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FEBRUARY 25 1980

Docket Nos. 50-259
and 50-260

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

Dear Mr. Parris:

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

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

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

Sincerely,

Original Signed by
T. A. Ippolito

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

cp

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

8003130304

No legal objection to form of amendment or notice SER not reviewed

cc w/encl:
~~See next page~~

OFFICE	DOR:ORB#3	DOR:ORB#3	DOR:ORB#3	DOR:ORP	OELD
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 59
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 October 4, 1979 as supplemented by letters dated January 15, 1980 and January 29, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility License No. DPR-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. 59, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 25, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 59

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

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

vii/viii	25/26	133/134	159/160	181/182
9/10	29/30	145/146	167/168	218/219
11/12	97/98	147/148	169/170	220/221
15/16	111/112	149/150	171/172	254/255
17/18	131/132	157/158	172a	276/277
				330/331

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

3. Add the following page:

   172b

LIST OF TABLES (Cont'd)

Amendment No. 59

LIST OF ILLUSTRATIONS

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

B. Core Thermal Power Limit (Reactor Pressure $\leq$800 psia)

When the reactor pressure is less than or equal to 800 psia,

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

In the event of operation with the core maximum fraction of limiting power density (CMFLPD) greater than fraction of rated thermal power (FRP) the setting shall be modified as follows:

$$S \leq (0.66W + 54\%) \frac{FRP}{CMFLPD}$$

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

(Note: These settings assume operation within the basic thermal hydraulic design criteria. These criteria are LHGR $\leq$ 18.5 kw/ft for 7X7 fuel and $\leq$ 13.4 kw/ft for 8X8, 8x8R, and P8x8R fuel, MCPR limits of Spec 3.5.k. If it is determined that either of these design criteria is being violated during operation, action shall be initiated within 15 minutes to restore operation within prescribed limits. Surveillance requirements for APRM scram setpoint are given in specification 4.1.B.

2. APRM--When the reactor mode switch is in the STARTUP POSITION, the APRM scram shall be set at less than or equal to 15% of rated power.

3. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.

B. APRM Rod Block Trip Setting

The APRM Rod block trip setting shall be:

SAFETY LIMIT

2.1 FUEL CLADDING INTEGRITY

or core coolant flow is less than 10% of rated, the core thermal power shall not exceed 823 MWt (about 25% of rated thermal power).

C. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 17.7 in. above the top of the normal active fuel zone.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

$$S_{RB} \leq (0.66W + 42\%)$$

where:

S_{RB} = Rod block setting is percent of rated thermal power (3293 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

In the event of operation with the core maximum fraction of limiting power density (CMFLPD) greater than fraction of rated thermal power (FRP) the setting shall be modified as follows:

$$S_{RB} \leq \{0.66W + 42\%\} \frac{FRP}{CMFLPD}$$

C. Scram and isolation--reactor low water $\geq$ 538 in. above vessel zero level

D. Scram--turbine stop valve closure $\leq$ 10 percent valve closure

E. Scram--turbine control valve

1. Fast closure — Upon trip of the fast acting solenoid valves

2. Loss of control oil pressure $\geq$ 550 psig

F. Scram--low condenser vacuum $\geq$ 23 inches Hg vacuum

G. Scram--main steam line isolation $\leq$ 10 percent valve closure

H. Main steam isolation valve closure--nuclear system low pressure $\geq$ 825 psig

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
1.1 Fuel Cladding Integrity	2.1 Fuel Cladding Integrity

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

Amendment No. 59

FIGURE DELETED

2.1 BASES: FUEL CLADDING INTEGRITY SAFETY LIMIT

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

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

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

.1.1 BASES

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of MCPR =1.07 would not produce boiling transition. Thus, although it is not required to establish the safety limit additional margin exists between the safety limit and the actual occurence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to BFNP operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operating (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

In addition to the boiling transition limit (MCPR = 1.06) operation is constrained to a maximum LHGR of 18.5 kw/ft for 7x7 fuel and 13.4 kw/ft for all 8x8 fuels. This limit is reached when the Core Maximum Fraction of Limiting Power Density equals 1.0 (CMFLPD = 1.0). For the case where Core Maximum Fraction of Limiting Power Density exceeds the Fraction of Rated Thermal Power, operation is permitted only at less than 100% of rated power and only with reduced APRM scram settings as required by specification 2.1.A.1.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flow will always be greater than 4.56 psi. Analyses show that with a flow of $28X10^3$ lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than $28x10^3$ lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

For the fuel in the core during periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If water level should drop below the top of the fuel during this time, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation. As long as the fuel remains covered with water, sufficient cooling is available to prevent fuel clad perforation.

