.5-1.

SURVEILLANCE REQUIREMENTS

4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)

1. a. Each of the RHRSW pumps normally assigned to automatic service on the EECW headers will be tested automatically each time the diesel generators are tested. Each of the RHRSW pumps and all associated essential control valves for the EECW headers and RHR heat exchanger headers shall be demonstrated to be OPERABLE in accordance with Specification 1.0.MM.

   b. Annually each RHRSW pump shall be flow-rate tested. To be considered OPERABLE, each pump shall pump at least 4500 gpm through its normally assigned flow path.

   c. Monthly verify that each valve (manual, power-operated, or automatic) in the flowpath servicing safety-related equipment in the affected unit that is not locked, sealed, or otherwise secured in position, is in its correct position.

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)
2. During REACTOR POWER OPERATION, RHRSW pumps must be OPERABLE and assigned to service as indicated in Table 3.5-1 for the specified time limits.	2. No additional surveillance is required.
3. During Unit 2 REACTOR POWER OPERATION, any two RHRSW pumps (D1, D2, B1, and B2) and associated valves normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection must be OPERABLE except as specified in 3.5.C.4 and 3.5.C.5 below. (Note: Because standby coolant supply capability is not a short-term requirement, a component is not considered inoperable if standby coolant supply capability can be restored to service within 5 hours.)	3. Routine surveillance for these pumps is specified in 4.5.C.1.

3.5 BASES

3.5.A. Core Spray System (CSS) and 3.5.B Residual Heat Removal System (RHRS)

Analyses presented in the FSAR* and analyses presented in conformance with 10 CFR 50, Appendix K, demonstrated that the core spray system in conjunction with two LPCI pumps provides adequate cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel clad temperature to below 2,200°F which assures that core geometry remains intact and to limit the core average clad metal-water reaction to less than 1 percent. Core spray distribution has been shown in tests of systems similar to design to BFNP to exceed the minimum requirements. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel.

The RHRS (LPCI mode) is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is completely independent of the core spray system; however, it does function in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI mode of the RHRS and the core spray system provide adequate cooling for break areas of approximately 0.2 square feet up to and including the double-ended recirculation line break without assistance from the high-pressure emergency core cooling subsystems.

The intent of the CSS and RHRS specifications is to not allow startup from the cold condition without all associated equipment being OPERABLE. However, during operation, certain components may be out of service for the specified allowable repair times. The allowable repair times have been selected using engineering judgment based on experiences and supported by availability analysis.

Should one core spray loop become inoperable, the remaining core spray loop, the RHR System, and the diesel generators are required to be OPERABLE should the need for core cooling arise. These provide extensive margin over the OPERABLE equipment needed for adequate core cooling. With due regard for this margin, the allowable repair time of seven days was chosen.

Should one RHR pump (LPCI mode) become inoperable, three RHR pumps (LPCI mode) and the core spray system are available. Since adequate core cooling is assured with this complement of ECCS, a seven day repair period is justified.

Should two RHR pumps (LPCI mode) become inoperable, there remains no reserve (redundant) capacity within the RHRS (LPCI mode). Therefore, the affected unit shall be placed in cold shutdown within 24 hours.

*A detailed functional analysis is given in Section 6 of the BFNP FSAR.

Should one RHR pump (containment cooling mode) become inoperable, a complement of three full capacity containment heat removal systems is still available. Any two of the remaining pumps/heat exchanger combinations would provide more than adequate containment cooling for any abnormal or postaccident situation. Because of the availability of equipment in excess of normal redundancy requirements, a 30-day repair period is justified.

Should two RHR pumps (containment cooling mode) become inoperable, a full heat removal system is still available. The remaining pump/heat exchanger combinations would provide adequate containment cooling for any abnormal postaccident situation. Because of the availability of a full complement of heat removal equipment, a 7-day repair period is justified.

Observation of the stated requirements for the containment cooling mode assures that the suppression pool and the drywell will be sufficiently cooled, following a loss-of-coolant accident, to prevent primary containment overpressurization. The containment cooling function of the RHRS is permitted only after the core has reflooded to the two-thirds core height level. This prevents inadvertently diverting water needed for core flooding to the less urgent task of containment cooling. The two-thirds core height level interlock may be manually bypassed by a keylock switch.

Since the RHRS is filled with low quality water during power operation, it is planned that the system be filled with demineralized (condensate) water before using the shutdown cooling function of the RHR System. Since it is desirable to have the RHRS in service if a "pipe-break" type of accident should occur, it is permitted to be out of operation for only a restricted amount of time and when the system pressure is low. At least one-half of the containment cooling function must remain OPERABLE during this time period. Requiring two OPERABLE CSS pumps during cooldown allows for flushing the RHRS even if the shutdown were caused by inability to meet the CSS specifications (3.5.A) on a number of OPERABLE pumps.

When the reactor vessel pressure is atmospheric, the limiting conditions for operation are less restrictive. At atmospheric pressure, the minimum requirement is for one supply of makeup water to the core. Requiring two OPERABLE RHR pumps and one CSS pump provides redundancy to ensure makeup water availability.

Verification that the LPCI subsystem cross-tie valve is closed and power to its operator is disconnected ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other LPCI subsystem.

Since the RHR system cross-connect capability provides added long term redundancy to the other emergency and containment cooling systems, a

3.5 Bases (Cont'd)

5-hour time to establish flow path availability is allowed. This time limit does not reduce the other requirements associated with RHR system pump operability.

Should one RHR pump or associated heat exchanger located on the unit cross-connection in the adjacent unit become inoperable, an equal capability for long-term fluid makeup to the reactor and for cooling of the containment remains OPERABLE. Because of the availability of an equal makeup and cooling capability, a 30-day repair period is justified.

The suppression chamber can be drained when the reactor vessel pressure is atmospheric, irradiated fuel is in the reactor vessel, and work is not in progress which has the potential to drain the vessel. By requiring the fuel pool gate to be open with the vessel head removed, the combined water inventory in the fuel pool, the reactor cavity, and the separator/dryer pool, between the fuel pool low level alarm and the reactor vessel flange, is about 65,800 cubic feet (492,000 gallons). This will provide adequate low-pressure cooling in lieu of CSS and RHR (LPCI and containment cooling mode) as currently required in Specifications 3.5.A.4 and 3.5.B.9. The additional requirements for providing standby coolant supply available will ensure a redundant supply of coolant supply. Control rod drive maintenance may continue during this period provided no more than one drive is removed at a time unless blind flanges are installed during the period of time CRDs are not in place.

