August 2, 1989

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

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

Dear Mr. Kingsley:

SUBJECT: TECHNICAL SPECIFICATION CHANGES TO DELETE REDUNDANT TESTING FOR EMERGENCY CORE COOLING AND REACTOR CORE ISOLATION COOLING SYSTEMS (TAC 71960, 71961, 71962) (TS 256)- BROWNS FERRY NUCLEAR PLANTS, UNITS 1, 2, AND 3

The Commission has issued the enclosed Amendment Nos. 169 , 169 , and 140 to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3, respectively. These amendments are in response to your application dated January 13, 1989.

These amendments delete certain surveillance testing requirements on redundant Emergency Core Cooling and Reactor Core Isolation Cooling Systems during a Limiting Condition of Operation.

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

Sincerely,

Original signed by

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 169 to License No. DPR-33
2. Amendment No. 169 to License No. DPR-52
3. Amendment No. 140 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

DF01
1/1

DISTRIBUTION:
Docket File
NRC PDR
Local PDR
TDaniels
ADSP Reading
DCrutchfield
BDLiaw
SBlack
RPierson
BWilson
WSLittle
MSimms
GGears
DMoran
GPA/CA
NMarkisohn
OGC
BGrimes
EJordan
ACRS(10)
GPA/PA
BFN Rdg. File
T. Meek (12)
WJones
EButcher
ARM/LFMB

DF01
1/1

*SEE PREVIOUS CONCURRENCE

OFC	:NRR:TVA/LA*	:NRR:TVA/PM*	:OGC*	:TVA:AD/P	:
NAME	:MSimms	:GGears:as	:	:SBlack SB	:
DATE	: 6/24/89	: 6/28/89	: 6/30/89	:7/5/89	: 8/2/89

OFFICIAL RECORD COPY

August 2, 1989

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

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

Dear Mr. Kingsley:

SUBJECT: TECHNICAL SPECIFICATION CHANGES TO DELETE REDUNDANT TESTING FOR EMERGENCY CORE COOLING AND REACTOR CORE ISOLATION COOLING SYSTEMS (TAC 71960, 71961, 71962) (TS 256)- BROWNS FERRY NUCLEAR PLANTS, UNITS 1, 2, AND 3

The Commission has issued the enclosed Amendment Nos. 169 , 169 , and 140 to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3, respectively. These amendments are in response to your application dated January 13, 1989.

These amendments delete certain surveillance testing requirements on redundant Emergency Core Cooling and Reactor Core Isolation Cooling Systems during a Limiting Condition of Operation.

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

Sincerely,

Original signed by

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 169 to License No. DPR-33
2. Amendment No. 169 to License No. DPR-52
3. Amendment No. 140 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:
Docket File
NRC PDR
Local PDR
TDaniels
ADSP Reading
DCrutchfield
BDLiaw
SBlack
RPierson
BWilson
WSLittle
MSimms
GGears
DMoran
GPA/CA
NMarkisohn
OGC
BGrimes
EJordan
ACRS(10)
GPA/PA
BFN Rdg. File
T. Meek (12)
WJones
EButcher
ARM/LFMB

*SEE PREVIOUS CONCURRENCE

OFC	:NRR:TVA/LA*	:NRR:TVA/PM*	:OGC*	:TVA:AD/P SB	:	:
NAME	:MSimms	:GGears:as	:	:SBlack	:	:
DATE	: 6/24/89	: 6/28/89	: 6/30/89	:7/5/89	:	:

; with Delay in receipt of clean TS pages TS package was delayed issuance SE 8/2/89

OFFICIAL RECORD COPY

cc:
General Counsel
Tennessee Valley Authority
400 West Summit Hill Drive
ET 11B 33H
Knoxville, Tennessee 37902

Mr. F. L. Moreadith
Vice President, Nuclear Engineering
Tennessee Valley Authority
400 West Summit Hill Drive
WT 12A 12A
Knoxville, Tennessee 37902

Dr. Mark O. Medford
Vice President and Nuclear
Technical Director
Tennessee Valley Authority
6N 38A Lookout Place
Chattanooga, Tennessee 37402-2801

Manager, Nuclear Licensing
and Regulatory Affairs
Tennessee Valley Authority
5N 157B Lookout Place
Chattanooga, Tennessee 37402-2801

Mr. O. J. Zeringue
Site Director
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P. O. Box 2000
Decatur, Alabama 35602