1.1 BASES

The safety limit has been established at 17.7 in. above the top of the irradiated fuel to provide a point which can be monitored and also provide adequate margin. This point corresponds approximately to the top of the actual fuel assemblies and also to the lower reactor low water level trip (378" above vessel zero).

REFERENCE

1. General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO 10958 and NEDE 10958.

PAGE DELETED

2.1 BASES

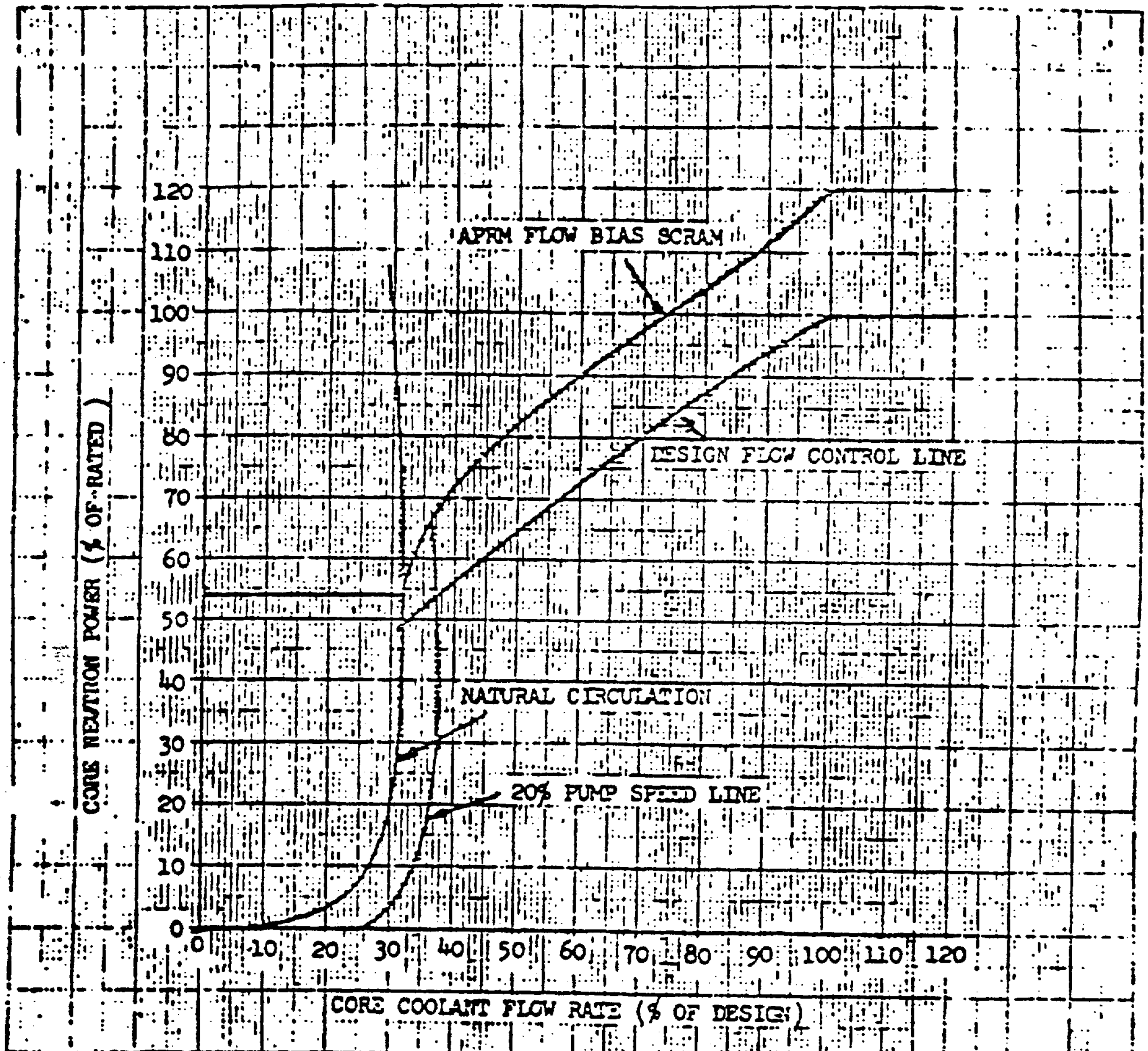
I. J. & K. Reactor low water level set point for initiation of HPCI and RCIC, closing main steam isolation valves, and starting LPCI and core spray pumps.

