11/12/80

Distribution:
Docket
NRC PDR
Local PDR
ORB Reading
NRR Reading
DEisenhut
RPurple
TNovak
RTedesco
GLainas
JRoe
SNorris
RJClark
TIppolito
OELD
OI&E (5)
BJones ~~(12)~~ 4
BScharf (10)
JWetmore
ACRS (16)
OPA (Clare Miles)
RDiggs
HDenton
JHeltemes, AEOD
NSIC
TERA

Docket No. 50-260

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

Dear Mr. Parris:

The Commission has issued the enclosed Amendment No. 58 to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit No. 2. This amendment changes the Technical Specifications in response to your request of July 14, 1980 (BFNP TS 140), as supplemented by your letters of August 29, and October 7, 1980. This amendment permits operation of Browns Ferry Unit No. 2 in Cycle No. 4 following the current refueling outage.

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

Sincerely,

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

Enclosures:
1. Amendment No. 58 to DPR-52
2. Safety Evaluation
3. Notice

cc w/encls:
See next page

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no legal objection to amend & FR notice

OFFICE	DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:OR	OELD	
SURNAME	SNorris	RJClark	TAIppolito	TANovak		
DATE	10/28/80	10/28/80	11/12/80	10/ /80	10/ /80	

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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. 58
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 July 14, 1980, as supplemented by letters dated August 29 and October 7, 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. 58, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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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 #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 12, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 58

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:

9/10
15/16
17/18
25/26
27/28
29/30
79/80
131/132
133/134
159/160
167/168
169/170
172a
219/220
249/250
269/270
329/330

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

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

B. Core Thermal Power Limit (Reactor Pressure ≤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 ≤ 18.5 kw/ft for 7x7 fuel and ≤ 13.4 kw/ft for 8x8, 8x8R, and P8x8R, and MCPR within limits of Specification 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:

7

SAFETY LIMIT

1.1 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 ≥ 538 in. above vessel zero lev[el]

D. Scram--turbine stop valve closure ≤ 10 percent valve closure

E. Scram--turbine control valve

1. Fast closure — Upon trip of the fast actin[g] solenoid valves

2. Loss of control oil pressure ≥ 550 psig

F. Scram--low condenser vacuum ≥ 23 inches Hg vacuum

G. Scram--main steam line isolation ≤ 10 percent valve closure

H. Main steam isolation valve closure--nuclear system low pressure ≥ 825 psig

Amendment No. 32

8

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 perforation 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.

9

.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 = 1.07) operation is constrained to a maximum LHGR of 18.5 kw/ft for 7x7 fuel and 13.4 kw/ft for 8x8, 8x8R, and P8x8R fuel. This limit is reached when the Core Maximum Fraction of Limiting Power Density equals 1.0 (CMFLPD = 1.0). For the case where Core Maximum Fraction of Limiting Power Density exceeds the Fraction of Rated Thermal Power, operation is permitted only at less than 100% of rated power and only with reduced APRM scram settings as required by specification 2.1.A.1.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flow will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

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

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.