Should the capability for providing flow through the cross-connect lines be lost, a 10-day repair time is allowed before shutdown is required. This repair time is justified based on the very small probability for ever needing RHR pumps and heat exchangers to supply an adjacent unit.

REFERENCES

1. Residual Heat Removal System (BFNP FSAR subsection 4.8)

2. Core Standby Cooling Systems (BFNP FSAR Section 6)

3.5.C. RHR Service Water System and Emergency Equipment Cooling Water System (EECWS)

The EECW has two completely redundant and independent headers (north and south) in a loop arrangement inside and outside the Reactor Building. Four RHRSW pumps, two per header, (A3, B3, C3 and D3) are dedicated to automatically supplying the EECW system needs. Four additional pumps (A1, B1, C1 and D1) can serve as RHRSW system pumps or be manually valved into the EECW system headers and serve as backup for the RHRSW pumps dedicated to supplying the EECW system. Those components requiring EECW, except the control air compressors which

are not safety related, are able to be fed from both headers thus assuring continuity of operation if either header becomes inoperable. The control air compressors only use the EECW north header as an emergency backup supply.

There are four RHR heat exchanger headers (A, B, C, & D) with one RHR heat exchanger from each unit on each header. There are two RHRSW pumps on each header; one normally assigned to each header (A2, B2, C2, or D2) and one on alternate assignment (A1, B1, C1, or D1). One RHR heat exchanger header can adequately deliver the flow supplied by both RHRSW pumps to any two of the three RHR heat exchangers on the header. One RHRSW pump can supply the full flow requirement of one RHR heat exchanger. Two RHR heat exchangers can more than adequately handle the cooling requirements of one unit in any abnormal or postaccident situation.

The RHR Service Water System was designed as a shared system for three units. The specification, as written, is conservative when consideration is given to particular pumps being out of service and to possible valving arrangements. If unusual operating conditions arise such that more pumps are out of service than allowed by this specification, a special case request may be made to the NRC to allow continued operation if the actual system cooling requirements can be assured.

Since the standby coolant supply capability provides added long term redundancy to the other emergency and containment cooling systems, a 5-hour time to establish flow path availability is allowed. This time limit does not reduce the other requirements associated with RHRSW/EECW system pump operability.

Should one of the two RHRSW pumps normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection become inoperable, an equal capability for long-term fluid makeup to the unit reactor and for cooling of the unit containment remains OPERABLE. Because of the availability of an equal makeup and cooling capability, a 30-day repair period is justified. Should the capability to provide standby coolant supply be lost, a 10-day repair time is justified based on the low probability for ever needing the standby coolant supply.

With only one unit fueled, four RHRSW pumps are required to be OPERABLE for indefinite operation to meet the requirements of Specification 3.5.B.1 (RHR system). If only three RHRSW pumps are OPERABLE, a 30-day LCO exists because of the requirement of Specification 3.5.B.5 (RHR system).

3.5.D Equipment Area Coolers

There is an equipment area cooler for each RHR pump and an equipment area cooler for each set (two pumps, either the A and C or B and D pumps) of core spray pumps. The equipment area coolers take suction

3.5 BASES (Cont'd)

near the cooling air discharge of the motor of the pump(s) served and discharge air near the cooling air suction of the motor of the pump(s) served. This ensures that cool air is supplied for cooling the pump motors.

The equipment area coolers also remove the pump, and equipment waste heat from the basement rooms housing the engineered safeguard equipment. The various conditions under which the operation of the equipment air coolers is required have been identified by evaluating the normal and abnormal operating transients and accidents over the full range of planned operations. The surveillance and testing of the equipment area coolers in each of their various modes is accomplished during the testing of the equipment served by these coolers. This testing is adequate to assure the OPERABILITY of the equipment area coolers.

REFERENCES

1. Residual Heat Removal System (BFN FSAR Section 4.8)

2. Core Standby Cooling System (BFN FSAR Section 6)

3.5.E. High Pressure Coolant Injection System (HPCIS)

The HPCIS is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCIS continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or Core Spray system operation maintains core cooling. The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to pump 5000 gpm at reactor pressures between 1120 and 150 psig. The HPCIS is not required to be OPERABLE below 150 psig since this is well within the range of the low pressure cooling systems and below the pressure of any events for which HPCI is required to provide core cooling.

The minimum required NPSH for HPCI is 21 feet. There is adequate elevation head between the suppression pool and the HPCI pump, such that the required NPSH is available with a suppression pool temperature up to 140°F with no containment back pressure.

The HPCIS is not designed to operate at full capacity until reactor pressure exceeds 150 psig and the steam supply to the HPCI turbine is automatically isolated before reactor pressure decreases below 100 psig. The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the HPCI turbine for OPERABILITY testing yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. The ADS provides additional backup to reduce pressure to the range where the CSS and RHRS will inject into the vessel if necessary. Considering the low reactor pressure, the redundancy and availability of CSS, RHRS, and ADS during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate HPCI OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate HPCI OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the HPCIS inoperable, a seven-day period to return the system to service is justified based on the availability of the ADS, CSS, RHRS (LPCI) and the RCICS. The availability of these redundant and diversified systems provides adequate assurance of core cooling while HPCIS is out of service.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the HPCIS will be OPERABLE when required.

3.5.F Reactor Core Isolation Cooling System (RCICS)

The RCICS functions to provide makeup water to the reactor vessel during shutdown and isolation from the main heat sink to supplement or replace the normal makeup sources. The RCICS provides its design flow between 150 psig and 1120 psig reactor pressure. Below 150 psig, RCICS is not required to be OPERABLE since this pressure is substantially below that for any events in which RCICS is needed to maintain sufficient coolant to the reactor vessel. RCICS will continue to operate below 150 psig at reduced flow until it automatically isolates at greater than or equal to 50 psig reactor steam pressure. 150 psig is also below the shutoff head of the CSS and RHRS, thus, considerable overlap exists with the cooling systems that provide core cooling at low reactor pressure. The minimum required NPSH for RCIC is 20 feet. There is adequate elevation head between the suppression pool and the RCIC pump, such that the required NPSH is available with a suppression pool temperature up to 140°F with no containment back pressure.