These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram set point and initiation set points. Transient analyses reported in Section 14 of the FSAR demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

L. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, Feb., 1973.

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



APRM FLOW BIAS SCRAM Vs. REACTOR CORE FLOW
FIG. 2.1-2

1.2 BASES

pressure monitor higher in the vessel. Therefore, following any transient that is severe enough to cause concern that this safety limit was violated, a calculation will be performed using all available information to determine if the safety limit was violated.

REFERENCES

1. Plant Safety Analysis (BFNP FSAR Section 14.0)
2. ASME Boiler and Pressure Vessel Code Section III
3. USAS Piping Code, Section B31.1
4. Reactor Vessel and Appurtenances Mechanical Design (BFNP FSAR Subsection 4.2)
5. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

To meet the safety design basis, thirteen relief valves have been installed on the unit with a total capacity of 82.6% of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steamline isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves operable, results in adequate margin to the code allowable overpressure limit of 1375 psig.

To meet the operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open)shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowable vessel overpressure of 1375 psig.

TABLE 4.2.B (Continued)

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

TABLE 4.2.B (Continued)

Function	Functional Test	Calibration	Instrument Check
Instrument Channel RHR Pump Discharge Pressure	(1)	once/3 months	none
Instrument Channel Core Spray Pump Discharge Pressure	(1)	once/3 months	none
Core Spray Sparger to RPV d/p	(1)	once/3 months	once/day
Trip System Bus Power Monitor	once/operating cycle	N/A	none
Instrument Channel Condensate Storage Tank Low Level	(1)	once/3 months	none
Instrument Channel Suppression Chamber High Level	(1)	once/3 months	none
Instrument Channel Reactor High Water Level	(1)	once/3 months	once/day
Instrument Channel RCIC Turbine Steam Line High Flow	(1)	once/3 months	none
Instrument Channel RCIC Steam Line Space High Temperature	(1)	once/3 months	none

3.2 BASES

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

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

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

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

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

The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel (Table 3.2.B) also initiate the RCIC and HPCI,

3.2 BASES

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

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

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

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

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

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

does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

   A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit, (ie, MCPR given by Spec. 3.5.K or LHGR of 18.5 for 7x7 or 13.4 for 8x8,8x8R,& P8x8R). During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is normally the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform these functions may be designated by the plant superintendent to perform these functions.

Scram Insertion Times

The control rod system is designated to bring the reactor subcritical at the rate fast enough to prevent fuel damage; ie, to prevent the MCPR from becoming less than 1.07. The limiting power transient is given in Reference 1. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification provide the required protection, and MCPR remains greater than 1.07.

On an early BWR, some degradation of control rod scram performance occured during plant startup and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model drive with a modified (larger screen size) internal filter which is less prone to plugging. Data has been documented by surveillance reports in various operating plants. These include Oyster Creek, Monticello, Dresden 2 and Dresden 3. Approximately 5000 drive tests have been recorded to date.

Following identification of the "plugged filter" problem, very frequent scram tests were necessary to ensure proper performance. However, the more frequent scram tests are now considered totally unnecessary and unwise for the following reasons:

1. Erratic scram performance has been identified as due to an obstructed drive filter in type "A" drives. The drives in BFNP are of the new "B" type design whose scram performance is unaffected by filter condition.

2. The dirt load is primarily released during startup of the reactor when the reactor and its systems are first subjected to flows and pressure and thermal stresses. Special attention and measures are now being taken to assure cleaner systems. Reactors with drives identical or similar (shorter stroke, smaller piston areas) have operated through many refueling cycles with no sudden or erratic changes in scram performance. This preoperational and startup testing is sufficient to detect anomalous drive performance.

3. The 72-hour outage limit which initiated the start of the frequent scram testing is arbitrary, having no logical basis other than quantifying a "major outage" which might reasonably be caused by an event so severe as to possibly affect drive performance. This requirement is unwise because it provides an incentive for shortcut actions to hasten returning "on line" to avoid the additional testing due a 72-hour outage.