11

PAGE DELETED

12

2.1 BASES

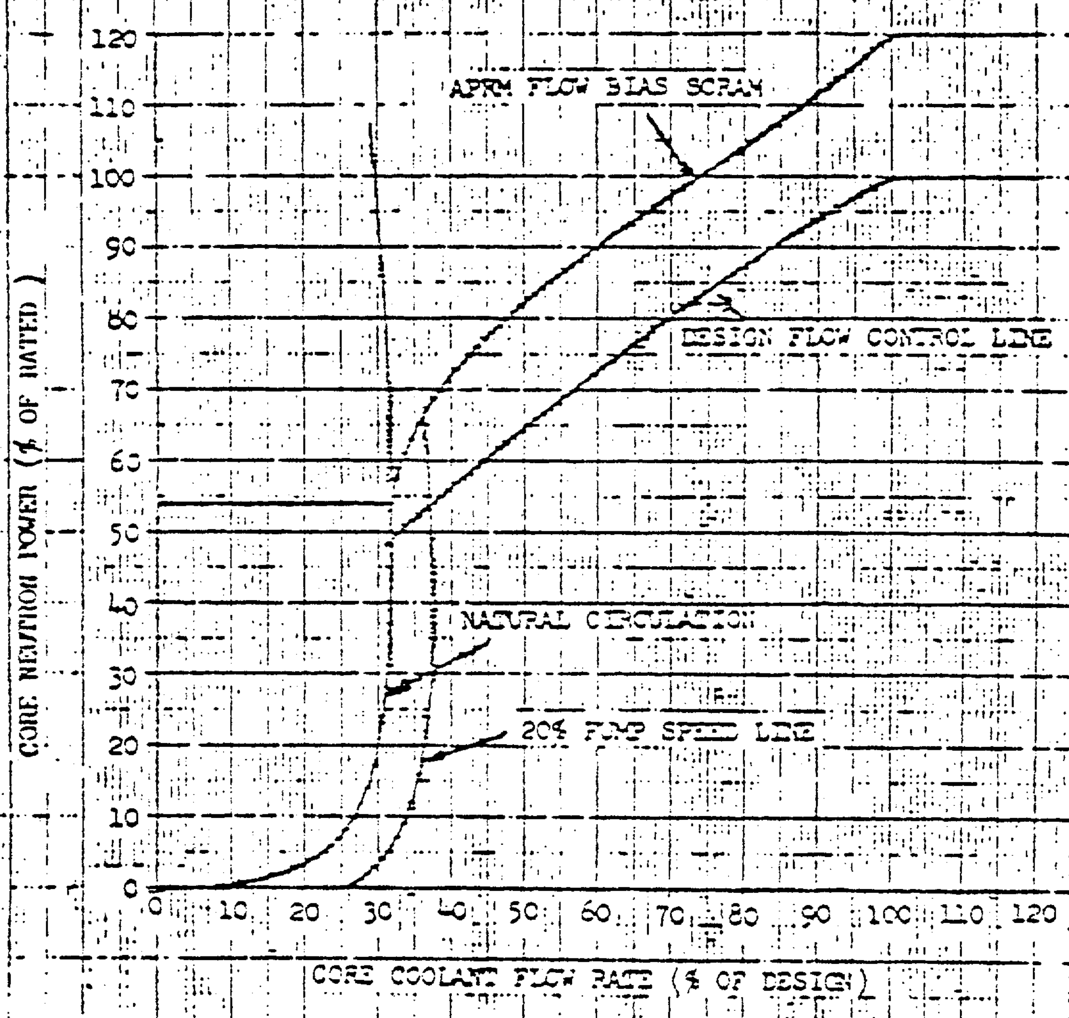
I. J. & K. Reactor low water level set point for initiation of HPCI and RCIC, closing main steam isolation valves, and starting LPCI and core spray pumps.

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-20411-P-A, and Addenda.



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

SAFETY LIMIT

1.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

Applies to limits on reactor coolant system pressure

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification

A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

LIMITING SAFETY SYSTEM SETTING

2.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specification

The limiting safety system settings shall be as specified below:

Protective Action	Limiting Safety System Setting
A. Nuclear system safety valves open--nuclear system pressure	1250 psig ± 13 psi (2 valves)
B. Nuclear system relief valves open--nuclear system pressure	1105 psig ± 11 psi (4 valves)
	1115 psig ± 11 psi (4 valves)
	1125 psig ± 11 psi (3 valves)
C. Scram--nuclear system high pressure	≤ 1,055 psig

1.2 BASES:

REACTOR COOLANT SYSTEM INTEGRITY

The safety limits for the reactor coolant system pressure have been selected such that they are below pressures at which it can be shown that the integrity of the system is not endangered. However, the pressure safety limits are set high enough such that no foreseeable circumstances can cause the system pressure to rise over these limits. The pressure safety limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The design pressure (1,250 psig) of the reactor vessel is established such that, when the 10-percent allowance (125 psi) allowed by the ASME Boiler and Pressure Vessel Code Section III for pressure transients is added to the design pressure, a transient pressure limit of 1,375 psig is established.

Correspondingly, the design pressure (1,148 psig for suction and 1,326 psig for discharge) of the reactor recirculation system piping are such that when the 20-percent allowance (230 and 265 psi) allowed by USAS Piping Code, Section B31.1 for pressure transients are added to the design pressures, transient pressure limits of 1,378 and 1,591 psig are established. Thus, the pressure safety limit applicable to power operation is established at 1,375 psig (the lowest transient overpressure allowed by the pertinent codes), ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The current cycle's safety analysis concerning the most severe abnormal operational transient resulting directly in a reactor coolant system pressure increase is given in the supplemental reload licensing submittal for the current cycle. The reactor vessel pressure code limit of 1,375 psig given in subsection 4.2 of the safety analysis report is well above the peak pressure produced by the overpressure transient described above. Thus, the pressure safety limit applicable to power operation is well above the peak pressure that can result due to reasonably expected overpressure transients.