Mr. P. Carier
Site Licensing Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P. O. Box 2000
Decatur, Alabama 35602

Mr. G. Campbell
Plant Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P. O. Box 2000
Decatur, Alabama 35602

Chairman, Limestone County Commission
P. O. Box 188
Athens, Alabama 35611

Claude Earl Fox, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36130

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W.
Atlanta, Georgia 30323

Mr. Danny Carpenter
Senior Resident Inspector
Browns Ferry Nuclear Plant
U.S. Nuclear Regulatory Commission
Route 12, Box 637
Athens, Alabama 35611

Dr. Henry Myers, Science Advisor
Committee on Interior
and Insular Affairs
U.S. House of Representatives
Washington, D.C. 20515

Tennessee Valley Authority
Rockville Office
11921 Rockville Pike
Suite 402
Rockville, Maryland 20852

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 169
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 January 13, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 169, 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 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne Black

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 2, 1989

THIS PAGE INTENTIONALLY LEFT BLANK

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.

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.

3.5 Bases (Cont'd)

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)

There are two EECW headers (north and south) with four automatic starting RHRSW pumps on each header. All components requiring emergency cooling water are fed from both headers thus assuring continuity of operation if either header is OPERABLE. Each header alone can handle the flows to all components. Two RHRSW pumps can supply the full flow requirements of all essential EECW loads for any abnormal or postaccident situation.

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 RHRSW 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.

3.5 BASES (Cont'd)

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.

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. 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.

3.5.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 HPCIS 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 this required core cooling. The HPCI pump is designed to pump 5,000 gpm at reactor pressures between 1,120 and 150 psig. Two sources of water are available. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor.

When the HPCI System begins operation, the reactor depressurizes more rapidly than would occur if HPCI was not initiated due to the condensation of steam by the cold fluid pumped into the reactor vessel by the HPCI system. As the reactor vessel pressure continues to decrease, the HPCI flow momentarily reaches equilibrium with the flow through the break. Continued depressurization caused the break flow to decrease below the HPCI flow and the liquid inventory begins to rise. This type of response is typical of the small breaks. The core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the capacity range of the HPCI.

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 serves as a backup to the RCICS as a source of feedwater makeup during primary system isolation conditions. The ADS serves as a backup to the HPCIS for reactor depressurization for postulated transients and accident. The CSS and RHRS (LPCI) provide adequate core cooling at low reactor pressure when RCICS and ADS are no longer necessary. Considering the redundant systems, an allowable repair time of seven days was selected.

The HPCI and RCIC as well as all other Core Standby Cooling Systems must be OPERABLE when starting up from a Cold Condition. It is realized that the HPCI 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 the reactor pressure decreases below 100 psig. It is the intent of this specification to assure that when the reactor is being started up from a Cold Condition, the HPCI is not known to be inoperable.

3.5.F Reactor Core Isolation Cooling System (RCICS)

The various conditions under which the RCICS plays an essential role in providing makeup water to the reactor vessel have been identified by evaluating the various plant events over the full range of planned operations. The specifications ensure that the function for which the RCICS was designed will be available when needed. 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.

Because the low-pressure cooling systems (LPCI and core spray) are capable of providing all the cooling required for any plant event when nuclear system pressure is below 122 psig, the RCICS is not required below this pressure. Between 122 psig and 150 psig the RCICS need not provide its design flow, but reduced flow is required for certain events. RCICS design flow (600 gpm) is sufficient to maintain water level above the top of the active fuel for a complete loss of feedwater flow at design power (105 percent of rated).

Consideration of the availability of the RCICS reveals that the average risk associated with failure of the RCICS to cool the core when required is not increased if the RCICS is inoperable for no longer than seven days, provided that the HPCIS is OPERABLE during this period.

REFERENCE

1. Reactor Core Isolation Cooling System (BFNP FSAR Subsection 4.7)

3.5.G Automatic Depressurization System (ADS)

This specification ensures the OPERABILITY of the ADS under all conditions for which the depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low-pressure coolant injection (LPCI) and the core spray subsystems can operate to protect the fuel barrier. Note that this specification 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.

Because the automatic depressurization system does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS.

With two ADS valves known to be incapable of automatic operation, four valves remain OPERABLE to perform their ADS function. The ECCS loss-of-coolant accident analyses for small line breaks assumed that four of the six ADS valves were OPERABLE. Reactor operation with three ADS valves inoperable is allowed to continue for seven days provided that the HPCI system is OPERABLE. Operation with more than three of the six ADS valves inoperable is not acceptable.