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10% of the control rods at 16-week intervals is adequate for determining the operability of the control rod system yet is not so frequent as to cause excessive wear on the control rod system components.

The numerical values assigned to the predicted scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on Browns Ferry Nuclear Plant.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

In the analytical treatment of the transients, 390 milliseconds are allowed between a neutron sensor reaching the scram point and the start of negative reactivity insertion. This is adequate and conservative when compared to the typically observed time delay of about 270 milliseconds. Approximately 70 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid power supply voltage goes to zero an approximately 200 milliseconds later, control rod motion begins. The 200 milliseconds are included in the allowable scram insertion times specified in Specification 3.3.C.

* In order to perform scram time testing as required by specification 4.3.C.1, the relaxation of certain restraints in the rod sequence control system is required. Individual rod bypass switches may be used as described in specification 4.3.C.1.

The position of any rod bypassed must be known to be in accordance with rod withdrawal sequence. Bypassing of rods in the manner described in specification 4.3.C.1 will allow the subsequent withdrawal of any rod scrammed in the 100 percent to 50 percent rod density groups; however, it will maintain group notch control over all rods in the 50 percent density to preset power level range. In addition, RSCS will prevent movement of rods in the 50 percent density to preset power level range until the scrammed rod has been withdrawn.

3.3/4.4 BASES:

D. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% ΔK Deviations in core reactivity greater than 1% Δk are not expected and require thorough evaluation. One percent reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

References

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

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

1. The RHRS shall be operable:

   (1) prior to a reactor startup from a Cold Condition; or

   (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in specifications 3.5.B.2, through 3.5.B.7 and 3.9.B.3.

2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain operable) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are operable.

3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain operable.

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

1.

a.	Simulated Automatic Actuation Test	Once/ Operating Cycle
b.	Pump Operability	Once/ month
c.	Motor Operated valve operability	Once/ month
d.	Pump Flow Rate	Once/3 months
e.	Test Check Valve	Once/ Operating Cycle

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

2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

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

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

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

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

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

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

4. No additional surveillance required.

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

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

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

are operable.

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

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

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

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

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

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

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

8. No additional surveillance required.

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

10. No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION

3.5.B <u>Residual Heat Removal System (RHRS)</u> (LPCI and Containment Cooling)

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

12. If one RHR pump or associated heat exchanger located on the unit cross-connection in the adjacent unit is inoperable for any reason (including valve inoperability, pipe break, etc.), the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pump and associated diesel generator are operable.

SURVEILLANCE REQUIREMENTS

4.5.B <u>Residual Heat Removal System (RHRS)</u> (LPCI and Containment Cooling)

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

12. When it is determined that 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 are returned to normal service.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
13. If RHR cross-connection flow or heat removal capability is lost, the unit may remain in operation for a period not to exceed 10 days unless such capability is restored.	13. No additional surveillance required.
14. All recirculation pump discharge valves shall be operable prior to reactor startup (or closed if permitted elsewhere in these specifications).	14. All recirculation pump discharge valves shall be tested for operability during any period of reactor cold shutdown exceeding 48 hours, if operability tests have not been performed during the preceding 31 days.

3.5.F Reactor Core Isolation Cooling

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

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

G. Automatic Depressurization System (ADS)

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

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

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

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

4.5.F Reactor Core Isolation Cooling

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

G. Automatic Depressurization System (ADS)

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

   a. A simulated automatic actuation test shall be performed prior to startup after each refueling outage. Manual surveillance of the relief valves is covered, in 4.6.D.2.

2. When it is determined that more than two of the ADS valves are incapable of automatic operation, the HPCIS shall be demonstrated to be operable immediately and daily thereafter as long as Specification 3.5.G.2 applies.

3.5.G Automatic Depressurization System (ADS)

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

H. Maintenance of Filled Discharge Pipe

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

4.5.G Automatic Depressurization System (ADS)

H. Maintenance of Filled Discharge Pipe

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

3.5.H **Maintenance of Filled Discharge Pipe**

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.

