Docket

January 12, 1981

Docket No. 50-296

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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 No. 37 to Facility License No. DPR-68 for the Browns Ferry Nuclear Plant, Unit No. 3. This amendment changes the Technical Specifications in response to your request of August 27, 1980 (TVA BFNP TS 148), as supplemented by your letters of September 23, 1980 and October 14, 1980, and to your requests dated September 5, 1980 and October 17, 1980.

The changes to the Technical Specifications (1) incorporate the limiting conditions for operation during the fourth fuel cycle, (2) reflect new primary containment hydrogen monitoring instrumentation being installed during the current refueling outage and (3) reflect the addition of 480 volt motor generator sets during the refueling outage to supply reactor motor operated valve boards 3D and 3E.

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 #2
Division of Licensing

Enclosures:
1. Amendment No. 37 to DPR-68
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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OFFICE	ORB #2	ORB #2	AD:OR	OELD	ORB #2		
SURNAME	SNorris	RClark:mjf	TNovak	JHLaverty	TIppolito		
DATE	1/ /81	1/6/81	1/8/81	1/12/81 no legal objections	1/6/81		

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

January 12, 1981

Docket No. 50-296

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

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Sincerely,

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

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cc w/enclosures:
See next page

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 37
License No. DPR-68

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

   A. The applications for amendment by Tennessee Valley Authority (the licensee) dated August 27, 1980, (as supplemented by letters dated September 23, 1980 and October 14, 1980,) and September 5, 1980 and October 17, 1980, comply 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 applications, 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-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 37, 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 12, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 37

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

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

17	177
24	178
26	225
27	225a
28	261
29	286A
30	286B (new page)
82	320
136	321
166	325a
167	326
176	328

2. Marginal lines on each page indicate the revised area.

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.

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.

position, where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure. With the scrams set at 10 percent of valve closure, neutron flux does not increase.

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

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 N14 of the FSAR demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

L. <u>References</u>

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.

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.

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 limts 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	1,250 psig ± 13 psi (2 valves)
B. Nuclear system relief valves open--nuclear system pressure	1,105 psig ± 11 psi (4 valves) 1,115 psig ± 11 psi (4 valves)

Amendment No. 28,37

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
1.2 REACTOR COOLANT SYSTEM INTEGRITY	2.2 REACTOR COOLANT SYSTEM INTEGRITY
	1,125 psiq ± 11 psi (3 valves)
	C. Scram--nuclear system high pressure — ≤ 1,055 psig

Amendment No. 28, 37

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 ressulting 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 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 a priori determined for a 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 N14.0)
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 each unit with a 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, 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 plant isolation transient (turbine trip with bypass valve failure to open) assuming a turbine trip scram is presented in the supplemental reload licensing submittal for the current cycle. This analysis shows that the 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.

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)

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 to 0 percent rod density groups. In addition, RSCS will prevent movement of rods in the 50 percent density to a preset power level range until the scrammed rod has been withdrawn.

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 limit is considered safe since an insertion of the 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 CORE AND CONTAINMENT COOLING SYSTEMS

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 13.4 kW/ft.

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

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

J. Linear Heat Generation Rate (LHGR)

The LHGR shall be checked daily during reactor operation at ≥25% rated thermal power.

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

and corresponding action shall continue until reactor operation is within the prescribed limits.

K. Minimum Critical Power Ratio (MCPR)

The MCPR operating limit is 1.24 for 8x8 fuel, and 1.25 for 8x8R fuel, and 1.25 for 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. 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.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

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.

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.

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

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

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all tne rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting value for MAPLHGR is shown in Tables 3.5.I-1, -2, -3. The analyses supporting these limiting values is presented in reference 4.

J. Linear Heat Generation Rate (LHGR)

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

The LHGR shall be checked daily during reactor operation at ≥ 25% 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.

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 thermal limit.

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 circumtances, this will not be the only alternative.

M. References

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

3.6/4.6 BASES

To meet the safety design basis, thirteen safety-relief valves have been installed on unit 3 with a 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 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 plant isolation transient (turbine trip with bypass valve failure to open) assuming a turbine trip scram is presented in the supplemental reload licensing submittal for the current cycle. This analysis shows that the 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.