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.

I. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

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.

3.5 BASES (Cont'd)

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. The limiting value for MAPLHGR is shown in Tables 3.5.I-1 and -2. The analyses supporting these limiting values are presented in Reference 1.

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 rated thermal power, the R factor would have to be less than 0.241 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 a maximum LHGR of 18.5 kW/ft for 7x7 fuel and 13.4 kW/ft. 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. 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.

Maximum Average Planar LHGR, LHGR, and MCPR

The MAPLHGR, 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.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 169
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 January 13, 1989, 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. 169, 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 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne Black

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 2, 1989

THIS PAGE INTENTIONALLY LEFT BLANK

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

LIMITING CONDITIONS FOR OPERATION

3.5.D **Equipment Area Coolers**

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

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

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

1. The HPCI system shall be OPERABLE:

(1) PRIOR TO STARTUP from a COLD CONDITION; or

(2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 122 psig, except as specified in Specification 3.5.E.2.

SURVEILLANCE REQUIREMENTS

4.5.D **Equipment Area Coolers**

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

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

1. HPCI Subsystem testing shall be performed as follows:

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

LIMITING CONDITIONS FOR OPERATION

3.5.E High Pressure Coolant Injection System (HPCIS)

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

3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 122 psig or less within 24 hours.

F. Reactor Core Isolation Cooling System (RCICS)

1. The RCICS shall be OPERABLE:

(1) PRIOR TO STARTUP from a COLD CONDITION; or

SURVEILLANCE REQUIREMENTS

4.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

e. Flow Rate at 150 psig — Once/operating cycle

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

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

2. No additional surveillances are required.

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

F. Reactor Core Isolation Cooling System (RCICS)

1. RCIC Subsystem testing shall be performed as follows:

a. Simulated Automatic Actuation Test — Once/operating cycle

LIMITING CONDITIONS FOR OPERATION

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

3.5.F.1 (Cont'd)

(2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 122 psig, except as specified in 3.5.F.2.

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

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

SURVEILLANCE REQUIREMENTS

4.5.F Reactor Core Isolation Cooling System (RCICS)

4.5.F.1 (Cont'd)

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

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

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

2. No additional surveillances are required.

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

LIMITING CONDITIONS FOR OPERATION

3.5.G Automatic Depressurization System (ADS)

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

   (1) PRIOR TO STARTUP from a COLD CONDITION, or,

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

2. If three of the six ADS valves are known to be incapable of automatic operation, the reactor may remain in operation for a period not to exceed 7 days, provided the HPCI system is OPERABLE. (Note that the pressure relief function of these valves is assured by Section 3.6.D of these specifications and that this specification only applies to the ADS function.) If more than three of the six ADS valves are known to be incapable of automatic operation, an immediate orderly shutdown shall be initiated, with the reactor in a HOT SHUTDOWN CONDITION in 6 hours, and in a COLD SHUTDOWN CONDITION in the following 18 hours.

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

SURVEILLANCE REQUIREMENTS

4.5.G Automatic Depressurization System (ADS)

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

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

2. No additional surveillances are required.

LIMITING CONDITIONS FOR OPERATION

3.5.H. Maintenance of Filled Discharge Pipe

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

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

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

SURVEILLANCE REQUIREMENTS

4.5.H. Maintenance of Filled Discharge Pipe

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

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

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

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

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

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.

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.

3.5 Bases (Cont'd)

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)

There are two EECW headers (north and south) with four automatic starting RHRSW pumps on each header. All components requiring emergency cooling water are fed from both headers thus assuring continuity of operation if either header is OPERABLE. Each header alone can handle the flows to all components. Two RHRSW pumps can supply the full flow requirements of all essential EECW loads for any abnormal or postaccident situation.

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 RHRSW 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.

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. 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.

3.5.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 HPCIS 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 this required core cooling. The HPCI pump is designed to pump 5,000 gpm at reactor pressures between 1,120 and 150 psig. Two sources of water are available. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor.