Higher design pressures have been established for piping within the reactor coolant system than for the reactor vessel. These increased design pressures create a consistent design which assures that, if the the pressure within the reactor vessel does not exceed 1,375 psig, the pressures within the piping cannot exceed their respective transient pressure limits due to static and pump heads.

The safety limit of 1,375 psig actually applies to any point in the reactor vessel; however, because of the static water head, the highest pressure point will occur at the bottom of the vessel. Because the pressure is not monitored at this point, it cannot be directly determined if this safety limit has been violated. Also, because of the potentially varying head level and flow pressure drops, an equivalent pressure cannot be apriori determined for a

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)

2.2 BASES:

REACTOR COOLANT SYSTEM INTEGRITY

The pressure relief system for each unit at the Browns Ferry Nuclear Plant has been sized to meet two design bases. First, the total safety/relief valve capacity has been established to meet the overpressure protection criteria of the ASME Code. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet design basis 4.4.4-1 of subsection 4.4 which states that the nuclear system relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance with the ASME Code requirements is presented in subsection 4.4 of the FSAR and the Reactor Vessel Overpressure Protection Summary Technical Report submitted in response to question 4.1 dated December 1, 1971.

To meet the safety design basis, thirteen safety-relief valves have been installed on unit 2 with total capacity of 84.2% 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 one relief valve is inoperable, has adequate margin to the code allowable overpressure limit of 1375 psig. To meet the operational design basis, the total safety-relief capacity of 84.2% of nuclear boiler rated has been divided into 70% relief (11 valves) and 14.2% safety (2 valves). The analysis of the limiting plant isolation transient is presented in the supplemental reload licensing submittal for the current cycle. This analysis shows that 10 of 11 relief valves limit pressure at the safety valves to a value which is below the setting of the safety valves. Therefore, the safety valves will not open. This analysis shows that peak system pressure is limited to a value which is well below the allowed vessel overpressure of 1375 psig.

Amendment No. 36, 42, 58

TABLE 3.2.F

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	H_2M - 76 - 94 H_2M - 76 - 104	Drywell and Torus Hydrogen Concentration	0.1 - 20%	(1)
2	PdI-64-137 PdI-64-138	Drywell to Suppression Chamber Differential pressure	Indicator 0 to 2 psid	(1) (2) (5)

NOTES FOR TABLE 3.2.F

(1) From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

(2) From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made operable.

(3) If the requirements of notes (1) and (2) cannot be met, either the requirements of 3.5.H shall be complied with or an orderly shutdown shall be initiated and the reactor shall be in a Cold Condition within 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

(5) If the requirements of notes (1) and (2) cannot be met, and if one of the indications cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown within 24 hours.

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 per level operation. Two RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from 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, (i.e., MCPR given by Specification 3.5.k or LHGR of 18.5 kw/ft for 7 x 7 or 13.4 for 8 x 8, 8 x 8R & P8 x 8R fuel). 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: i.e., 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 occurred 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.

3.3/4.3 **BASES:**

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.

.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 of Tables 3.5.I-1,-2,-3,-4, and -5. 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.

$LHGR_{max} \leq LHGR_d [1 - (\Delta P/P)_{max} (L/LT)]$

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

$(\Delta P/P)_{max}$ = 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 & 8x8
= 12.5 ft for 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)

The MCPR operating limit for BFNP 2 cycle 4 is 1.32 for 7X7, 1.27 for 8X8, 8x8R, and P8x8R fuels. 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.

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 state 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 remedial 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 $\geq$ 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.

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.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 will reflect approximately 30 psig for a water level at the high point and 45 psig for a water level in the pressure suppression chamber head tank and are monitored daily to ensure that the discharge lines are filled.

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

3.5.I. 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 10CFR50.46, 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. The analyses supporting these limiting values is presented in reference 4.