3.6/4.6 BASES

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)

LIMITING CONDITIONS FOR OPERATION

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.

SURVEILLANCE REQUIREMENTS

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.

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 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 than 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 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 within two hours at a flow rate of 100 scfm will limit the peak drywell

Inerting (Cont'd)

and wetwell hydrogen concentration to 3.9% (at 3 hours) and 3.9% (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 ($\leq$ 3.9%), as a guide for CAD/Purge operations.

3.9 AUXILIARY ELECTRICAL SYSTEM

b. The fourth operable unit 3 diesel generator.

4. Buses and Boards Available

a. Both start buses to unit 3 are energized.

b. The 4-kV bus tie board and shutdown boards (3EA, 3EB, 3EC, 3ED) are energized.

c. The 480-V shutdown boards associated with the unit are energized.

d. Undervoltage relays operable on start buses 1A or 1B and 4-kV shutdown boards, 3EA, 3EB, 3EC, and 3ED.

e. The 480V diesel Aux Boards are energized.

f. The 480V Rx. MOV Boards D & E are energized with M-G Sets 3DN, 3DA, 3EN, and 3EA in service.

4.9 AUXILIARY ELECTRICAL SYSTEM

4. Undervoltage Relays

a. Once every 6 months, the condition under which the undervoltage relays are required shall be simulated with an undervoltage on start buses 1A and 1B to demonstrate that the diesel generators will start.

b. Once every 6 months, the conditions under which the undervoltage relays are required shall be simulated with an undervoltage on each shutdown board to demonstrate that the associated diesel generator will start.

3.9 AUXILIARY ELECTRICAL SYSTEM

5. The 250-Volt Shutdown Board battery and unit batteries and a battery charger for each battery and associated battery boards are operable.

6. Logic Systems

a. Accident signal logic system is operable.

7. There shall be a minimum of 103,300 gallons of diesel fuel in the unit 3 standby diesel generator fuel tanks.

4.9 AUXILIARY ELECTRICAL SYSTEM

c. The undervoltage relays which start the diesel generators from start buses 1A and 1B and the 4-kV shutdown boards, shall be calibrated annually for trip and reset and the measurements logged.

5. 480-V RMOV boards D and E

a. Once per operating cycle, the automatic transfer feature for 480-V RMOV boards D and E shall be functionally tested to verify auto-transfer capabi.

3.9 AUXILIARY ELECTRICAL SYSTEM

8. Undervoltage relays on 1A or 1B start bus may be inoperable for a period of 7 days provided the other start bus and undervoltage relay are operable (within surveillance schedule or 4.9.A.4.a).

9. Undervoltage relays on a shutdown board may be inoperable 5 days provided the other shutdown boards and undervoltage relays are operable (within surveillance schedule of 4.9.A.4.b).

10. When one 480 volt shutdown board is found to be inoperable, the reactor will be placed in hot standby within 12 hours and cold shutdown within 24 hours.

11. If one 480-V RMOV board m-g set is inoperable, the reactor may remain in operation for a period not to exceed seven days, provided the remaining 480-V RMOV board m-g sets and their associated loads remain operable.

12. If any two 480-V RMOV board m-g sets become inoperable, the reactor shall be placed in the cold shutdown condition within 24 hours.

13. If the requirements for operating in the conditions specified by 3.9.B.1 through 3.9.B.12 cannot be met, an orderly shutdown shall be initiated and the reactor shall be shutdown and in the cold condition within 24 hours.

LIMITING CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

C. Operation in Cold Shutdown Condition

Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor, the availability of electric power shall be as specified in Section 3.9.A except as specified herein.

1. At least two unit 3 diesel generators and their associated 4-kV shutdown boards shall be operable.

2. An additional source of power consisting of one of the following:

   a. One 161-kV transmission line and its associated cooling tower transformer capable of supplying power to the unit 3 shutdown boards.

   A third operable diesel generator.

3. At least one unit 3 480-V shutdown board must be operable.

4. A total of two 480-V RMOV board motor-generator (m-g) sets may be inoperable. One loop of the RHR system (LPCI mode) must remain fully operable at all times. (One m-g set for 480-V RMOV boards D and E must be in service at all times. The two operable m-g sets may not be supplied from the same 480-V shutdown board.)