When the HPCI System begins operation, the reactor depressurizes more rapidly than would occur if HPCI was not initiated due to the condensation of steam by the cold fluid pumped into the reactor vessel by the HPCI system. As the reactor vessel pressure continues to decrease, the HPCI flow momentarily reaches equilibrium with the flow through the break. Continued depressurization caused the break flow to decrease below the HPCI flow and the liquid inventory begins to rise. This type of response is typical of the small breaks. The core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the capacity range of the HPCI.

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 serves as a backup to the RCICS as a source of feedwater makeup during primary system isolation conditions. The ADS serves as a backup to the HPCIS for reactor depressurization for postulated transients and accident. The CSS and RHRS (LPCI) provide adequate core cooling at low reactor pressure when RCICS and ADS are no longer necessary. Considering the redundant systems, an allowable repair time of seven days was selected.

The HPCI and RCIC as well as all other Core Standby Cooling Systems must be OPERABLE when starting up from a Cold Condition. It is realized that the HPCI 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 the reactor pressure decreases below 100 psig. It is the intent of this specification to assure that when the reactor is being started up from a Cold Condition, the HPCI is not known to be inoperable.

3.5.F Reactor Core Isolation Cooling System (RCICS)

The various conditions under which the RCICS plays an essential role in providing makeup water to the reactor vessel have been identified by evaluating the various plant events over the full range of planned operations. The specifications ensure that the function for which the RCICS was designed will be available when needed. 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.

Because the low-pressure cooling systems (LPCI and core spray) are capable of providing all the cooling required for any plant event when nuclear system pressure is below 122 psig, the RCICS is not required below this pressure. Between 122 psig and 150 psig the RCICS need not provide its design flow, but reduced flow is required for certain events. RCICS design flow (600 gpm) is sufficient to maintain water level above the top of the active fuel for a complete loss of feedwater flow at design power (105 percent of rated).

Consideration of the availability of the RCICS reveals that the average risk associated with failure of the RCICS to cool the core when required is not increased if the RCICS is inoperable for no longer than seven days, provided that the HPCIS is OPERABLE during this period.

REFERENCE

1. Reactor Core Isolation Cooling System (BFNP FSAR Subsection 4.7)

3.5.G Automatic Depressurization System (ADS)

This specification ensures the OPERABILITY of the ADS under all conditions for which the depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low-pressure coolant injection (LPCI) and the core spray subsystems can operate to protect the fuel barrier. Note that this specification 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.

Because the automatic depressurization system does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS.

With two ADS valves known to be incapable of automatic operation, four valves remain OPERABLE to perform their ADS function. The ECCS loss-of-coolant accident analyses for small line breaks assumed that four of the six ADS valves were OPERABLE. Reactor operation with three ADS valves inoperable is allowed to continue for seven days provided that the HPCI system is OPERABLE. Operation with more than three of the six ADS valves inoperable is not acceptable.

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.

I. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

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. The limiting value for MAPLHGR is shown in Tables 3.5.I-1, -2, -3, -4, -5, and -6. The analyses supporting these limiting values are presented in Reference 4.

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 rated thermal power, the MTPF would have to be greater than 10 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 BASES (Cont'd)

3.5.M. 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. Mooré 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.

Maximum Average Planar LHGR, LHGR, and MCPR

The MAPLHGR, 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.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 140
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 January 13, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 140, 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 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne Black

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 2, 1989

THIS PAGE INTENTIONALLY LEFT BLANK

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.

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.

3.5 Bases (Cont'd)

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)

There are two EECW headers (north and south) with four automatic starting RHRSW pumps on each header. All components requiring emergency cooling water are fed from both headers thus assuring continuity of operation if either header is OPERABLE. Each header alone can handle the flows to all components. Two RHRSW pumps can supply the full flow requirements of all essential EECW loads for any abnormal or postaccident situation.

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.

3.5 BASES (Cont'd)

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.

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. 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.

3.5.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 BASES (Cont'd)

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 HPCIS 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 this required core cooling. The HPCI pump is designed to pump 5,000 gpm at reactor pressures between 1,120 and 150 psig. Two sources of water are available. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor.

When the HPCI System begins operation, the reactor depressurizes more rapidly than would occur if HPCI was not initiated due to the condensation of steam by the cold fluid pumped into the reactor vessel by the HPCI system. As the reactor vessel pressure continues to decrease, the HPCI flow momentarily reaches equilibrium with the flow through the break. Continued depressurization caused the break flow to decrease below the HPCI flow and the liquid inventory begins to rise. This type of response is typical of the small breaks. The core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the capacity range of the HPCI.