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

I. **Average Planar Linear Heat Generation Rate**

During steady state power operation, the Maximum Average Planar Heat Generation Rate (MAPLHGR) for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Tables 3.5.I-1, -2, -3, -4, -5, -6, & -7. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

J. **Linear Heat Generation Rate (LHGR)**

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

4.5.H **Maintenance of Filled Discharge Pipe**

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

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

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

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

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

The MAPLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

J. **Linear Heat Generation Rate (LHGR)**

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

$LHGR_{max} < LHGR_{max} - (\Delta P/P) \quad (L/LT)$

$LHGR_d$ = Design LHGR = 18.5 kW/ft for 7x7 fuel
=13.4 kW/ for 8x8 fuel
8x8R and P8x8R fuel

$(\Delta P/P)$ = Maximum power spiking penalty
= 0.026 for 7x7 fuel
= 0.022 for 8x8,8x8R and P8x8R fuel

LT* = Total core length = 12.0 ft for 7x7 fuel and 8x8
= 12.5 ft for 8x8, 8x8R & P8x8R

L = Axial position above bottom of core

If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

K. Minimum Critical Power Ratio (MCPR)

From BOC to EOC-2000 MWD/T the MCPR operating limit for BFNP 1 cycle 4 is 1.23 for 7x7 fuel, 1.24 for 8x8 fuel, and 1.25 for 8x8R and P8x8R fuel. These limits apply to steady state power operation at rated power and flow. For core flows other than rated the MCPR shall be greater than the above limits times K_f. K_f is the value shown in Figure 3.5.2. From EOC -2000 to EOC the MCPR limits will be 1.23, 1.27, and 1.28 for 7x7, 8x8 and 8x8R/P8x8R respectively. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

L. Reporting Requirements

If any of the limiting values identified in Specifications 3.5.I, J, or K are exceeded and the specified action is taken, the event shall be logged and reported in a 30-day written report.

K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.

* 12.5 feet for 8x8R fuel

3.5 BASES

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 demonstrated to be operable. Operation with more than three of the six ADS valves inoperable is not acceptable.

3.5 BASES

3.5.H Maintenance of Filled Discharge Pipe

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

The core spray and RHR system discharge piping high point vent is visually checked for water flow once a month 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 highpoint 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 willreflect approximately 30 psig for a water level at the high point and 45 psig for a water level in the pressuresuppression chamber head tank and are monitored daily to ensure that the discharge lines are filled.

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

3.5.I. 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 10CFR50, 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 10CFR50 Appendix K limit. The limiting value for MAPLHGR is shown in Tables 3.5.I-1, -2, -3, -4, -5, -6, and -7.per 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 power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 as modified in References 2 and 3, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% 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% 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%, 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% 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. Reporting Requirements

The LCO's associated with monitoring the fuel rod operating conditions are required to be met at all times, i.e., there is no allowable time in which the plant can knowingly exceed the limiting values for MAPLHGR, LHGR, and MCPR. It is a requirement, as stated in Specifications 3.5.I, .J, and .K. that if at any time during steady state power operation, it is determined that the limiting values for MAPLHGR, LHGR, or MCPR are exceeded action is then initiated to restore operation to within the prescribed limits. This action is initiated as soon as normal surveillance indicates that an operating limit has been reached. Each event involving steady state operation beyond a specified limit shall be logged and reported quarterly. It must be recognized that there is always an action which would return any of the parameters (MAPLHGR, LHGR, or MCPR) to within prescribed limits, namely power reduction. Under most circumstances, this will not be the only alternative.

M. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-10735, August 1973.
2. Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (USA Regulatory Staff).
3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.
4. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and addenda.

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, judgement 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 each month to assure their operability. A simulated automatic actuation test once each cycle combined with monthly tests of the pumps and injection valves is deemed to be adequate testing of these systems.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, cause the outage, then the demonstration of operability should be thorough enough to assure that a generic problem does not exist. For example, if an out-of-service period was caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

Whenever a CSCS system or loop is made inoperable because of a required test or calibration, 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.

Redundant operable components are subjected to increased testing during equipment out-of-service times. This adds further conservatism and increases assurance that adequate cooling is available should the need arise.

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.