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

The 250-Volt d-c system is so arranged, and the batteries sized such, that the loss of any one unit battery will not prevent the safe shutdown and cooldown of all three units in the event of the loss of offsite power and a design basis accident in any one unit. Loss of control power to any engineered safeguard control circuit is annunciated in the main control room of the unit affected.

The station battery supplies loads that are not essential for safe shutdown and cooldown of the nuclear system. This battery was not considered in the accident load calculations.

There are two 480-V ac Reactor Motor-Operated Valve (RMOV) Boards that contain motor-generator (M-G) sets in their feeder lines. These 480-V ac RMOV boards have an automatic transfer from their normal to alternate power source (480-V ac shutdown boards). The M-G sets act as electrical isolators to prevent a fault from propagating between electrical divisions due to an automatic transfer. The 480-V ac RMOV boards involved provide motive power to valves associated with the LPCI mode of the RHR system. Having an M-G set out of service reduces the assurance that full RHR (LPCI) capacity will be available when required. Since sufficient equipment is available to maintain the minimum complement required for RHR (LPCI) operation, a 7-day servicing period is justified. Having two M-G sets out of service can considerably reduce equipment availability. Therefore, the affected unit shall be placed in cold shutdown within 24 hours.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 37 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

DOCKET NO. 50-296

1.0 Introduction

By letter dated August 27, 1980 (TVA BFNP TS 148), which was supplemented by letters dated September 23, 1980 and October 14, 1980, and by letters dated Setpember 5, 1980 and October 17, 1980, the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-68 for the Browns Ferry Nuclear Plant, Unit No. 3. The proposed amendments and revised Technical Specifications were to: (1) incorporate the limiting conditions for operation associated with the fourth fuel cycle, (2) reflect new primary containment atmospheric hydrogen monitoring instrumentation being installed during the current refueling outage, and (3) reflect the addition of 480 volt motor generator sets during the refueling outage to supply reactor motor operated valve (RMOV) boards 3D and 3E.

2.0 Discussion

Browns Ferry Unit No. 3 (BF-3) shutdown for its third refueling on November 23, 1980. The initial core loading for BF-3 consisted of 764 of the single water rod 8X8 fuel assemblies, each containing 63 fuel rods. During the first refueling in September 1978, 208 of the fuel assemblies were replaced with 8X8R fuel assemblies containing 62 fuel rods in each. During the second refueling outage starting in August 1979, an additional 144 of the initial fuel bundles were replaced with P8X8 fuel assemblies, each containing 62 fuel rods. During the current refueling outage, an additional 124 of the original 8X8 fuel assemblies are being replaced with a like number of new P8X8R fuel assemblies. The prepressurized fuel assemblies (P8X8R) 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 P8X8R 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-3, TVA submitted by letter (1) dated August 27, 1980, a supplemental reload licensing document(2) prepared by the General Electric Company (G.E.) for TVA and proposed changes to the BF-3 Technical Specifications(3). This initial submittal was supplemented by a letter (4) dated September 23, 1980 relating to the GEXL critical power correlation and a letter(5) dated October 14, 1980 submitted additional proposed changes to the Technical Specifications(6) to remove references to the power spiking-penalty in the Linear Heat Generation Rate (LHGR) calculations.

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 September 5, 1980(7). The new hydrogen monitoring system being installed in BF-3 is the same as the new hydrogen monitoring system which was installed in Browns Ferry Unit No. 2 (BF-2) during the September to November 1980 refueling outage. Use of the new system for BF-2 was approved as part of the reload amendment - Amendment No. 58 to Facility Operating License No. DPR-52 dated November 12, 1980. The present and new hydrogen monitoring systems were described in detail in Amendment No. 58 for BF-2 and this descriptive material is incorporated herein by reference.