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 serves as a backup to the RCICS as a source of feedwater makeup during primary system isolation conditions. The ADS serves as a backup to the HPCIS for reactor depressurization for postulated transients and accident. The CSS and RHRS (LPCI) provide adequate core cooling at low reactor pressure when RCICS and ADS are no longer necessary. Considering the redundant systems, an allowable repair time of seven days was selected.

The HPCI and RCIC as well as all other Core Standby Cooling Systems must be OPERABLE when starting up from a Cold Condition. It is realized that the HPCI 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 the reactor pressure decreases below 100 psig. It is the intent of this specification to assure that when the reactor is being started up from a Cold Condition, the HPCI is not known to be inoperable.

3.5.F Reactor Core Isolation Cooling System (RCICS)

The various conditions under which the RCICS plays an essential role in providing makeup water to the reactor vessel have been identified by evaluating the various plant events over the full range of planned operations. The specifications ensure that the function for which the RCICS was designed will be available when needed. 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.

Because the low-pressure cooling systems (LPCI and core spray) are capable of providing all the cooling required for any plant event when nuclear system pressure is below 122 psig, the RCICS is not required below this pressure. Between 122 psig and 150 psig the RCICS need not provide its design flow, but reduced flow is required for certain events. RCICS design flow (600 gpm) is sufficient to maintain water level above the top of the active fuel for a complete loss of feedwater flow at design power (105 percent of rated).

Consideration of the availability of the RCICS reveals that the average risk associated with failure of the RCICS to cool the core when required is not increased if the RCICS is inoperable for no longer than seven days, provided that the HPCIS is OPERABLE during this period.

REFERENCE

1. Reactor Core Isolation Cooling System (BFNP FSAR Subsection 4.7)

3.5.G Automatic Depressurization System (ADS)

This specification ensures the OPERABILITY of the ADS under all conditions for which the depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low-pressure coolant injection (LPCI) and the core spray subsystems can operate to protect the fuel barrier. Note that this specification 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.

Because the automatic depressurization system does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS.

With two ADS valves known to be incapable of automatic operation, four valves remain OPERABLE to perform their ADS function. The ECCS loss-of-coolant accident analyses for small line breaks assumed that four of the six ADS valves were OPERABLE. Reactor operation with three ADS valves inoperable is allowed to continue for seven days provided that the HPCI system is OPERABLE. Operation with more than three of the six ADS valves inoperable is not acceptable.

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.

I. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

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.

3.5 BASES (Cont'd)

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. The limiting value for MAPLHGR is shown in Tables 3.5.I-1 through 7. The analyses supporting these limiting values are presented in Reference 1.

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 rated thermal power, the MTPF would have to be greater than 10 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 a maximum LHGR of 18.5 kW/ft for 7x7 fuel and 13.4 kW/ft for 8x8, 8x8R, and P8x8R. 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 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.

Maximum Average Planar LHGR, LHGR, and MCPR

The MAPLHGR, 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.

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

ENCLOSURE 4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NO. DPR 33

AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NO. DPR-52

AND AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3

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

1.0 INTRODUCTION

By submittal dated January 13, 1989, the Tennessee Valley Authority (TVA) proposed to modify the Browns Ferry Nuclear Plant (BFN), Units 1, 2 and 3, Technical Specifications (TS) to delete certain Surveillance Requirement (SR) testing of redundant equipment during a Limiting Condition for Operation (LCO). TVA has proposed that in place of the existing TS SR testing for the remaining operable equipment during an LCO condition, TVA perform periodic American Society of Mechanical Engineers (ASME), Section XI tests coupled with monthly valve alignment checks. These ASME tests are typically conducted once every three months for both Emergency Core Cooling System (ECCS) pumps and valves per BFN TS 1.0.MM. These tests are more rigorous than the existing TS SR operability verifications.

2.0 EVALUATION

TVA has proposed, by submittal referenced above, to modify the BFN TS SR for ECCS related pumps during LCOs in which a redundant component is declared inoperable. Specifically, during instances where an ECCS loop is declared inoperable, TVA has proposed to delete the current requirements for an immediate demonstration of operability of the remaining redundant loop and the requirement for demonstration of operability of that same redundant loop "daily thereafter." TVA also proposes to add TS SR for monthly verification of correct valve position for valves in the injection flow path of the operable redundant loop.