Amendment No. ~~35~~, ~~48~~, 58

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 R factor would have to be less than 0.241 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, 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-5

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DRB284 and P8DRB284

AVERAGE PLANAR EXPOSURE (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)
200	11.2	1685
1,000	11.3	1667
5,000	11.8	1671
10,000	12.0	1647
15,000	12.0	1669
20,000	11.8	1672
25,000	11.2	1633
30,000	10.8	1596

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 Safety and Relief Valves

The safety and relief valves are required to be operable above the pressure (105 psig) at which the core spray systems is not designed to deliver full flow. The pressure relief system for each unit at the Browns Ferry Nuclear Plant has been sized to meet two design bases. First, the total safety/relief valve capacity has been established to meet the overpressure protection criteria of the ASME Code. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet design basis 4.4.4-1 of subsection 4.4 which states that the nuclear system relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance, as modified by Reference 4, with the ASME Code requirements is presented in subsection 4.4 of the FSAR and the Reactor Vessel Overpressure Protection Summary Technical Report submitted in Amendment 22 in response to question 4.1 dated December 6, 1971

To meet the safety design basis, thirteen safety-relief valves have been installed on unit 2 with total capacity of 84.2% 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 one relief valve is inoperable, has adequate margin to the code allowable overpressure limit of 1375 psig. To meet the operational design basis, the total safety-relief capacity of 84.2% of nuclear boiler rated has been divided into 70% relief (11 valves) and 14.2% safety (2 valves). The analysis of the limiting plant isolation transient is presented in the supplemental reload licensing submittal for the current cycle. This analysis shows that 10 of 11 relief valves limit pressure at the safety valves to a value which is below the setting of the safety valves. Therefore, the safety valves will not open. This analysis shows that peak system pressure is limited to a value which is well below the allowed vessel overpressure of 1375 psig.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief and safety valves are benchtested every second operating cycle to ensure that their 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.

3.7 CONTAINMENT SYSTEMS

H. Containment Atmosphere Monitoring (CAM) System - H_2 Analyzer

1. Whenever the reactor is not in cold shutdown, two independent gas analyzer systems shall be operable for monitoring the drywell and the torus.

2. With one hydrogen analyzer inoperable, restore at least two hydrogen analyzers to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 24 hours.

3. With no hydrogen analyzer OPERABLE the reactor shall be in HOT SHUTDOWN within 24 hours.

4.7 CONTAINMENT SYSTEMS

H. Containment Atmosphere Monitoring (CAM) System - H_2 Analyzer

1. Each hydrogen analyzer system shall be demonstrated OPERABLE at least once per quarter by performing a CHANNEL CALIBRATION using standard gas samples containing a nominal eight volume percent hydrogen balance nitrogen.

2. Each hydrogen analyzer system shall be demonstrated OPERABLE by performing a CHANNEL FUNCTIONAL TEST monthly.

TABLE 3.7.A
PRIMARY CONTAINMENT ISOLATION VALVES

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
1	Main steamline isolation valves (FCV-1-14,26,37,&51;1-15, 27, 38, & 52)	4	4	$3 < T < 5$	O	GC
1	Main steamline drain isolation valves FCV-1-55 & 1-56	1	1	15	C	SC
1	Reactor Water sample line isolation valves	1	1	5	C	SC
2	RHRS shutdown cooling supply isolation valves FCV-74-48 & 47	1	1	40	C	SC
2	RHRS - LPCI to reactor FCV-74-53, 67		2	30	C	SC
2	Reactor vessel head spray isolation valves FCV-74-77, 78	1	1	30	C	SC
2	RHRS flush and drain vent to suppression chamber FCV-74-102, 103, 119, & 120		4	20	C	SC
2	Suppression Chamber Drain FCV-74-57, 58		2	15	C	SC
2	Drywell equipment drain discharge isolation valves FCV-77-15A, & 15B		2	15	O	GC
2	Drywell floor drain discharge isolation valves FCV-77-2A & 2B		2	15	O	GC

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed ("Torus Support System and Attached Piping Analysis for the Browns Ferry Nuclear Plant Units 1, 2, and 3," dated September 9, 1976 and supplemented October 12, 1976) which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure of 1.3 psid and a suppression chamber water level corresponding to a downcomer submergence range of 4.0 feet to 4.60 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

Inerting

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent or so) reaction of the zirconium and steam during a loss-of-coolant accident could lead to the liberation of hydrogen combined with an air atmosphere to result in a flammable concentration in the containment. If a sufficient amount of hydrogen is generated and oxygen is available in stoichiometric quantities the subsequent ignition of the hydrogen in rapid recombination rate could lead to failure of the containment to maintain a low leakage integrity. The <4% hydrogen concentration minimizes the possibility of hydrogen combustion following a loss-of-coolant accident.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable then the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

To ensure that the hydrogen concentration is maintained less than 4% following an accident, liquid nitrogen is maintained on-site for containment atmosphere dilution. About 2260 gallons would be sufficient as a 7-day supply, and replenishment facilities can deliver liquid nitrogen to the site within one day; therefore, a requirement of 2500 gallons is conservative. Following a loss of coolant accident the Containment Air Monitoring (CAM) System continuously monitors the hydrogen concentration of the containment volume. Two independent systems (a system consists of one hydrogen sensing circuit) are installed in the drywell and the torus. Each sensor and associated circuit is periodically checked by a calibration gas to verify operation. Failure of one system does not reduce the ability to monitor system atmosphere as a second independent and redundant system will still be operable.