Another modification which TVA is planning to accomplish during the present refueling outage of BF-3 is to add four 480-volt motor generator (MG) sets to supply reactor motor operated valve boards 3D and 3E. 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 complete the modifications for BF-3 by the end of the second refueling outage and to submit proposed Technical Specification changes with the reload amendment request for each unit. For BF-3, the modifications consisted of the following:

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 recirculation pump discharge valves, LPCI injection valves, RHR pump minimum flow bypass valves, and RHR test isolation valves. The change also modifies independent valve a.c. power supplies, and modified d.c. power supplies to 4kV shutdown board control power to provide adequate independence such that a station battery failure does not jeopardize core cooling capabilities.

During the second refueling outage of BF-3 (August 24 to December 8, 1979) all of the electrical changes related to the LPCI modification were completed except for the addition of the MG sets. Due to a strike at the manufacturer's facility, the MG sets were not delivered in time to be installed during the last refueling outage.

There are two 480-V ac Reactor Motor-Operated Valve (RMOV) Boards that contain motor-generator (M-G) sets in their feeder lines. These 480-V ac RMOV boards have an automatic transfer from their normal to alternate power source (480-V ac shutdown boards). The M-G sets act as electrical isolators to prevent a fault from propagating between electrical divisions due to an automatic transfer. The 480-V ac RMOV boards involved provide motive power to those valves necessary for automatic operation of RHR injection (Recirculation pump discharge valves, LPCI injection valves, RHR pump minimum flow bypass valves and RHR test isolation valves) and will interface with the divisionalized 480 V shutdown boards through the M-G sets. Each RMOV board will have two sets, and although only one M-G set will normally supply power to the RMOV board, both M-G sets will run at all times to assure readiness of the alternate M-G set to accept load if required.

By letter[8] dated October 17, 1980, TVA submitted proposed changes to the Technical Specifications to reflect the addition of the 480 Volt MH sets to specify surveillance and operability requirements for this equipment.

3.0 Reload

This refueling (Reload 3) is the second for BF-3 to incorporate GE's P8X8R fuel design on a batch basis. The description of the nuclear and mechanical design of the Reload 3 P8X8R fuel and the exposed unpressurized 8X8 and 8X8R fuels, used in the initial and first reload cores, is contained in GE's generic licensing topical report for BWR reloads[9]. Reference 9 also contains a complete set of references to topical reports which describe GE's analytical methods for the nuclear, thermal-hydraulic, transient and accident calculations performed for this reload together with information on the applicability of these methods to cores containing a mixture of different fuel designs. Portions of the plant-specific data, such as operating conditions and design parameters, which are used in transient and accident calculations, have also been included in the topical report. The use and safety implications of prepressurized fuel are presented in Reference 9 and have been found acceptable per Reference 10 (Enclosed in Appendix C of Reference 9).

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 9. Additional plant and cycle dependent information is provided in the reload application (Reference 2) which closely follows the outline of Appendix A of Reference 9. Reference 10 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 10.

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 8X8, 8X8R and P8X8R fuels, are acceptable.

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 9, 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 evalution report, the reader is referred to Reference 9.

3.1.1 Nuclear Characteristics

For Cycle 4, 124 fresh pressurized type P8DRB265L fuel bundles will be loaded into the core. The remainder of the fuel bundles in the core will be a combination 8X8, 8X8R and P8X8R fuel bundles exposed during the previous three cycles.

The fresh fuel will be loaded and the previously peripheral fuel will be shuffled inward so as to constitute an octant-symmetric core pattern, which is acceptable.

Based on the data provided in Sections 4 and 5 of Reference 2, both the control rod system and the standby liquid control system will have an 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 9, for BWR cores which reload with GE's P8X8R fuel, the allowable minimum critical power ratio (MCPR) resulting from either core-wide or localized abnormal operational transients is equal to 1.07. When meeting this MCPR safety limit during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The 1.07 safety limit minimum critical power ratio (SLMCPR) to be used for Cycle 4 is unchanged from the SLMCPR previously approved for Cycles 2 and 3. The basis for this safety limit is addressed in Reference 9, while our generic approvals are given in Reference 10.

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 safety limit MCPR 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. These events have been analyzed for both the exposed 8X8, 8X8R, and P8X8R fuel and the fresh P8X8R fuel. Addition of the largest reductions in critical power ratio to the safety limit MCPR establishes the operating limits for each fuel type. The transient events analyzed were load rejection without bypass, feedwater controller failure, loss of 100°F feedwater heating and control rod withdrawal error.