The TVA submittal supporting these proposed TS changes contains proposed changes applicable to the BFN Units 1, 2 and 3 TS. However, TVA has stated that the contents of the TS for the three units differ slightly in wording. The intent of the existing TS, however, is the same and the TVA justification provided for the proposed changes is appropriate for all three BFN Units. The following list of proposed TS deletions and additions have been grouped together, as the justification for these changes is essentially the same.

CURRENT 4.5.A.2 - DELETE THE FOLLOWING SURVEILLANCE REQUIREMENT:

"When it is determined that one core spray loop is inoperable, at a time when operability is required, the other core spray loop, the RHRS (LPCI mode), and the diesel generators shall be demonstrated to be OPERABLE immediately. The OPERABLE core spray loop shall be demonstrated to be OPERABLE daily thereafter."

PROPOSED 4.5.A.2 - ADD THE FOLLOWING SURVEILLANCE REQUIREMENT:

"No additional surveillance required."

CURRENT 4.5.B.3 - DELETE THE FOLLOWING SURVEILLANCE REQUIREMENT:

"When it is determined that one RHR pump (LPCI mode) is inoperable at a time when operability is required, the remaining RHR pumps (LPCI mode) and active components in both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators shall be demonstrated to be OPERABLE immediately and daily thereafter."

PROPOSED 4.5.B.3 - ADD THE FOLLOWING SURVEILLANCE REQUIREMENT:

"No additional surveillance required."

CURRENT 4.5.B.5 - DELETE THE FOLLOWING SURVEILLANCE REQUIREMENT:

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

PROPOSED 4.5.B.5 - ADD THE FOLLOWING SURVEILLANCE REQUIREMENT:

"No additional surveillance required."

CURRENT 4.5.B.6 - DELETE THE FOLLOWING SURVEILLANCE REQUIREMENT:

"When it is determined that two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable at a time when operability is required, the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all active components in the access paths of the RHRS (containment cooling mode) shall be demonstrated to be OPERABLE immediately and daily thereafter until at least three RHR pumps (containment cooling mode) and associated heat exchangers are returned to normal service."

PROPOSED 4.5.B.6 - ADD THE FOLLOWING SURVEILLANCE REQUIREMENT:

"No additional surveillance required."

CURRENT 4.5.B.7 - DELETE THE FOLLOWING SURVEILLANCE REQUIREMENT:

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

PROPOSED 4.5.B.7 - ADD THE FOLLOWING SURVEILLANCE REQUIREMENT:

"No additional surveillance required."

CURRENT 4.5.B.12 - DELETE THE FOLLOWING SURVEILLANCE REQUIREMENT:

"When it is determined that one RHR pump or associated heat exchanger located on the unit cross-connection in the adjacent unit is inoperable at a time when operability is required, the remaining RHR pump and associated heat exchanger on the unit cross-connection and the associated diesel generator shall be demonstrated to be operable immediately and every 15 days thereafter until the inoperable pump and associated heat exchanger on the unit cross-connection and the associated diesel generator shall be demonstrated to be operable immediately and every 15 days thereafter until the inoperable pump and associated heat exchanger are returned to normal service."

PROPOSED 4.5.B.12 - ADD THE FOLLOWING SURVEILLANCE REQUIREMENT:

"No additional surveillance required."

CURRENT 4.5.C.2 - DELETE THE FOLLOWING SURVEILLANCES:

"a. If no more than two RHRSW pumps are inoperable, increased surveillance is not required.

b. When three RHRSW pumps are inoperable, the remaining pumps and associated essential control valves shall be operated weekly.

c. When four RHRSW pumps are inoperable, the remaining pumps, associated essential control valves, and associated diesel generators shall be operated daily."

PROPOSED 4.5.C.2 - ADD THE FOLLOWING SURVEILLANCE REQUIREMENT:

"No additional surveillance required."

CURRENT 4.5.C.4 - DELETE THE FOLLOWING SURVEILLANCE REQUIREMENT:

"When it is determined that one of the RHRSW pumps supplying standby coolant is inoperable at a time when operability is required, the operable RHRSW pump on the same header and the RHR heat exchanger header and associated essential control valves shall be demonstrated to be operable immediately and every 15 days thereafter."

PROPOSED 4.5.C.4 - ADD THE FOLLOWING SURVEILLANCE REQUIREMENT:

"No additional surveillance required."