The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the RCIC turbine for OPERABILITY testing, yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. Considering the low reactor pressure and the availability of the low pressure coolant systems during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate RCIC OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate RCIC OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

3.5 BASES (Cont'd)

With the RCICS inoperable, a seven-day period to return the system to service is justified based on the availability of the HPCIS to cool the core and upon consideration that the average risk associated with failure of the RCICS to cool the core when required is not increased.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the RCICS will be OPERABLE when required.

3.5.G Automatic Depressurization System (ADS)

The ADS consists of six of the thirteen relief valves. It is designed to provide depressurization of the reactor coolant system during a small break loss of coolant accident (LOCA) if HPCI fails or is unable to maintain the required water level in the reactor vessel. ADS operation reduces the reactor vessel pressure to within the operating pressure range of the low pressure emergency core cooling systems (core spray and LPCI) so that they can operate to protect the fuel barrier. Specification 3.5.G applies only to the automatic feature of the pressure relief system.

Specification 3.6.D specifies the requirements for the pressure relief function of the valves. It is possible for any number of the valves assigned to the ADS to be incapable of performing their ADS functions because of instrumentation failures, yet be fully capable of performing their pressure relief function.

The emergency core cooling system LOCA analyses for small line breaks assumed that four of the six ADS valves were operable. By requiring six valves to be OPERABLE, additional conservatism is provided to account for the possibility of a single failure in the ADS system.

Reactor operation with one of the six ADS valves inoperable is allowed to continue for fourteen days provided the HPCI, core spray, and LPCI systems are OPERABLE. Operation with more than one ADS valve inoperable is not acceptable.

With one ADS valve known to be incapable of automatic operation, five valves remain OPERABLE to perform the ADS function. This condition is within the analyses for a small break LOCA and the peak clad temperature is well below the 10 CFR 50.46 limit. Analysis has shown that four valves are capable of depressurizing the reactor rapidly enough to maintain peak clad temperature within acceptable limits.

3.5.H. Maintenance of Filled Discharge Pipe

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

The core spray and RHR system discharge piping high point vent is visually checked for water flow once a month and prior to testing to ensure that the lines are filled. The visual checking will avoid starting the core spray or RHR system with a discharge line not filled. In addition to the visual observation and to ensure a filled discharge line other than prior to testing, a pressure suppression chamber head tank is located approximately 20 feet above the discharge line high point to supply makeup water for these systems. The condensate head tank located approximately 100 feet above the discharge high point serves as a backup charging system when the pressure suppression chamber head tank is not in service. System discharge pressure indicators are used to determine the water level above the discharge line high point. The indicators will reflect approximately 30 psig for a water level at the high point and 45 psig for a water level in the pressure suppression chamber head tank and are monitored daily to ensure that the discharge lines are filled.

When in their normal standby condition, the suction for the HPCI and RCIC pumps are aligned to the condensate storage tank, which is physically at a higher elevation than the HPCIS and RCICS piping. This assures that the HPCI and RCIC discharge piping remains filled. Further assurance is provided by observing water flow from these systems' high points monthly.

3.5.I. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit.

3.5.J. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated.

3.5 BASES (Cont'd)

The LHGR shall be checked daily during reactor operation at $\geq$ 25 percent power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 percent of rated thermal power, the largest total peaking would have to be greater than approximately 9.7 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

3.5.L. APRM Setpoints

Operation is constrained to the LHGR limit of Specification 3.5.J. This limit is reached when core maximum fraction of limiting power density (CMFLPD) equals 1.0. For the case where CMFLPD exceeds the fraction of rated thermal power, operation is permitted only at less than 100-percent rated power and only with APRM scram settings as required by Specification 3.5.L.1. The scram trip setting and rod block trip setting are adjusted to ensure that no combination of CMFLPD and FRP will increase the LHGR transient peak beyond that allowed by the 1-percent plastic strain limit. A 6-hour time period to achieve this condition is justified since the additional margin gained by the setdown adjustment is above and beyond that ensured by the safety analysis.

3.5.M. Core Thermal-Hydraulic Stability

The minimum margin to the onset of thermal-hydraulic instability occurs in Region I of Figure 3.5.M-1. A manually initiated scram upon entry into this region is sufficient to preclude core oscillations which could challenge the MCPR safety limit.

Because the probability of thermal-hydraulic oscillations is lower and the margin to the MCPR safety limit is greater in Region II than in Region I of figure 3.5.M-1, an immediate scram upon entry into the region is not necessary. However, in order to minimize the probability of core instability following entry into Region II, the operator will take immediate action to exit the region. Although formal surveillances are not performed while exiting Region II (delaying exit for surveillances is undesirable), an immediate manual scram will be initiated if evidence of thermal-hydraulic instability is observed.

Clear indications of thermal-hydraulic instability are APRM oscillations which exceed 10 percent peak-to-peak or LPRM oscillations which exceed 30 percent peak-to-peak (approximately equivalent to APRM oscillations of 10 percent during regional oscillations). Periodic LPRM upscale or downscale alarms may also be indicators of thermal hydraulic instability and will be immediately investigated.

Periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

Normal operation of the reactor is restricted to thermal power and core flow conditions (i.e., outside Regions I and II) where thermal-hydraulic instabilities are very unlikely to occur.

3.5.N. References

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 2, NEDO - 24088-1 and Addenda.

2. "BWR Transient Analysis Model Utilizing the RETRAN Program," TVA-TR81-01-A.

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

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by OPERABILITY of the remaining redundant equipment.

Whenever a CSCS system or loop is made inoperable, the other CSCS systems or loops that are required to be OPERABLE shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Average Planar LHGR, LHGR, and MCPR

The APLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 199
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 June 2, 1995, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 199 , 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: November 2, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 199

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
3.2/4.2-30	3.2/4.2-30
3.2/4.2-31	3.2/4.2-31*
3.2/4.2-53	3.2/4.2-53
3.2/4.2-54	3.2/4.2-54*
3.5/4.5-9	3.5/4.5-9*
3.5/4.5-10	3.5/4.5-10
3.5/4.5-27	3.5/4.5-27*
3.5/4.5-28	3.5/4.5-28
3.5/4.5-29	3.5/4.5-29
3.5/4.5-30	3.5/4.5-30
3.5/4.5-31	3.5/4.5-31**
3.5/4.5-32	3.5/4.5-32**
3.5/4.5-33	3.5/4.5-33**
3.5/4.5-34	3.5/4.5-34**
3.5/4.5-35	3.5/4.5-35**
3.5/4.5-36	3.5/4.5-36**
3.5/4.5-37	3.5/4.5-37**
---	3.5/4.5-38