3.1.2.2.1 Abnormal Operational Transient Analysis Methods

The generic methods used for these calculations, including cycle-independent initial conditions and transient input parameters, are described in Reference 9. Our acceptance of the cycle-independent values appears in Reference 10. Additionally, our evaluation of the transient analysis methods, together with a description and summary of the outstanding issues associated with these methods, appears in Reference 10. Supplementary cycle-dependent 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[10] has also addressed the methods used to develop these supplementary input values.

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.

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 2). On this topic, it is acceptable if fuel specific operating limits are established for prepressurized fuel (Appendix C, Reference 9). 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.

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 Redution	Operating Limit MCPR
8x8	0.17 Load Rejection w/o Bypass	1.24
8X8R	0.18 Load Rejection w/o Bypass	1.25
P8X8R	0.18 Load Rejection w/o Bypass	1.25

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.

3.1.2.3 Fuel Cladding Integrity Safety Limit LHGR

The control rod withdrawal error and fuel loading error events were reanalyzed by the licensee to also determine the maximum transient linear heat generation rates (LHGRs). The results for BF-3 Cycle 4 are given in Appendix B of the Supplemental Reload Licensing Submittal[2]. The calculated Fuel Loading Error LHGR is 16.9 kW/ft for a rotated bundle and 18.1 kW/ft for a misplaced bundle. These results indicate that the fuel type-dependent and exposure-dependent safety limit LHGRs, shown in Table 2-3 of Reference 9, will not be violated should these events occur. Thus, fuel failure due to excessive cladding strain will be precluded. We find these results, which adequately account for the effects of fuel densification power spiking, to be acceptable.

3.1.3 Accident Analyses

3.1.3.1 ECCS Appendix K Analysis

In our safety evaluation of Reference 9, 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 10 evaluation we extended that conclusion to prepressurized fuel. On these bases, the MAPLHGR limits, which remain unchanged from the previous cycle, are acceptable.

3.1.3.2 Control Rod Drop Accident

For Cycle 4, the key plant-specific and cycle-specific nuclear characteristics for the worst case control rod drop accident (CRDA) occurring during both cold and hot startup conditions are conservatively bounded by the values used in bounding CRDA analysis given in Reference 9. The results of G.E.'s analysis are presented in Section 16 of the Supplemental Reload Licensing Submittal(2). The bounding analysis, which includes the adverse effects of fuel densification power spiking, shows that the peak enthalpy will not exceed the 280 cal/gm design limit. Therefore, for Cycle 4 of BF-3, the peak fuel enthalpy associated with a CRDA from the hot and cold startup condition will also be within the 280 cal/gm design limit.

Thus, we conclude that the peak enthalpy associated with a control rod drop accident occurring from any in-sequence control rod movement will be below the 280 cal/gm design limit.

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 2 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(2). The analyses determined that a rotated P8X8R fresh fuel assembly was the most limiting loading error event; the ΔCPR for this event was 0.17. This is the same, or almost the same, as the ΔCPR for the limiting transient which determines the safety limit MCPR. As shown in Section 3.1.2.2.2, above, the ΔCPR for the limiting transient is also 0.17 for 8X8 fuel and 0.18 for 8X8R and P8X8R fuel assemblies.

During the recent refueling (September - November 1980) of Browns Ferry Unit 2, it was discovered that two 7X7 fuel assemblies had gone thru cycle 3 misoriented 90° and that there is presently a 7X7 fuel element (core location 11-06) in Browns Ferry Unit 1 that is misoriented 90°. During recent refuelings, there has also been a significant number of misoriented fuel assemblies detected at the final core verification stage. By letter dated November 6, 1980, TVA committed to make changes in their fuel handling procedures. These new procedures are being followed during the fuel shuffling operations for BF-3. For BF-3, an independent QA inspector on the refueling bridge is being used to check fuel element orientation and location.