In terms of separability, redundancy for a failure of the torus system is based upon at least one operable drywell system. The drywell hydrogen concentration can be used to limit the torus hydrogen concentration during post LOCA conditions. Post LOCA calculations show that the CAD system initiated within two hours at a flow rate of 100 scfm will limit the peak drywell and wetwell hydrogen concentration to 3.6% (at 4 hours) and 3.8% (at 32 hours), respectively. This is based upon purge initiation after 20 hours at a flow rate of 100 scfm to maintain containment pressure below 30 psig. Thus, peak torus hydrogen concentration can be controlled below 4.0 percent using either the direct torus hydrogen monitoring system or the drywell hydrogen monitoring system with appropriate conservatism ($\leq 3.8\%$), as a guide for CAD/Purge operations.

4. Daily tests of annunciation lights and audible devices are performed as a routine operation function.

5. The CO_2 system manufacturer recommends semiannual testing of CO_2 system fire detection circuits.

Figure 6.3-1 describes the in-plant fire protection organization including the roving fire watch. In addition, other operating personnel periodically inspect the plant during their normal operating activities for fire hazards and other abnormal conditions.

Smoke detectors will be tested "in-place" using inert freon gas applied by a pyrotronics type applicator which is accepted throughout the industrial fire protection industry for testing products of combustion detectors or by use of the MSA chemical smoke generators. At the present time the manufacturers have only approved the use of "punk" for creating smoke. TVA will not use "punk" for testing smoke detectors.

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

Browns Ferry unit 2 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).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 58 TO FACILITY OPERATING LICENSE NO. DPR-52

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

DOCKET NO. 50-260

1.0 Introduction

By letter dated July 14, 1980 (TVA BFNP TS 140), as supplemented by letters dated August 29, 1980 and October 7, 1980, the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit No. 2 (BF-2). The proposed Technical Specifications would incorporate the limiting conditions for operation of the facility in the fourth fuel cycle following the third refueling of the reactor. In support of this reload application for BF-2, the licensee has submitted a reload licensing document(1) prepared by the General Electric Company (GE), additional information related to the refueling outage(2) and proposed changes to the Technical Specifications(2,3).

2.0 Discussion

BF-2 shutdown for its third refueling on September 5, 1980. BF-2 was initially fueled with 764 of the GE 7x7 fuel assemblies containing 49 fuel rods each. During the first refueling, which began March 18, 1978, 132 of the 7x7 fuel elements were replaced with one water rod 8x8 fuel assemblies. In the second refueling, which started April 27, 1979, 232 of the 7x7 fuel assemblies were replaced with a like number of two water rod, retrofit 8x8 (8x8R) bundles. During the second refueling, an additional 36 7x7 fuel assemblies were also replaced with 8x8 fuel that had originally been procured for fuel cycle 2 but not used. During this third refueling, an additional 240 of the original 7x7 fuel bundles are being replaced with prepressurized two water rod 8x8 retrofit (P8x8R) fuel assemblies. The prepressurized fuel assemblies are essentially identical from a core physics standpoint to the two water rod fuel assemblies (8x8R) except that they are prepressurized with about three rather than one atmospheres of helium to minimize fuel clad interaction. Our evaluation of the P8x8 fuel is discussed in the safety evaluation attached to our letter of April 16, 1979, to GE approving the use of this fuel in Boiling Water Reactor (BWR) reload licensing applications. The

8012080 301

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 (LHGRs) and are therefore expected to yield performance benefits in terms of fuel reliability. The increased prepressurization also results in improved margin to maximum average planar LHGR (MAPLHGR) limits by reducing stored energy. This will be the first use of the P8x8R fuel in BF-2. This fuel is being used in the current fuel cycles for Units 1 and 3. The first use of P8x8R fuel in a Browns Ferry unit was approved for the second reload of Unit 3 (Amendment No. 28 to Facility Operating License No. DPR-58 dated November 30, 1979). We subsequently approved the use of P8x8R fuel in Unit 1 by Amendment No. 59 to Facility Operating License No. DPR-33 dated February 25, 1980. Thus, there is operating experience with this fuel in two of the Browns Ferry units as well as a number of other BWRs.