TABLE 3.2.F

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	LI-3-58A LI-3-58B	Reactor Water Level	Indicator - 155" to +60"	(1) (2) (3)
2	PI-3-74A PI-3-74B	Reactor Pressure	Indicator 0-1200 psig	(1) (2) (3)
2	XR-64-50 PI-64-67B	Drywell Pressure	Recorder -15 to +65 psig Indicator -15 to +65 psig	(1) (2) (3)
2	XR-64-50 TI-64-52AB	Drywell Temperature	Recorder, Indicator 0-400°F	(1) (2) (3)
1	XR-64-52	Suppression Chamber Air Temperature	Recorder 0-400°F	(1) (2) (3)
1	N/A	Control Rod Position	6V Indicating Lights)	
1	N/A	Neutron Monitoring	SRM, IRM, LPRM 0 to 100% power)	(1) (2) (3) (4)
1	PS-64-67B	Drywell Pressure	Alarm at 35 psig)	
1	TS-64-52A & PIS-64-58A & IS-64-67A	Drywell Temperature and Pressure and Timer	Alarm if temp. > 281°F and pressure >2.5 psig after 30 minute delay)	(1) (2) (3) (4)
1	LI-84-2A	CAD Tank "A" Level	Indicator 0 to 100%	(1)
1	LI-84-13A	CAD Tank "B" Level	Indicator 0 to 100%	(1)

TABLE 3.2.F (cont'd)

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	H_2M - 76 - 94 H_2M - 76 - 104	Drywell and Torus Hydrogen Concentration	0.1 - 20%	(1)
2	PdI-64-137 PdI-64-138	Drywell to Suppression Chamber Differential Pressure	Indicator 0 to 2 psid	(1) (2) (3)
1/Valve		Relief Valve Tailpipe Thermocouple Temperature or Acoustic Monitor on Relief Valve Tailpipe		(5)
2	LI-64-159A XR-64-159	Suppression Chamber Water Level-Wide Range	Indicator, Recorder 0-240"	(1) (2) (3)
2	PI-64-160A XR-64-159	Drywell Pressure Wide Range	Indicator, Recorder) 0-300 psig)	(1) (2) (3)
2	TI-64-161 TR-64-161 TI-64-162 TR-64-162	Suppression Pool Bulk Temperature	Indicator, Recorder)) 30° - 230° F))	(1) (2) (3) (4) (6)
1	RR-90-272 RR-90-273 RM-90-272A RM-90-273A	High Range Primary Containment Radiation Monitors and Recorders	Monitor, Recorder $1 - 10^7$ R/Hr	(7) (8)
1	RM-90-306 RR-90-360	Wide Range Gaseous Effluent Radiation Monitor and Recorder	Monitor, Recorder (Noble Gas $10^{-7} - 10^{+5}$ μCi/cc)	(7)(8)(9)

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

Instrument Channel	Calibration Frequency	Instrument Check
1) Reactor Water Level (LI-3-58A & B)	Once/18 months	Each Shift
2) Reactor Pressure (PI-3-74A & B)	Once/6 months	Each Shift
3) Drywell Pressure	Once/6 months	Each Shift
4) Drywell Temperature	Once/6 months	Each Shift
5) Suppression Chamber Air Temperature	Once/6 months	Each Shift
8) Control Rod Position	N/A	Each Shift
9) Neutron Monitoring	(2)	Each Shift
10) Drywell Pressure (PS-64-67B)	Once/6 months	N/A
11) Drywell Pressure (PIS-64-58A)	Once/18 months	N/A
12) Drywell Temperature (TS-64-52A)	Once/6 months	N/A
13) Timer (IS-64-67A)	Once/6 months	N/A
14) CAD Tank Level	Once/6 months	Once/day
15) Containment Atmosphere Monitors	Once/6 months	Once/day

TABLE 4.2.F (Cont'd)
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
16) Drywell to Suppression Chamber Differential Pressure	Once/6 months	Each Shift
17) Relief Valve Tailpipe Thermocouple Temperature	N/A	Once/month (24)
18) Acoustic Monitor on Relief Valve Tailpipe	Once/cycle (25)	Once/month (26)
19) Suppression Chamber Water Level-Wide Range (LI-64-159A) (XR-64-159)	Once/cycle	Once/month
20) Drywell Pressure - Wide Range (PI-64-160A) (XR-64-159)	Once/cycle	Once/shift
21) Suppression Pool Bulk Temperature (TI-64-161) (TR-64-161) (TI-64-162) (TR-64-162)	Once/cycle	Once shift
22) High Range Primary Containment Radiation Monitors and Recorders (RR-90-272, RR-90-273, RM-90-272A, RM-90-273A)	Once/18 months (32)	Once/month
23) Wide Range Gaseous Effluent Radiation Monitor and Recorder (RM-90-306 and RR-90-360)	Once/18 months	Once/shift

LIMITING CONDITIONS FOR OPERATION

3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)

1. PRIOR TO STARTUP from a COLD SHUTDOWN CONDITION, the RHRSW pumps, including pump B1 or B2, shall be OPERABLE and assigned to service as indicated in Table 3.5-1.

SURVEILLANCE REQUIREMENTS

4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)

1. a. Each of the RHRSW pumps normally assigned to automatic service on the EECW headers will be tested automatically each time the diesel generators are tested. Each of the RHRSW pumps and all associated essential control valves for the EECW headers and RHR heat exchanger headers shall be demonstrated to be OPERABLE in accordance with Specification 1.0.MM.

   b. Annually each RHRSW pump shall be flow-rate tested. To be considered OPERABLE, each pump shall pump at least 4500 gpm through its normally assigned flow path.

   c. Monthly verify that each valve (manual, power-operated, or automatic) in the flowpath servicing safety-related equipment in the affected unit that is not locked, sealed, or otherwise secured in position, is in its correct position.

LIMITING CONDITIONS FOR OPERATION

3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)

2. During REACTOR POWER OPERATION, RHRSW pumps must be OPERABLE and assigned to service as indicated in Table 3.5-1 for the specified time limits.

3. During Unit 3 REACTOR POWER OPERATION, both RHRSW pumps B1 and B2 and associated valves normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection must be OPERABLE except as specified in 3.5.C.4 and 3.5.C.5 below. (Note: Because standby coolant supply capability is not a short-term requirement, a component is not considered inoperable if standby coolant supply capability can be restored to service within 5 hours.)