Table 3.5.I-1

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: Initial Core - Type 1 & 3

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)
200	15.0
1,000	15.1
5,000	16.0
10,000	16.3
15,000	16.1
20,000	15.4
25,000	14.2
30,000	13.1

Table 3.5.I-2

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: Initial Core - Type 2

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)
200	15.6
1,000	15.5
5,000	16.2
10,000	16.5
15,000	16.5
20,000	15.8
25,000	14.5
30,000	13.3

Table 3.5.I-3

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8D274L

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.3
5,000	11.9
10,000	12.1
15,000	12.2
20,000	12.1
25,000	11.6
30,000	10.9

Table 3.5.I-4

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8D274H

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)
200	11.1
1,000	11.2
5,000	11.8
10,000	12.1
15,000	12.2
20,000	12.0
25,000	11.5
30,000	10.9

TABLE 3.5.I-5

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DR265H

Average Planar Exposure (MWd/t)	MAPLHGR (kw/ft))
200	11.5
1000	11.6
5000	11.9
10,000	12.1.
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7

TABLE 3.5.I-6

MAPPHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DR265L

Average Planar Exposure (MWd/t)	MAPLHGR (kw/ft)
200	11.6
1000	11.6
5000	12.1
10,000	12.1
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7

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TABLE 3.5.I-7

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: P8DR284L

Average Planar Exposure (Mwd/t)	MAPLHGR (kw/ft)
200	11.2
1000	11.3
5000	11.8
10000	12.0
15000	12.0
20000	11.8
25000	11.2
30000	10.8

3.6.C Coolant Leakage

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

D. **Relief Valves**

1. When more than one valve, is known to be failed, an ordery shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours.

E. **Jet Pumps**

1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be shutdown in the Cold Condition within 24 hours.

4.6.C Coolant Leakage

D. **Relief Valves**

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

2. Once during each operating cycle, each relief valve shall be manually opened until thermocouples downstream of the valve indicate steam is flowing from the valve.

3. The integrity of the relief valve bellows shall be continuously monitored.

4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. **Jet Pumps**

1. Whenever there is recirculation flow with the reactor in the startup or run modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

   a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

3.6.E Jet Pumps

3.6.F Jet Pump Flow Mismatch

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

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

3. Steady state operation with both recirculation pumps out of service for up to 12 hrs is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature of the reactor vessel water as determined by dome pressure.

G. Structural Integrity

1. The structural integrity of the primary system shall be

4.6.E Jet Pumps

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

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

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

F. Jet Pump Flow Mismatch

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

G. Structural Integrity

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

3.6/4.6 BASES:

The basis for the equilibrium coolant iodine activity limit is a computed dose to the thyroid of 36 rem at the exclusion distance during the 2-hour period following a steam line break. This dose is computed with the conservative assumption of a release of 140,000 lbs of coolant prior to closure of the isolation valves, and a X/Q value of 3.4 x 10^{-4} Sec/m^3.

The maximum activity limit during a short term transient is established from consideration of a maximum iodine inhalation dose less than 300 rem. The probability of a steam line break accident coincident with an iodine concentration transient is significantly lower than that of the accident alone, since operation of the reactor with iodine levels above the equilibrium value is limited to 5 percent of total operation.

The sampling frequencies are established in order to detect the occurrence of an iodine transient which may exceed the equilibrium concentration limit, and to assure that the maximum coolant iodine concentrations are not exceeded. Additional sampling is required following power changes and off-gas transients, since present data indicate that the iodine peaking phenomenon is related to these events.

3.6.C/4.6.C Coolant Leakage

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite a-c power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the unit should be shut down to allow further investigation and corrective action.

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

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

REFERENCES

1. Nuclear System Leakage Rate Limits (BFNP FSAR Subsection 4.10)

3.6.D/4.6.D

Relief Valves

To meet the safety basis thirteen relief valves have been installed on the unit with a total capacity of 82.6% of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves inoperable, results in adequate margin to the code allowable overpressure limit of 1375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1375 psig.

3.6/4.6 BASES:

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

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

REFERENCES

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

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

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

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

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

3.6.E/4.6.E Jet Pumps

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

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

3.6/4.6 BASES:

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

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

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

NOTES FOR TABLE 3.7.A

Key: O = Open

C = Closed

SC = Stays Closed

GC = Goes Closed

Note: Isolation groupings are as follows:

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

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

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

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

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

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

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

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

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

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

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

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

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

1. Reactor vessel low water level (470")

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

2. High Drywell pressure

in the rugged shipboard environment on the NS Savannah (ORNL 3726). Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow RDT Standard M-16-1T. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

All elements of the heater should be demonstrated to be functional and operable during the test of heater capacity. Operation of each filter train for a minimum of 10 hrs each month will prevent moisture buildup in the filters and adsorber system.
With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repairs and test repeated.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other systems must be tested daily. This substantiates the availability of the operable systems and thus reactor operation and refueling operation can continue for a limited period of time.