CURRENT 4.5.E.2 - DELETE THE FOLLOWING SURVEILLANCE REQUIREMENT:

"When it is determined that the HPCIS is inoperable the ADS actuation logic, the RCICS, the RHRS (LPCI), and the CSS shall be demonstrated to be operable immediately. The RCICS and ADS logic shall be demonstrated to be operable daily thereafter."

PROPOSED 4.5.E.2 - ADD THE FOLLOWING SURVEILLANCE REQUIREMENT:

"No additional surveillance required."

CURRENT 4.5.F.2 - DELETE THE FOLLOWING SURVEILLANCE REQUIREMENT:

"When it is determined that the RCICS is inoperable, the HPCIS shall be demonstrated to be operable immediately."

PROPOSED 4.5.F.2 - ADD THE FOLLOWING SURVEILLANCE REQUIREMENT:

"No additional surveillance required."

CURRENT 4.5.G.2 - DELETE THE FOLLOWING SURVEILLANCE REQUIREMENT"

"When it is determined that three of the six ADS valves are incapable of automatic operation, the HPCIS shall be demonstrated to be OPERABLE immediately and daily thereafter as long as Specification 3.5.G.2 applies."

PROPOSED 4.5.G.2 - ADD THE FOLLOWING SURVEILLANCE REQUIREMENT:

"No additional surveillance required."

TVA has provided justification for the above proposed TS deletions and additions. TVA has stated that BFN TS 235 implemented ASME Section XI testing for Class 1, 2, and 3 components that are required to perform a specific function in shutting down the reactor or in mitigating the consequences of an accident. The frequency of these inservice tests is nominally every 3 months during normal plant operations.

The required ASME Section XI testing is a more rigorous verification of pump and valve operability than that required by the existing BFN TS operability verification for redundant operable equipment. Under the current BFN TS, the redundant operable ECCS and Reactor Core Isolation Cooling (RCIC) systems could be tested seven times a week, for the worst case, just to verify operability. TVA has stated that this is excessive testing and over the life of BFN could result in undue and unnecessary component wear. Daily testing also increases the probability of equipment failure, and the potential for human error during system line-up, actual system testing, or returning the operable system to service.

TVA has also stated that during the performance of this additional testing, the operable ECCS or RCIC loop could be in a condition where it may not be able to perform its intended safety function. In addition, this additional testing requires additional attention by the operator to line up the system for testing, perform the actual test, and restore it back to its required mode of operation. Doing this daily while the other ECCS or RCIC loop is inoperable would provide an additional and unnecessary distraction from the operator's other activities and responsibilities in the control room.

In summary, TVA has stated that by deleting the existing surveillance tests and complying with the ASME Section XI testing, BFN is in compliance with the intent of 10 CFR 50.55a and current industry standards.

The staff has evaluated the above TVA proposed changes and the justification provided and agrees with TVA's conclusion that ASME Section XI testing is a more rigorous verification of pump and valve operability and that daily testing of operable equipment could, over the life of BFN, result in undue and unnecessary component wear. By deleting the above existing TS SR tests and complying with the ASME Section XI test requirements and frequencies per BFN TS 1.0.MM, the staff concludes that BFN is in compliance with the intent of 10 CFR 50.55a and current industry standards. Thus, these changes are acceptable.

In order to provide additional verification of ECCS and RCIC system operability, TVA has proposed new TS SR for verification of proper valve positions in the injection flow path. The valve types to be position verified include manual, power-operated, and automatic valves which are not locked, sealed, or otherwise secured in position. The SR frequency proposed for these new TS is once per month. TVA has noted an exception for those cases where an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

The following list of proposed TS additions have been grouped together as the justification for these additions is essentially the same.

PROPOSED 4.5.A.1.f, 4.5.B.1.f, 4.5.E.1.f, and 4.5..F.1.f - ADD THE FOLLOWING SURVEILLANCE REQUIREMENT:

> "Verify that each valve (manual, power-operated, or automatic) in the injection flowpath is not locked, sealed, or otherwise secured in position, is in its correct* position.
>
> This will be performed "once/month."
>
> "*Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation."

These new SRs are being added to ensure the correct position of each of the systems' valves in the injection flowpath. This requires a monthly alignment check of the valves that are not locked or sealed in a position which may affect the ability of the system to perform its intended safety function.