With this refueling, BF-2 will be on a nominal 18-month refueling cycle. Units 1 and 3 are also on 18-month refueling cycles.

As noted above, this reload involves loading of prepressurized GE 8x8 retrofit (P8x8R) fuel. The description of the nuclear and mechanical designs of P8x8 fuel is contained in Reference 4. The use and safety implications of prepressurized fuel are presented in Reference 4 and have been found acceptable per Reference 5 (enclosed in Appendix C of Reference 4).

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 4. Additional plant and cycle dependent information is provided in the reload application[3] which closely follows the outline of Appendix A of Reference 4. Reference 5 includes a description of the NRC 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 5.

Our safety evaluation of the GE generic reload licensing topical report has also concluded that the nuclear, and mechanical design of the 8x8R and P8x8R fuels, and GE's analytical methods for nuclear and thermal-hydraulic calculations as applied to mixed cores containing 7x7, 8x8, 8x8R and P8x8R 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 8x8R and P8x8R fuels in mixed loadings, and on the basis of the evaluations which have been presented in Reference 4, only a limited number of additional areas of review have been included in this safety evaluation. For evaluations of areas not specifically addressed in this safety evaluation refer to Reference 4.

During this outage, TVA had proposed to accomplish major modifications of the BF-2 electrical systems. These are described in TVA's submittal and application of August 6, 1980 (BFNP-TS-143). However, several critical items of equipment, including the common station service transformers and MG sets, were not delivered in time to be installed during this outage. Also, partly as a result of our review, TVA is reevaluating the design of the system. Some of the electrical modifications for BF-2 are scheduled to be completed during the Unit 1 refueling outage (April to June 1981); however; the overall modifications are not scheduled to be completed until the spring 1982 refueling outage.

One of the modifications which TVA is accomplishing during this refueling outage and which is discussed herein is a replacement of the primary containment hydrogen monitoring system. A description of the new hydrogen monitoring system and proposed changes to the Technical Specifications were submitted by TVA's letter of August 29, 1980(2). In response to our concerns, additional information was submitted in TVA's letter of October 7, 1980(6). Like most BWRs, Browns Ferry operates with an inerted (nitrogen) containment. The present Containment Atmosphere Monitoring System (CAM) at Browns Ferry consists of hydrogen monitors located inside the containment. Four sensors are located in the drywell and two are in the torus. Only two sensors in the drywell and one in the torus are in active service at any time with the remaining sensors acting as backup. The purpose of the sensors is to monitor hydrogen concentration in the containment post-LOCA (Loss of Coolant Accident) to provide guidance for use of the Containment Atmosphere Dilution System (CAD). As noted above, post-accident hydrogen control is provided by inerting the primary containment during normal operation. After a postulated accident, long-term combustible gas concentrations are controlled by the CAD system. This system is designed to purge small quantities through a 2" line to the standby gas treatment system while adding makeup nitrogen to the containment. The present CAD system meets NRC requirements on redundancy, single-failure criteria and the TMI-2 lessons learned requirement for dedicated hydrogen control penetrations.

The present hydrogen monitoring analyzers were supplied by GE. The present sensors are specified as having an accuracy of ± 2 percent of scale with a range of 0 to 20 percent hydrogen concentration. The instruments were qualification tested to a radiation dose of approximately 3.2×10^7 RAD. The hydrogen sensors have been integrity qualified up to 340°F. The instrument accuracy is not guaranteed to be within specifications above 200°F; however, this is not a significant shortcoming since calculated drywell temperature returns to 175°F within 15 minutes of the LOCA and calculated torus temperature is less than 130°F following a postulated LOCA. Although the present hydrogen monitors are environmentally qualified and have apparently performed adequately, as part of the TMI-2 Lessons Learned program, TVA prepared a design change to move the sensors outside the containment to improve the maintainability during unit operation (access considerations) and to provide post accident sampling capability for other gases.