SURVEILLANCE REQUIREMENTS

4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)

2. No additional surveillance is required.

3. Routine surveillance for these pumps is specified in 4.5.C.1.

3.5 BASES

3.5.A. Core Spray System (CSS) and 3.5.B Residual Heat Removal System (RHRS)

Analyses presented in the FSAR* and analyses presented in conformance with 10 CFR 50, Appendix K, demonstrated that the core spray system provides adequate cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel clad temperature to below 2,200°F which assures that core geometry remains intact and to limit the core average clad metal-water reaction to less than 1 percent. Core spray distribution has been shown in tests of systems similar in design to BFNP to exceed the minimum requirements. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel.

The RHRS (LPCI mode) is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is completely independent of the core spray system; however, it does function in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI mode of the RHRS and the core spray system provide adequate cooling for break areas of approximately 0.2 square feet up to and including the double-ended recirculation line break without assistance from the high-pressure emergency core cooling subsystems.

The intent of the CSS and RHRS specifications is to not allow startup from the cold condition without all associated equipment being OPERABLE. However, during operation, certain components may be out of service for the specified allowable repair times. The allowable repair times have been selected using engineering judgment based on experiences and supported by availability analysis.

Should one core spray loop become inoperable, the remaining core spray loop, the RHR System, and the diesel generators are required to be OPERABLE should the need for core cooling arise. These provide extensive margin over the OPERABLE equipment needed for adequate core cooling. With due regard for this margin, the allowable repair time of seven days was chosen.

Should one RHR pump (LPCI mode) become inoperable, three RHR pumps (LPCI mode) and the core spray system are available. Since adequate core cooling is assured with this complement of ECCS, a seven day repair period is justified.

Should two RHR pumps (LPCI mode) become inoperable, there remains no reserve (redundant) capacity within the RHRS (LPCI mode). Therefore, the affected unit shall be placed in cold shutdown within 24 hours.

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*A detailed functional analysis is given in Section 6 of the BFNP FSAR.

Should one RHR pump (containment cooling mode) become inoperable, a complement of three full capacity containment heat removal systems is still available. Any two of the remaining pumps/heat exchanger combinations would provide more than adequate containment cooling for any abnormal or postaccident situation. Because of the availability of equipment in excess of normal redundancy requirements, a 30-day repair period is justified.

Should two RHR pumps (containment cooling mode) become inoperable, a full heat removal system is still available. The remaining pump/heat exchanger combinations would provide adequate containment cooling for any abnormal postaccident situation. Because of the availability of a full complement of heat removal equipment, a 7-day repair period is justified.

Observation of the stated requirements for the containment cooling mode assures that the suppression pool and the drywell will be sufficiently cooled, following a loss-of-coolant accident, to prevent primary containment overpressurization. The containment cooling function of the RHRS is permitted only after the core has reflooded to the two-thirds core height level. This prevents inadvertently diverting water needed for core flooding to the less urgent task of containment cooling. The two-thirds core height level interlock may be manually bypassed by a keylock switch.

Since the RHRS is filled with low quality water during power operation, it is planned that the system be filled with demineralized (condensate) water before using the shutdown cooling function of the RHR System. Since it is desirable to have the RHRS in service if a "pipe-break" type of accident should occur, it is permitted to be out of operation for only a restricted amount of time and when the system pressure is low. At least one-half of the containment cooling function must remain OPERABLE during this time period. Requiring two OPERABLE CSS pumps during cooldown allows for flushing the RHRS even if the shutdown were caused by inability to meet the CSS specifications (3.5.A) on a number of OPERABLE pumps.

When the reactor vessel pressure is atmospheric, the limiting conditions for operation are less restrictive. At atmospheric pressure, the minimum requirement is for one supply of makeup water to the core. Requiring two OPERABLE RHR pumps and one CSS pump provides redundancy to ensure makeup water availability.

Verification that the LPCI subsystem cross-tie valve is closed and power to its operator is disconnected ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other LPCI subsystem.

Since the RHR system cross-connect capability provides added long term redundancy to the other emergency and containment cooling systems, a

5-hour time to establish flow path availability is allowed. This time limit does not reduce the other requirements associated with RHR system pump operability.

Should one RHR pump or associated heat exchanger located on the unit cross-connection in the adjacent unit become inoperable, an equal capability for long-term fluid makeup to the reactor and for cooling of the containment remains OPERABLE. Because of the availability of an equal makeup and cooling capability, a 30-day repair period is justified.

The suppression chamber can be drained when the reactor vessel pressure is atmospheric, irradiated fuel is in the reactor vessel, and work is not in progress which has the potential to drain the vessel. By requiring the fuel pool gate to be open with the vessel head removed, the combined water inventory in the fuel pool, the reactor cavity, and the separator/dryer pool, between the fuel pool low level alarm and the reactor vessel flange, is about 65,800 cubic feet (492,000 gallons). This will provide adequate low-pressure cooling in lieu of CSS and RHR (LPCI and containment cooling mode) as currently required in Specifications 3.5.A.4 and 3.5.B.9. The additional requirements for providing standby coolant supply available will ensure a redundant supply of coolant supply. Control rod drive maintenance may continue during this period provided no more than one drive is removed at a time unless blind flanges are installed during the period of time CRDs are not in place.

Should the capability for providing flow through the cross-connect lines be lost, a 10-day repair time is allowed before shutdown is required. This repair time is justified based on the very small probability for ever needing RHR pumps and heat exchangers to supply an adjacent unit.

REFERENCES

1. Residual Heat Removal System (BFNP FSAR subsection 4.8)

2. Core Standby Cooling Systems (BFNP FSAR Section 6)

3.5.C. RHR Service Water System and Emergency Equipment Cooling Water System (EECWS)

The EECW has two completely redundant and independent headers (north and south) in a loop arrangement inside and outside the Reactor Building. Four RHRSW pumps, two per header, (A3, B3, C3 and D3) are dedicated to automatically supplying the EECW system needs. Four additional pumps (A1, B1, C1 and D1) can serve as RHRSW system pumps or be manually valved into the EECW system headers and serve as backup for the RHRSW pumps dedicated to supplying the EECW system. Those components requiring EECW, except the control air compressors which

are not safety related, are able to be fed from both headers thus assuring continuity of operation if either header becomes inoperable. The control air compressors only use the EECW north header as an emergency backup supply.