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-3. 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 100% of design pressure, i.e., 1375 psig. The reanalysis, which is presented in Section 12 of the supplemental reload submittal(2), shows that the peak pressure at the bottom of the reactor vessel does not exceed 1299 psig for worst case end-of-cycle conditions, even when assuming the effects of one failed safety valve. This is a decrease of 1 psig from the previous fuel cycle and is the reason for the changes on pages 30 and 225 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 9.

3.1.4 Thermal Hydraulic Stability

A thermal-hydraulic stability analysis was performed for this reload using the methods described in Reference 9. The results, which are presented in Section 13 of the Supplemental Reload Licensing Submittal(2) show that the fuel dependent channel hydrodynamic stability decay ratios and reactor core stability decay ratio at the least stable operating state (corresponding to the intersection of the natural circulation power curve and the 105% rod line) are 0.29 (8X8R/P8X8R), 0.36 (8X8) and 0.85 respectively. These predicted decay ratios are all well below the 1.0 Ultimate Performance Limit decay ratio proposed by GE.

Prior to Cycle 3 operation, the staff as an interim measure, added a requirement to the BF-3 Technical Specifications which restricted planned plant operation in the natural circulation mode. Continuation of this restriction will also provide a significant increase in the reactor core stability operating margins during Cycle 4. On the basis of the foregoing, the staff considers the thermal-hydraulic stability of BF-3 during Cycle 4 to be 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 Addition of MG Sets

As noted in the Discussion above, the modifications to the LPCI systems at Browns Ferry Units 1, 2 and 3 were approved by Amendment Nos. 51, 45 and 23 issued May 11, 1979. The modification for BF-3 was completed during the previous refueling outage except for the addition of the MG sets in the feeder lines to the RMOV Boards. The reload amendment for the previous fuel cycle (cycle three) - Amendment No. 28 to Facility License No. DPR-58 issued November 30, 1979 - included changes to the Technical Specifications resulting from this modification as well as our evaluation of TVA's reanalysis of the Loss of Coolant Accident (LOCA) for BF-3 with the LPCI modifications in place. The only changes to the Technical Specifications related to the modifications in this amendment is the addition of operability and surveillance requirements for the new MG sets and the bases therefore. We have reviewed the proposed additions to the Technical Specifications and find them acceptable.

3.3 Hydrogen Monitoring System

The proposed changes to the BF-3 Technical Specifications involve the number of gas analyzer systems in the drywell (changed from 2 to 1 system) and the time interval between performing a calibration test on the gas analyzer systems (from once a month to quarterly).

Hydrogen concentrations in the containments of the Browns Ferry reactors are currently measured by hydrogen electrode sensors installed in the drywell and in the suppression pool torus of each reactor. TVA proposes to replace these sensors with thermal conductivity gas analyzers located outside the primary containment for easy access. Gas sample lines will lead from the upper part of each drywell and torus through existing penetrations to a sampling cabinet outside the primary containment. The sample will pass through approximately 100 feet of 1/2-inch stainless steel pipe, a water trap and chiller to remove entrained moisture, a bellows pump and through either of two independent thermal conductivity sensors, exhausting back into the drywell.

The hydrogen monitoring system will be operating continuously during reactor operation and the sample will reach the sensor in less than 2 minutes at the pumping speed of the bellows pump. 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 (2% of 20% full scale).

We have found the proposed hydrogen monitoring system to be acceptable after evaluating it against the acceptance criteria and requirements listed in the following:

(1) Standard Review Plan 6.2.4, "Containment Isolation System."

(2) Standard Review Plan 6.2.5, "Combustible Gas Control in Containment."

(3) NUREG-0578, "Lessons Learned Task Force Status Report and Short-Term Recommendations," Sections 2.1.5, 2.1.8.a and Appendix A (2.1.8a), "Improved Post-Accident Sampling Capability."

(4) NUREG-0737, "Clarification of TMI Action Plan Requirements."

We find the proposed thermal conductivity method has adequate sensitivity and is at least as reliable as the currently used hydrogen electrode method.