This monthly check may be accomplished by visual inspection (where possible) or simulated automatic actuation signals in accordance with ASME Section XI pump and valve testing as required by BFN Technical Specification 1.0.MM. A valve that is capable of automatic return to its ECCS position, when an ECCS signal is present, can be in a position for another mode of operation during this verification. This is applicable only if the valve auto-repositions and fully opens within the time required for its ECCS function.

In addition, verification of the valves in the injection flowpath provides a passive check of the flow path to verify that the valves in the system are in their correct position.

As stated above, this verification process may be accomplished by visual inspection (where possible) or verification of flow through the appropriate flowpath when the appropriate pump is tested. The staff finds this method of testing is consistent with current industry standards, NRC accepted practices, and is in compliance with the intent of ASME Section XI and 10 CFR 50.55a(g) testing requirements. The addition of these SRs ensures that the subject safety injection flowpaths are aligned properly to allow the associated pumps to perform their intended safety function as analyzed by the BFN Final Safety Analysis Report and is acceptable.

PROPOSED 4.5.B.1.g - ADD THE FOLLOWING SURVEILLANCE REQUIREMENT:

> "Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator."
>
> This will be done "once/month".

TVA has stated that the low pressure coolant injection (LPCI) system is designed to provide emergency coolant by flooding the reactor core in the event of a loss-of-coolant accident (LOCA). Though completely independent of the core spray system (CSS), LPCI functions in combination with the CSS to prevent excessive fuel clad temperatures. The LPCI mode of the RHRS and the core spray system (CSS) provides adequate cooling for pipe break areas including the double ended recirculation pipe break. A cross-tie valve exists in order to provide the capability to supply water from one loop of LPCI to the other if needed.

This surveillance is being added to verify that the LPCI subsystem cross-tie valve is closed and electric power to the operator is disconnected to ensure that a failure of the potential flowpath in one LPCI subsystem will not affect the flowpath of the other LPCI system. The staff finds that this passive check provides additional assurance of operability with no adverse impact on the intended safety function of the LPCI system. Thus, this change is acceptable.

PROPOSED 4.5.C.1.c - INSERT THE FOLLOWING SURVEILLANCE REQUIREMENT:

> "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."

TVA has proposed to insert the above new TS 4.5.C.1.c in the BFN TS. The justification provided by TVA is the same as that provided for proposed new TS 4.5.A.1.f above, therefore, the staff conclusion is the same.

CURRENT 4.5.H.1 - CHANGE THE FOLLOWING SURVEILLANCE REQUIREMENT:

> "Every month prior to the testing..."

PROPOSED 4.5.H.1 - THE NEW CHANGE WILL READ:

> "Every month and prior to testing..."

TVA stated that the existing surveillance requires that every month prior to testing the residual heat removal system (RHRS) and the CSS systems, the systems shall be vented from the high points and water flow determined. Since these systems would now be tested quarterly, per ASME Section XI, the existing surveillance must be changed to ensure that the piping is still vented monthly and also vented prior to the testing of the RHRS to maintain the intent of the TS.

If the discharge piping of the subject systems are not filled with water, a water hammer may 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, the revised TS would require 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 the line must be assumed inoperable.

The discharge piping high point is visually checked for water flow once per month and prior to testing to ensure that the lines are filled and therefore minimize the potential water hammer. The proposed revision to this surveillance will help ensure this potential effect and is acceptable.

TVA has also proposed to modify the applicable TS Bases Section to reflect the proposed changes documented above.

The staff has evaluated all of the above proposed TS deletions and changes and has found that over the life of BFN, these proposed TS could result in a significant reduction in excessive and unnecessary component testing. The ASME Section XI testing frequencies proposed, once per 3 months, for the ECCS and RCIC system components (as required by BFN TS 1.0 MM) is adequate to ensure system functional operability, when combined with the monthly valve alignment checks, for instances when redundant equipment is declared inoperable. These frequencies are equivalent to those required by NUREG-0123, Revision 3, Standard Technical Specifications for General Electric Boiling Water Reactors. The proposed changes are also consistent with the intent of the testing required by 10 CFR 50.55a and current industry standards. Therefore, the staff finds the proposed deletions, additions and changes discussed above to be acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that these amendments involve no significant increases in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (54 FR 21316) on May 17, 1989 and consulted with the State of Alabama. No public comments were received and the State of Alabama did not have any comments.

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

Principal Contributor: T. Rotella

Dated: August 2, 1989