There are four RHR heat exchanger headers (A, B, C, & D) with one RHR heat exchanger from each unit on each header. There are two RHRSW pumps on each header; one normally assigned to each header (A2, B2, C2, or D2) and one on alternate assignment (A1, B1, C1, or D1). One RHR heat exchanger header can adequately deliver the flow supplied by both RHRSW pumps to any two of the three RHR heat exchangers on the header. One RHRSW pump can supply the full flow requirement of one RHR heat exchanger. Two RHR heat exchangers can more than adequately handle the cooling requirements of one unit in any abnormal or postaccident situation.

The RHR Service Water System was designed as a shared system for three units. The specification, as written, is conservative when consideration is given to particular pumps being out of service and to possible valving arrangements. If unusual operating conditions arise such that more pumps are out of service than allowed by this specification, a special case request may be made to the NRC to allow continued operation if the actual system cooling requirements can be assured.

Since the standby coolant supply capability provides added long term redundancy to the other emergency and containment cooling systems, a 5-hour time to establish flow path availability is allowed. This time limit does not reduce the other requirements associated with RHRSW/EECW system pump operability.

Should one of the two RHRSW pumps normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection become inoperable, an equal capability for long-term fluid makeup to the unit reactor and for cooling of the unit containment remains OPERABLE. Because of the availability of an equal makeup and cooling capability, a 30-day repair period is justified. Should the capability to provide standby coolant supply be lost, a 10-day repair time is justified based on the low probability for ever needing the standby coolant supply.

With only one unit fueled, four RHRSW pumps are required to be OPERABLE for indefinite operation to meet the requirements of Specification 3.5.B.1 (RHR system). If only three RHRSW pumps are OPERABLE, a 30-day LCO exists because of the requirement of Specification 3.5.B.5 (RHR system).

3.5.D Equipment Area Coolers

There is an equipment area cooler for each RHR pump and an equipment area cooler for each set (two pumps, either the A and C or B and D pumps) of core spray pumps. The equipment area coolers take suction

3.5 BASES (Cont'd)

near the cooling air discharge of the motor of the pump(s) served and discharge air near the cooling air suction of the motor of the pump(s) served. This ensures that cool air is supplied for cooling the pump motors.

The equipment area coolers also remove the pump, and equipment waste heat from the basement rooms housing the engineered safeguard equipment. The various conditions under which the operation of the equipment air coolers is required have been identified by evaluating the normal and abnormal operating transients and accidents over the full range of planned operations. The surveillance and testing of the equipment area coolers in each of their various modes is accomplished during the testing of the equipment served by these coolers. This testing is adequate to assure the OPERABILITY of the equipment area coolers.

REFERENCES

1. Residual Heat Removal System (BFN FSAR Section 4.8)

2. Core Standby Cooling System (BFN FSAR Section 6)

3.5.E. High Pressure Coolant Injection System (HPCIS)

The HPCIS is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCIS continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or Core Spray system operation maintains core cooling. The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to pump 5000 gpm at reactor pressures between 1120 and 150 psig. The HPCIS is not required to be OPERABLE below 150 psig since this is well within the range of the low pressure cooling systems and below the pressure of any events for which HPCI is required to provide core cooling.

The minimum required NPSH for HPCI is 21 feet. There is adequate elevation head between the suppression pool and the HPCI pump, such that the required NPSH is available with a suppression pool temperature up to 140°F with no containment back pressure.

The HPCIS is not designed to operate at full capacity until reactor pressure exceeds 150 psig and the steam supply to the HPCI turbine is automatically isolated before reactor pressure decreases below 100 psig. The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the HPCI turbine for OPERABILITY testing, yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. The ADS provides additional backup to reduce pressure to the range where the CSS and RHRS will inject into the vessel if necessary. Considering the low reactor pressure, the redundancy and availability of CSS, RHRS, and ADS during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate HPCI OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate HPCI OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the HPCIS inoperable, a seven-day period to return the system to service is justified based on the availability of the ADS, CSS, RHRS (LPCI) and the RCICS. The availability of these redundant and diversified systems provides adequate assurance of core cooling while HPCIS is out of service.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the HPCIS will be OPERABLE when required.

3.5.F Reactor Core Isolation Cooling System (RCICS)

The RCICS functions to provide makeup water to the reactor vessel during shutdown and isolation from the main heat sink to supplement or replace the normal makeup sources. The RCICS provides its design flow between 150 psig and 1120 psig reactor pressure. Below 150 psig, RCICS is not required to be OPERABLE since this pressure is substantially below that for any events in which RCICS is needed to maintain sufficient coolant to the reactor vessel. RCICS will continue to operate below 150 psig at reduced flow until it automatically isolates at greater than or equal to 50 psig reactor steam pressure. 150 psig is also below the shutoff head of the CSS and RHRS, thus, considerable overlap exists with the cooling systems that provide core cooling at low reactor pressure. The minimum required NPSH for RCIC is 20 feet. There is adequate elevation head between the suppression pool and the RCIC pump, such that the required NPSH is available with a suppression pool temperature up to 140°F with no containment back pressure.

The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the RCIC turbine for OPERABILITY testing, yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. Considering the low reactor pressure and the availability of the low pressure coolant systems during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate RCIC OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate RCIC OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the RCICS inoperable, a seven-day period to return the system to service is justified based on the availability of the HPCIS to cool the core and upon consideration that the average risk associated with failure of the RCICS to cool the core when required is not increased.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the RCICS will be OPERABLE when required.

3.5.G Automatic Depressurization System (ADS)

The ADS consists of six of the thirteen relief valves. It is designed to provide depressurization of the reactor coolant system during a small break loss of coolant accident (LOCA) if HPCI fails or is unable to maintain the required water level in the reactor vessel. ADS operation reduces the reactor vessel pressure to within the operating pressure range of the low pressure emergency core cooling systems (core spray and LPCI) so that they can operate to protect the fuel barrier. Specification 3.5.G applies only to the automatic feature of the pressure relief system.

Specification 3.6.D specifies the requirements for the pressure relief function of the valves. It is possible for any number of the valves assigned to the ADS to be incapable of performing their ADS functions because of instrumentation failures, yet be fully capable of performing their pressure relief function.

The emergency core cooling system LOCA analyses for small line breaks assumed that four of the six ADS valves were OPERABLE. By requiring six valves to be OPERABLE, additional conservatism is provided to account for the possibility of a single failure in the ADS system.