The thermal conductivity method, including the sampling system, will be in continuous operation before and after the initial phase of the accident with readings and controls in the control room. The system is designed to be operable under accident conditions and at negative pressures down to 2 psi below ambient atmospheric pressure and will measure H_2 concentration in the range of 0.1% V/o to 20% V/o. The response time of about 2 minutes is sufficiently rapid to provide timely warning of hazardous hydrogen concentrations in containment after an accident. The requirement in NUREG-0578, Appendix A (Section 2.1.8a) is for sampling and analysis within an hour after an accident. Exposure to the operator during sampling meets the "As Low As Reasonably Achievable" levels because there is no need for an operator to be near the analyzer during its operation. If maintenance after an accident is required, radioactive gases can be purged out of the analyzer remotely from the control room. The location of the sensors outside primary containment also makes them more accessible for maintenance and inspection during normal reactor operation.

No additional penetrations of the primary containment will be required since the new sampling lines will pass through unused existing spare penetrations. The power circuits to operate the hydrogen monitoring system will meet the safety requirements of engineered safety features. The redundancy requirement will be met by providing two independent thermal conductivity sensors to which gas samples from either the drywell or the torus atmosphere may be directed. The licensee has verbally indicated that each of the sampling lines contain two automatic isolation valves that will automatically isolate upon receipt of a containment isolation signal. The operator will then manually open these lines within 30 minutes of initiation of safety injection to sample the containment atmosphere.

Based on our evaluation, we conclude 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.

The system being replaced had two gas analyzer systems in the drywell and one gas analyzer system in the wetwell, and was calibrated monthly using standard gas samples. The new system will have one gas analyzer system for the drywell and one analyzer for the wetwell with monthly channel functional tests and quarterly channel calibration tests using standard gas samples. Redundancy is provided by the drywell purge system which limits the hydrogen concentration difference between the drywell and wetwell to 0.2V/o. Therefore, if either gas analyzer fails the operator can still be able to measure, to an acceptable degree of accuracy, the hydrogen concentration in both the wetwell and drywell using only one gas analyzer.

The interval between calibration of the gas analyzer system was lengthened from one month to 3 months because gas analyzer systems that use the thermal conductivity method are inherently more stable and less susceptible to drift.

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:

References

1. Tennessee Valley Authority letter (L. M. Mills) to USNRC (H. R. Denton) dated August 27, 1980 (TVA BFNP TS 148).

2. "Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Power Station Unit 3, Reload No. 3," Y1003J01A03 dated August 1980.

3. "Proposed Technical Specification Changes, Browns Ferry Nuclear Plant Unit 3" submitted as enclosure 1 to TVA letter (L. M. Mills) to USNRC (H. R. Denton) dated August 27, 1980.

4. Tennessee Valley Authority letter (L. M. Mills) to USNRC (T. A. Ippolito) dated September 23, 1980.

5. Tennessee Valley Authority letter (L. M. Mills) to USNRC (H. R. Denton) dated October 14, 1980 (TVA BFNP TS 148).

6. "Proposed Changes to Technical Specifications, Browns Ferry Nuclear Plant Unit 3" submitted as enclosure 1 to TVA letter (L. M. Mills) to USNRC (H. R. Denton) dated October 14, 1980.

7. Tennessee Valley Authority letter (J. L. Cross) to USNRC (H. R. Denton) dated September 5, 1980 (TVA BFNP TS 148).

8. Tennessee Valley Authority letter (L. M. Mills) to USNRC (H. R. Denton) dated October 17, 1980 (TVA BFNP TS 148).

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

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

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-296

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 37 to Facility Operating License No. DPR-68 issued to the Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit No. 3, located in Limestone County, Alabama. The amendment is effective as of the date of issuance.

This amendment changes the Technical Specifications to: (1) incorporate the limiting conditions for operation during the fourth fuel cycle, (2) reflect new primary containment hydrogen monitoring instrumentation being installed during the current refueling outage and (3) reflect the addition of 480 volt motor generator sets during the refueling outage to supply reactor motor operated valve boards 3D and 3E.

The applications for the amendment comply 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 Section 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated August 27, 1980, as supplemented by letters dated September 23, 1980 and October 14, 1980 and September 5, 1980 and October 17, 1980, (2) Amendment No. 37 to License No. DPR-68, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

Thomas A. Ippolito, Chief
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Dated at Bethesda, Maryland this 12th day of January 1981