3.7.D/4.7.D Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss of coolant accident.

BASES

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

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

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

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

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

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

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

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

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

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

Browns Ferry unit 1 is located at Browns Ferry Nuclear Plant site on property owned by the United States and in custody of the TVA. The site shall consist of approximately 840 acres on the north shore of Wheeler Lake at Tennessee River Mile 294 in Limestone County, Alabama. The minimum distance from the outside of the secondary containment building to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 4,000 feet.

5.2 REACTOR

A. The reactor core may contain 764 fuel assemblies consisting of 7x7 assemblies having 49 fuel rods each, 8x8 assemblies having 63 fuel rods each, and 8x8R (and P8x8R) assemblies having 62 fuel rods each.

B. The reactor core shall contain 185 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70 percent of theoretical density.

5.3 REACTOR VESSEL

The reactor vessel shall be as described in Table 4.2-2 of the FSAR. The applicable design codes shall be as described in Table 4.2-1 of the FSAR.

5.4 CONTAINMENT

A. The principal design parameters for the primary containment shall be as given in Table 5.2-1 of the FSAR. The applicable design codes shall be as described in Section 5.2 of the FSAR.

B. The secondary containment shall be as described in Section 5.3 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with the standards set forth in Section 5.2.3.4 of the FSAR.

5.5 FUEL STORAGE

A. The arrangement of fuel in the new-fuel storage facility shall be such that k_{eff}, for dry conditions, is less than 0.90 and flooded is less than 0.95 (Section 10.2 of FSAR).

5.0 MAJOR DESIGN FEATURES (Continued)

B. The k_{eff} of the spent fuel storage pool shall be less than or equal to 0.95. Fuel stored in the pool shall not contain more than 15.2 grams of uranium-235 per axial centimeter of fuel assembly.

C. Loads greater than 1000 pounds shall not be carried over spent fuel assemblies stored in the spent fuel pool.

5.6 SEISMIC DESIGN

The station class I structures and systems have been designed to withstand a design basis earthquake with ground acceleration of 0.2g. The operational basis earthquake used in the plant design assumed a ground acceleration of 0.1g (see Section 2.5 of the FSAR).

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 54
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 4, 1979, as supplemented by submittals dated January 15, 1980 and January 29, 1980 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 54 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

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

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: February 25, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 54

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

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

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

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

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
1.1 **Fuel Cladding Integrity**	2.1 **Fuel Cladding Integrity**

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

Amendment No. 54

FIGURE DELETED

TABLE 4.2.B (Continued)

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

TABLE 4.2.B (Continued)

Function	Functional Test	Calibration	Instrument Check
Instrument Channel RHR Pump Discharge Pressure	(1)	once/3 months	none
Instrument Channel Core Spray Pump Discharge Pressure	(1)	once/3 months	none
Core Spray Sparger to RPV d/p	(1)	once/3 months	once/day
Trip System Bus Power Monitor	once/operating cycle	N/A	none
Instrument Channel Condensate Storage Tank Low Level	(1)	once/3 months	none
Instrument Channel Suppression Chamber High Level	(1)	once/3 months	none
Instrument Channel Reactor High Water Level	(1)	once/3 months	once/day
Instrument Channel RCIC Turbine Steam Line High Flow	(1)	once/3 months	none
Instrument Channel RCIC Steam Line Space High Temperature	(1)	once/3 months	none

3.2 BASES

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

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

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

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

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

The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel (Table 3.2.B) also initiate the RCIC and HPCI.

3.2 BASES

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

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

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

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

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

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

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

1. The RHRS shall be operable:

   (1) prior to a reactor startup from a Cold Condition; or

   (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in specifications 3.5.B.2, through 3.5.B.7 and 3.9.B.3.

2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain operable) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are operable.

3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain operable.

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

1. a. Simulated Automatic Actuation Test — Once/Operating Cycle

   b. Pump Operability — Once/month

   c. Motor Operated valve operability — Once/month

   d. Pump Flow Rate — Once/3 months

   e. Test Check Valve — Once/Operating Cycle

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

2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

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

PAGE DELETED

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

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

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

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

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

4. No additional surveillance required

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

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

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

are operable.