During the current refueling outage of Unit 2 and upcoming refueling outages of Unit 3 (Nov.-Dec. 1980) and Unit 1 (April-May 1981), TVA is replacing the GE hydrogen monitoring system with a new Hayes-Republic System located outside containment. Each reactor will be equipped with two totally independent systems for monitoring hydrogen concentrations in the drywell and torus. Each system includes a thermal conductivity gas analyzer, sample pumps, chillers to remove entrained moisture from the gas stream and associated valves and controls, all mounted in a cabinet external to the primary containment. Gas samples are withdrawn from the upper part of each drywell and torus through existing penetrations to a sampling cabinet outside primary containment. The sample will pass through about 100 feet of 1/2 inch stainless steel pipe, a water trap and chiller to remove entrained moisture, a bellows pump and either of two independent thermal conductivity sensors and will be exhausted back into the drywell.

After the system is activated, the sample will reach the sensor in less than 2 minutes. The sensor will begin to respond in 3 seconds and will reach two-thirds of its steady reading in 21 seconds. The sensitivity of reading is ± 0.4 volume percent hydrogen (i.e., ± 2% of the 20% full scale hydrogen concentration the instrument is designed to measure). This accuracy is the same as the present GE hydrogen monitoring system.

3.0 Evaluation

3.1 Core Reload

3.1.1 Nuclear Characteristics

For cycle 4 operation, 240 fresh P8x8R fuel bundles of type P8DRB284 will be loaded into the core(1). The remainder of the 764 fuel bundles in the core will be previously irradiated bundles. Based on the data provided in Reference 1 both the control rod system and the standby liquid control system will have acceptable shutdown capability during cycle 4.

3.1.2.1 Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 4, for BWR cores which reload with GE's retrofit 8x8 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 4.

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 4. The staff evaluation, included as Appendix C of Reference 4, 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 1. Our evaluation of the methods used to develop these supplementary input values is also included in Appendix C of Reference 4.

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 1 were assumed.

The results of these analyses are outlined in Reference 1 sections 9 and 10. On this topic, it is acceptable if fuel specific operating limits are established for prepressurized fuel (Appendix C, Reference 4). 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.

As noted above, the calculated system responses and reductions in CPR during each of the operational transients have been provided in Sections 9 and 10 of the GE Supplemental Reload Licensing Submittal (Reference 1). It is interesting to note that for this reload, the local rod withdrawal error (with limiting instrument failure) dictates the operating limit MCPR (OLMCPR) for all fuel types. The ΔCPR calculated for the load rejection without bypass was always the controlling transient for 8x8R fuel in past reloads; in this reload, the ΔCPR for this event is the same (0.20) as for the rod withdrawal error.

The following table gives the limiting CPR reduction as calculated by GE, the event for which limiting CPR reduction occurs, and the required operating limit MCPR for each fuel type:

Fuel Type	Most Severe CPR Reduction	Operating Limit MCPR
7x7	0.25 (Control Rod Withdrawal)	1.32
8x8	0.20 (Control Rod Withdrawal)	1.27
8x8R	0.20 (Load Rejection Without Bypass) (Control Rod Withdrawal)	1.27
P8x8R	0.20 (Control Rod Withdrawal) (Load Rejection Without Bypass)	1.27

Thus, when the reactor is operated in accordance with the above operating limit MCPRs the 1.07 SLMCPR will not be violated in the event of the most severe abnormal operational transient. This is acceptable to the staff per the finding of the previous section. On this basis, operating limit MCPR Technical Specifications have been established.

3.1.3 Accident Analyses

3.1.3.1 ECCS Appendix K Analysis

In our safety evaluation of Reference 4, we concluded that "the continued application of the present GE ECCS-LOCA ("Appendix K") models to the 8x8 retrofit reload fuel is generally acceptable and in our Reference 5 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 4. 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 presented in Section 16 of Reference 1. The calculated resultant peak enthalpy was 131 cal/gm, 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 reported in Section 15 of the supplemental reload submittal(1). The analyses determined that a rotated P8x8R fresh fuel assembly was the most limiting loading error event; the ΔCPR for this event was 0.13. This is considerably less than the ΔCPR for the limiting transients. As shown in the table in Section 3.1.2.2.2, above, the ΔCPR for the limiting transients -- which determines the OLMCPR -- is 0.20 for all 8x8 fuel and 0.25 for 7x7 fuel. TVA has revised the core verification procedures and has committed to revise the fuel handling procedures with the objective of preventing the recent fuel loading errors that have occurred at Browns Ferry 1 and 2.