Reactor operation with one of the six ADS valves inoperable is allowed to continue for fourteen days provided the HPCI, core spray, and LPCI systems are OPERABLE. Operation with more than one ADS valve inoperable is not acceptable.

With one ADS valve known to be incapable of automatic operation, five valves remain OPERABLE to perform the ADS function. This condition is within the analyses for a small break LOCA and the peak clad temperature is well below the 10 CFR 50.46 limit. Analysis has shown that four valves are capable of depressurizing the reactor rapidly enough to maintain peak clad temperature within acceptable limits.

H. Maintenance of Filled Discharge Pipe

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

The core spray and RHR system discharge piping high point vent is visually checked for water flow once a month and prior to testing to ensure that the lines are filled. The visual checking will avoid starting the core spray or RHR system with a discharge line not filled. In addition to the visual observation and to ensure a filled discharge line other than prior to testing, a pressure suppression chamber head tank is located approximately 20 feet above the discharge line high point to supply makeup water for these systems. The condensate head tank located approximately 100 feet above the discharge high point serves as a backup charging system when the pressure suppression chamber head tank is not in service. System discharge pressure indicators are used to determine the water level above the discharge line high point. The indicators will reflect approximately 30 psig for a water level at the high point and 45 psig for a water level in the pressure suppression chamber head tank and are monitored daily to ensure that the discharge lines are filled.

When in their normal standby condition, the suction for the HPCI and RCIC pumps are aligned to the condensate storage tank, which is physically at a higher elevation than the HPCIS and RCICS piping. This assures that the HPCI and RCIC discharge piping remains filled. Further assurance is provided by observing water flow from these systems' high points monthly.

3.5.I. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit.

3.5.J. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated.

3.5 BASES (Cont'd)

The LHGR shall be checked daily during reactor operation at ≥ 25 percent power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 percent of rated thermal power, the largest total peaking would have to be greater than approximately 9.7 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

3.5.L. APRM Setpoints

Operation is constrained to the LHGR limit of Specification 3.5.J. This limit is reached when core maximum fraction of limiting power density (CMFLPD) equals 1.0. For the case where CMFLPD exceeds the fraction of rated thermal power, operation is permitted only at less than 100-percent rated power and only with APRM scram settings as required by Specification 3.5.L.1. The scram trip setting and rod block trip setting are adjusted to ensure that no combination of CMFLPD and FRP will increase the LHGR transient peak beyond that allowed by the one-percent plastic strain limit. A six-hour time period to achieve this condition is justified since the additional margin gained by the setdown adjustment is above and beyond that ensured by the safety analysis.

3.5.M. Core Thermal-Hydraulic Stability

The minimum margin to the onset of thermal-hydraulic instability occurs in Region I of Figure 3.5.M-1. A manually initiated scram upon entry into this region is sufficient to preclude core oscillations which could challenge the MCPR safety limit.

Because the probability of thermal-hydraulic oscillations is lower and the margin to the MCPR safety limit is greater in Region II than in Region I of Figure 3.5.M-1, an immediate scram upon entry into the region is not necessary. However, in order to minimize the probability of core instability following entry into Region II, the operator will take immediate action to exit the region. Although formal surveillances are not performed while exiting Region II (delaying exit for surveillances is undesirable), an immediate manual scram will be initiated if evidence of thermal-hydraulic instability is observed.

Clear indications of thermal-hydraulic instability are APRM oscillations which exceed 10 percent peak-to-peak or LPRM oscillations which exceed 30 percent peak-to-peak (approximately equivalent to APRM oscillations of 10 percent during regional oscillations). Periodic LPRM upscale or downscale alarms may also be indicators of thermal hydraulic instability and will be immediately investigated.

Periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

Normal operation of the reactor is restricted to thermal power and core flow conditions (i.e., outside Regions I and II) where thermal-hydraulic instabilities are very unlikely to occur.

3.5.N. References

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3, NEDO-24194A and Addenda.

2. "BWR Transient Analysis Model Utilizing the RETRAN Program," TVA-TR81-01-A.

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

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by OPERABILITY of the remaining redundant equipment.

Whenever a CSCS system or loop is made inoperable, the other CSCS systems or loops that are required to be OPERABLE shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Average Planar LHGR, LHGR, and MCPR

The APLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 225 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 240 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 199 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

On June 2, 1995, the Tennessee Valley Authority (the licensee) submitted an application for amendment to technical specifications (TS) for the Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3. The application modifies the definition of operability for the standby coolant supply valves of the Residual Heat Removal Service Water (RHRSW) system in BFN Units 1, 2, and 3 by adding a note to TS 3.5.C.3. The application also administratively updates instrument numbers in unit 3 TS tables 3.2.F and 4.2.F. Modifications to BFN Unit 3 TS table 3.2.F and 4.2.F (except for PIS-64-58A) are administrative changes to reflect instrument updates. Surveillance requirements for the instrumentation have not changed. Instrument PIS-64-58A reflects a change omitted from an earlier TS change request. The operability requirements of the remainder of the service water system and the RHR cross-connect valves are not affected by these changes.

The licensee was requested to add information to the TS bases to further clarify the scope of components subject to the new operability definition. By letter dated October 2, 1995, the licensee submitted planned changes to TS bases, sections 3.5.A. and 3.5.B, "Core Spray System (CSS) and Residual heat removal System (RHRS)," 3.5.C, "RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)." These bases changes do not affect the staff's proposed finding of no significant hazards considerations.

This change request resulted from the licensee's 10 CFR 50 Appendix R safe shutdown evaluation. Based on this evaluation, the residual heat removal (RHR) system cross-connect valves between units 2 and 3 (valve nos. 2-FCV-74-101 and 3-FCV-74-100) will have their power supply breakers open during reactor power operation, rendering the standby coolant supply unable to inject to the RHR system, because the RHR cross-connect valves are in the

Enclosure 4

standby coolant supply flowpath. TS 3.5.B.11 requires the cross-connect valves capable of being returned to service within 5 hours.

From a systems viewpoint, the review of the licensee request focused on whether the revised operability definition places the facility in an unanalyzed condition. The licensee safety analysis does not take credit for operability of the standby coolant supply for any accident or transient. Both the standby coolant supply valves and the cross-connect valves will continue to remain closed during normal operation.

2.0 DESCRIPTION OF PROPOSED TECHNICAL SPECIFICATIONS CHANGES

By letter dated June 2, 1995, the licensee has proposed the following TS changes:

Revise BFN Units 1, 2 and 3 TS 3.5.C.3, "Standby Coolant Supply", page 3/4.5-10, by inserting the following note:

> Note: Because standby coolant supply capability is not a short-term requirement, a component is not considered inoperable if standby coolant supply capability can be restored to service within 5 hours.