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

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

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

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

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

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

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

8. No additional surveillance required.

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

10. No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

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

13. No additional surveillance required.

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

3.5.F Reactor Core Isolation Cooling

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

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

G. Automatic Depressurization System (ADS)

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

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

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

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

4.5.F Reactor Core Isolation Cooling

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

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.[illegible].2.

2. When it is determined that more than two of the ADS valves are incapable of automatic operation, the HPCIS shall be demonstrated to be operable immediately and daily thereafter as long as Specification 3.5.G.2 applies.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.5.G Automatic Depressurization System (ADS)

3. If more than two 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 is operable.

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

H. Maintenance of Filled Discharge Pipe

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

4.5.G Automatic Depressurization System (ADS)

3. When it is determined that more than two ADS valves are incapable of automatic operation, the HPCIS shall be shown to be operable immediately and daily thereafter as long as 3.5.G.3 applies.

H. Maintenance of Filled Discharge Pipe

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

3.6.C Coolant Leakage

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

D. Safety and Relief Valves

1. When more than one relief valve or one or more safety valves are known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours.

E. Jet Pumps

1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be shutdown in the Cold Condition within 24 hours.

4.6.C Coolant Leakage

D. Safety and Relief Valves

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

2. Once during each operating cycle, each relief valve shall be manually opened until thermocouples downstream of the valve indicate steam is flowing from the valve.

3. The integrity of the relief/safety valve bellows shall be continuously monitored.

4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the startup or run modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

   a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

3.6.E Jet Pumps

3.6.F Jet Pump Flow Mismatch

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

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

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

G. Structural Integrity

1. The structural integrity of the primary system shall be

4.6.E Jet Pumps

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

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

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

F. Jet Pump Flow Mismatch

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

G. Structural Integrity

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

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

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

3.6.F/4.6.F Jet Pump Flow Mismatch

3.6/4.6 BASES:

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

3.6.G/4.6.G Structural Integrity

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

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

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

Only proven nondestructive testing techniques will be used.

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

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

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
	Standby liquid control system check valves CV 63-526 & 525	1	1	NA	C	Process
	Feedwater check valves CV-3-558, 572, 554, & 568	2	2	NA	0	Process
	Control rod hydraulic return check valves CV-85-576 & 573	1	1	NA	0	Process
	RHRS - LPCI to reactor check valves CV-74-54 & 68	2		NA	C	Process

NOTES FOR TABLE 3.7.A

Key: O = Open

C = Closed

SC = Stays Closed

GC = Goes Closed

Note: Isolation groupings are as follows:

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

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

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

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

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

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

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

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

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

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

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

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

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

1. Reactor vessel low water level (470")

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

2. High Drywell pressure

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

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

BASES

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

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

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

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

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

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

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

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

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

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

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

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

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

3.7.E/4.7.E Control Room Emergency Ventilation

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

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

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

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS NOS. 1 AND 2

DOCKET NOS. 50-259 AND 50-260

1.0 Introduction

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

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

2.0 Discussion

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

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

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

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

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

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

3.0 Evaluation

3.1 Core Reload

3.1.1 Nuclear Characteristics

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

3.1.2 Thermal Hydraulics

3.1.2.1 Fuel Cladding Integrity Safety Limit MCPR

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

3.1.2.2 Operating Limit MCPR

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

3.1.2.2.1 Transient Analysis Methods

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

3.1.2.2.2 Transient Analysis Results

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

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

3.1.3 Accident Analyses

3.1.3.1 ECCS Appendix K Analysis

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

3.1.3.2 Control Rod Drop Accident

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

3.1.3.3 Fuel Loading Error

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

3.1.3.4 Overpressure Analysis

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

3.1.4 Thermal Hydraulic Stability

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

3.1.5 Startup Test Program

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

3.2 Other Changes to Technical Specifications

3.2.1 Reactor Low Water Level

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

3.2.2 Surveillance Requirements in Section 4.5.B

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

3.2.3 LPCI Modifications

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

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

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

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

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

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

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

3.2.4 Single Loop Operation

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

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

4.0 Environmental Considerations

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendments.

5.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: February 25, 1980

References

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

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

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

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

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-259 AND 50-260

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

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

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

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendments dated October 4, 1979, as supplemented by submittals dated January 15, 1980 and January 29, 1980, (2) Amendment No. 59 to License No. DPR-33 and Amendment No. 54 to License No. DPR-52 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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