3.1.3.4 Overpressure Analysis

For Cycle 4, the licensee has reanalyzed the limiting pressurization event to demonstrate that the ASME Boiler and Pressure Vessel Code requirements are met for BF-2. The methods used for this analysis, when modified to account for one failed safety valve, have also been previously approved by the staff. The acceptance criteria for this event is that the calculated peak transient pressure not exceed 110% of design pressure, i.e., 1375 psig. The reanalysis, which is presented in Section 12 of the supplemental reload submittal[1], shows that the peak pressure at the bottom of the reactor vessel does not exceed 1301 psig for worst case end-of-cycle conditions, even when assuming the effects of one failed safety valve. This is an increase of 2 psig from the previous fuel cycle and is the reason for the changes on pages 30 and 219 of the proposed Technical Specifications. We conclude that there is sufficient margin between the peak calculated vessel pressure and the design limit pressure to allow for the failure of at least one valve. Therefore, the limiting overpressure event as analyzed by the licensee is considered acceptable on the bases outlined in Reference 4.

3.1.4 Thermal Hydraulic Stability

The results of the thermal hydraulic stability analysis presented in Section 13 of Reference 1 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 Hydrogen Monitoring System

We have evaluated the proposed hydrogen monitoring system against the criteria listed in the Standard Review Plan 6.2.5, "Combustible Gas Control in Containment," in Regulatory Guide 1.97, Revision 1, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," and in the letter to Licensees of September 5, 1980, entitled, "Preliminary Clarification of TMI Action Plan Requirements," Attachment 6 (II.F.1).

The proposed thermal conductivity method and equipment has adequate sensitivity and time response, and is at least as reliable as the currently used hydrogen electrode method. The relocation of the sensors outside primary containment makes them more accessible to maintenance and recalibration under LOCA conditions. No additional penetrations of primary containment will be required since the new sampling lines will pass through unused but existing spare penetrations.

The power circuits to operate the hydrogen monitoring system will meet the safety requirements of engineered safety features. The redundancy requirements will be met by having two independent hydrogen sensors to which gas samples from either the drywell or the torus atmosphere may be directed.

We have evaluated the information provided by TVA in their letter of August 29, 1980 and in the telephone conversation of September 30, 1980, and have concluded that the proposed changes in the hydrogen monitoring systems are acceptable and meet the requirements of General Design Criterion 41 (Containment Atmosphere Cleanup) of Appendix A to 10 CFR Part 50.

3.3 Technical Specification Modifications

Included in References 2 and 3 are several proposed modifications to the BF-2 Technical Specifications. Of these, changes to the overpressurization limit, revision of operating limit MCPRs, and adoption of MAPLHGR limits for P8x8R fuel are acceptable for reasons discussed in earlier sections of this report. The changes associated with the new hydrogen monitoring system have also been discussed and are acceptable.

Some of the modifications proposed in References 3 and 4 are merely changes in the bases to the Technical Specifications, reflecting the refueling of the core, and do not constitute changes in the limiting conditions for operation. We consider the proposals of this type to be acceptable.

Because this is the first cycle for which BF-2 will contain P8x8R fuel, a linear heat generation rate (LHGR) limit and a power spiking penalty associated with the LHGR limit have been proposed. These proposals are consistent with the requirements of our generic SER and are therefore acceptable for BF-2 for Cycle 4.

4.0 Environmental Considerations

We have determined that the amendment does 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 amendment involves 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 this amendment.

5.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 12, 1980

References

1. "Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Power Station Unit No. 2," Y1003J01A12, June 1980.

2. Letter from L. M. Mills of TVA to Harold R. Denton, NRC dated August 29, 1980 (TVA BFNP TS 140).

3. Letter from L. M. Mills of TVA to Harold R. Denton, NRC, dated July 14, 1980 (TVA BFNP TS 140).

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

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

6. Letter from L. M. Mills of TVA to Harold R. Denton, NRC, dated October 7, 1980 (TVA BFNP TS 140).

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-260

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 58 to Facility Operating License No. DPR-52 issued to Tennessee Valley Authority (the licensee), which revised the Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit No. 2 (the facility) located in Limestone County, Alabama. The amendment is effective as of the date of issuance.

This amendment permits operation of Browns Ferry Unit No. 2 with pre-pressurized 8x8 retrofit fuel in the fourth fuel cycle following the third refueling outage.

The application for this amendment 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 amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendment dated July 14, 1980, as supplemented by letters dated August 29, 1980 and October 7, 1980, (2) Amendment No. 58 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, NW., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 12th day of November, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

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