Revise unit 3 TS Table 3.2.F, p. 3/4.2-30, "Surveillance Instrumentation" by changing instrument numbers as shown below:

For the first Drywell Pressure:
Existing TS Instrument #: PI-64-67
Proposed TS Instrument #: PI-64-67B

For Drywell Temperature:
Existing TS Instrument #: TI-64-52 and XR-64-50
Proposed TS Instrument #: XR-64-50 and TI-64-52AB

For the second Drywell Pressure:
Existing TS Instrument #: PS-64-67
Proposed TS Instrument #: PS-64-67B

For the Drywell Temperature and Pressure and Timer:
Existing TS Instrument #: XR-64-50, PS-64-58B and IS-64-67
Proposed TS Instrument #: TS-64-52A, PIS-64-58A and IS-64-67A

Revise unit 3 TS Table 4.2.F, p. 3/4.2-53, "Minimum Test and Calibration Frequency for Surveillance Instrumentation" by changing instrument numbers as shown below:

Instrument Channel-10, Drywell Pressure:
Existing TS Instrument #: PS-64-67
Proposed TS Instrument #: PS-64-67B

Instrument Channel-12, Drywell Temperature:
Existing TS Instrument #: TR-64-52
Proposed TS Instrument #: TR-64-52A

Instrument Channel-13, Timer:
Existing TS Instrument #: IS-64-67
Proposed TS Instrument #: IS-64-67A

By letter dated October 2, 1995, the licensee has also proposed the following TS bases changes:

Revise units 1, 2 and 3 TS bases, sections 3.5.A and 3.5.B, by adding the following:

> Since the RHR system cross-connect capability provides added long term redundancy to the other emergency and containment cooling systems, a 5-hour time to establish flow path availability is allowed. This time limit does not reduce the other requirements associated with RHR system pump operability.

Revise units 1, 2 and 3 TS bases, section 3.5.C, by adding the following:

> Since the standby coolant supply capability provides added long term redundancy to the other emergency and containment cooling systems, a 5-hour time to establish flow path availability is allowed. This time limit does not reduce the other requirements associated with RHRSW/EECW system pump operability.

Revise units 1, 2 and 3 TS bases, sections 3.5.A, 3.5.B and 3.5.C by moving the following from section 3.5.C to sections 3.5.A/3.5.B:

> Verification that the LPCI subsystem cross-tie valve is closed and power to its operator is disconnected ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other LPCI subsystem.

3.0 EVALUATION

Standby Coolant Supply System - Operability Definition

The licensee has re-evaluated the programs initiated for restart of Browns Ferry unit 2 to determine restart requirements for unit 3. The 10 CFR 50 Appendix R safe shutdown analysis was re-evaluated based upon two-unit operation to determine plant changes necessary to provide assurance that the minimum safe shutdown systems (SSDS) were available for a fire event. As a result of the evaluation, the normally closed RHR system cross-connect valves 2-FCV-74-101 and 3-FCV-74-100 will have their power supply breakers open during reactor power operation. Manual operator action will be required to restore the power to the affected cross-connect valves, allowing injection by the standby coolant supply.

The standby coolant supply provides a connection between the service water system and the residual heat removal system. The discharge of certain service water pumps are provided with an alternate flowpath through the standby coolant supply valves (1,2-FCV-23-57) to the RHR system via the RHR system cross-connect valves (1,2-FCV-74-101 and 2,3-FCV-74-100). Both valves are normally closed and manually operated. Currently, TS 3.5.B.11 requires the cross-connect valves to be capable of being returned to service within five hours. The RHR system cross-connect valves are designed to allow the RHR pumps and heat exchangers on one unit to provide core/primary containment cooling to an adjoining unit whose RHR and/or core spray pumps have failed. The licensee does not assume availability of the standby coolant supply or the RHR cross-connect valves in the Final Safety Analysis Report (FSAR) Chapter 14 safety analyses. Furthermore, the operability requirements of the remainder of the service water system and the RHR cross-connect valves are unaffected by the proposed modifications. Therefore, this change, taken together with the change to the TS bases described below, is acceptable.

Browns Ferry Unit 3 Instrumentation Updates

Technical Specifications Tables 3.2.F and 4.2.F have been revised to reflect new instrument numbers for the updated instrumentation. Surveillance requirements for the updated instrumentation are unchanged. These changes, except for PIS-64-58A, are administrative in nature and are consistent with the existing Unit 2 TS. The change for PIS-64-58A reflects an instrument number change that was inadvertently omitted from an earlier change request. On these bases, the instrumentation changes are acceptable.

Modifications to Technical Specification bases section 3.5.A, 3.5.B and 3.5.C

The licensee was requested by the staff to add information to the TS bases to further clarify the scope of components subject to the new operability definition. By letter dated October 2, 1995, the licensee submitted the requested changes. The additions to the TS bases state that operability requirements of the RHR system and the RHRSW/EECW system are not affected by the proposed changes. This is acceptable to the staff.

4.0 CONCLUSION

The licensee has requested a modification to the BFN Units 1, 2, and 3 technical specifications. The proposal modifies the definition of operability for the standby coolant supply valves of the RHRSW system. The standby coolant supply system will be considered operable if the capability to inject can be restored within 5 hours. The proposal also changes instrument numbers for BFN Unit 3 updated instrumentation.

Review of the licensee request focused on ensuring that the revised definition of operability for the standby coolant system does not place the facility in an unanalyzed condition. Neither the standby coolant supply connection nor the RHR cross-connect capability is required for mitigation of any event analyzed in the Browns Ferry Final Safety Analysis Report (FSAR) Chapter 14 safety analyses. Furthermore, the operability requirements of the remainder of the service water system and the RHR cross-connect valves are unaffected by the proposed modifications. The changes to instrument numbers are administrative changes for the upgraded drywell temperature and pressure instrumentation. The licensee, at the request of the staff, has also added information to the TS bases, stating that operability requirements of the RHR system and the RHRSW/EECW system are not affected by the changes. On these bases, the changes are acceptable to the staff.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes the surveillance requirements. The NRC 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 the amendments involve no significant hazards consideration, and there has been no public comment on such finding (60 FR 42610). 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 or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based upon the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: November 2, 1995

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