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NOVEMBER 30 1979

Docket No. 50-296

REGULATORY DOCKET FILE COPY

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. 28 to Facility License No. DPR-58 for the Browns Ferry Nuclear Plant, Unit No. 3. This amendment changes the Technical Specifications in response to your request of August 6, 1979 (TVA BFNP TS 127) as supplemented by your two letters dated September 26, 1979 and your letters dated October 10, 1979 and October 25, 1979.

The changes to the Technical Specifications (1) incorporate the limiting conditions for operation during the third fuel cycle, (2) reflect facility modifications made during the current refueling outage to eliminate the low pressure coolant injection loop selection logic (the design for which was approved by the Commission's letter of May 11, 1979 transmitting Amendment No. 23 to License No. DPR-68), (3) reflect the rerouting of the reactor water cleanup system (RWCU) piping to reduce thermal cycling on the feedwater nozzles and thus provide increased margin against the initiation and propagation of cracks in these nozzles and (4) reflect replacement of two of the 11 safety relief valves with valves set at 1150 psi rather than 1125 psig.

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

Sincerely,
Original Signed by
T. A. Ippolito

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

Enclosures:
1. Amendment No. 28 to DPR-68
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

No legal objection to amendment. Issued notice. SER not reviewed

CP

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OFFICE	ORB #3	ORB #3	~~AD:ORP~~	OELD	ORB #3	
SURNAME	SSheppard	RClark:mjf	WGammill	Cutchin	TIppolito	
DATE	11/30/79	11/29/79	11/30/79	11/30/79	11/30/79	

NRC FORM 318 (9-76) NRCM 0240 ☆U.S. GOVERNMENT PRINTING OFFICE: 1979-289-369 cc:

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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. 28
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated August 6, 1979, as supplemented by two letters dated September 26, 1979 and additional letters dated October 10, 1979 and October 25, 1979, 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. These 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. 28 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

7912130706

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 #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 30, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 28

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:

11	153
13	154
17	167
24	169
26	176
27	178
29	181
30	182
64	195
66	196
67	225
68	225a (new page)
70	227
75	266
93	267
94	276
96	281
97	294
109	318
136	321
149	325
150	327
151	360

2. The marginal lines on each page indicates the revised area.

1.1 FUEL CLADDING INTEGRITY

2.1 FUEL CLADDING INTEGRITY

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 the prescribed limits.

Surveillance requirements for APRM scram set-points 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.

1.1 FUEL CLADDING INTEGRITY

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.

2.1 FUEL CLADDING INTEGRITY

C.	Scram and isolation reactor low water level	≥ 538 in. above vessel zero
D.	Scram--turbine stop valve closure	≤ 10 percent valve closure
E.	Scram--turbine control valve	
	1. Fast closure--Upon trip of the fast acting solenoid valves	
	2. Loss of control oil pressure	≥ 1,100 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	≤ 850 psig
I.	Core spray and LPCI actuation--reactor low water level	≥ 378 in. above vessel zero
J.	HPCI and RCIC actuation--reactor low water level	≥ 470 in. above vessel zero
K.	Main steam isolation valve closure--reactor low water level	≥ 470 in. above vessel zero

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.

2. General Electric Supplemental Reload Licensing Submittal for BFNP unit 3 Reload 2, NEDO-24199.

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 o. 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. General Electric Supplemental Reload Licensing Submittal for BFNP Unit 3 Reload 2, NEDO-24199.

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	
Target - Rocks	1,105 psig ± 11 psi (4 valves)
	1,115 psig ± 11 psi (4 valves)

1.2 REACTOR COOLANT SYSTEM INTEGRITY

2.2 REACTOR COOLANT SYSTEM INTEGRITY

	1,125 psig ± 11 psi (1 valve)
Crosbys**	1,150 psig + 0 psi - 22 psi (2 valves)
OR Target-Rock**	1,125 psig ± 11psi (2 valves)

C. Scram—nuclear system high pressure ≤ 1,055 psig

* Analyses have been run which allow operation with either 9 Target-Rocks and 2 Crosby's or 11 Target-Rocks as indicated in the above specification. The results of these analyses are presented in the Bases.

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)
5. General Electric Supplemental Reload Licensing Submittal for BFNP Unit 3 Reload 2, NEDO-24199.

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.

9 Target Rock And 2 Crosby Valves

To meet the safety design basis, thirteen safety-relief valves have been installed on each unit with a total capacity of 81.08% 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 of 1293 psig if a neutron flux scram is assumed. This results in a 82 psig margin to the code allowable overpressure limit of 1375 psig.

To meet the operational design basis, the total safety-relief capacity of 81.08% of nuclear boiler rated has been divided into 66.88% 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 Reference 5 on page 29. This analysis shows that the 11 relief valves limit pressure at the safety valves to 1218 psig, well below the setting of the safety valves. Therefore, the safety valves will not open. This analysis shows that peak system pressure is limited to 1243 psig which is 132 psig below the allowed vessel overpressure of 1375 psig.

11 Target Rock Valves Only

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 of 1280 psig if a neutron flux scram is assumed. This results in a 95 psig 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 Reference 5 on page 29. This analysis shows that the 11 relief valves limit pressure at the safety valves to 1206 psig, well below the setting of the safety valves. Therefore, the safety valves will not open. This analysis shows that peak system pressure is limited to 1232 psig which is 143 psig below the allowed vessel overpressure of 1375 psig.

Table 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero.	A	1. Below trip setting initiated HPCI.
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #1)	≥ 378" above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #2)	≥ 378" above vessel zero.	A	1. Below trip settings in conjunction with drywell high pressure, low water level permissive, 120 sec. del timer and CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184 & 185, SW #1)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LITS-3-52 & 62, SW #1)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of of containment spray during accident condition.

Table 3.2.B

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Pressure (PS-3-74 A & B, SW #2) (PS-68-95, SW #2) (PS-68-96, SW #2)	450 psig ± 15	A	1. Below trip setting permissive for opening CSS and LPCI admission valves.
2	Instrument Channel - Reactor Low Pressure (PS-3-74A & B, SW #1) (PS-68-95, SW #1) (PS-68-96, SW #1)	230 psig ± 15	A	1. Recirculation discharge valve actuation.
1	Instrument Channel - Reactor Low Pressure (PS-68-93 & 94, SW #1)	100 psig ± 15	A	1. Below trip setting in conjunction with containment isolation signal and both suction valves open will close RHR (LPCI) admission valves.
2	Core Spray Auto Sequencing Timers (5)	$6 \le t \le 8$ secs.	B	1. With diesel power 2. One per motor
2	LPCI Auto Sequencing Timers (5)	$0 \le t \le 1$ sec.	B	1. With diesel power 2. One per motor
1	RHRSW A1, B3, C1, and D3 Timers	$13 \le t \le 15$ sec.	A	1. With diesel power 2. One per pump

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Table 3.2.B

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Core Spray and LPCI Auto Sequencing Timers (6)	$0 \leq t \leq 1$ sec. $6 \leq t \leq 8$ sec. $12 \leq t \leq 16$ sec. $18 \leq t \leq 24$ sec.	B	1. With normal power 2. One per CSS motor 3. Two per RHR motor
1	RHRSW A1, B3, C1, and D3 Timers	$27 \leq t \leq 29$ sec.	A	1. With normal power 2. One per pump
1(16)	ADS Timer	120 sec ± 5	A	1. Above trip setting in conjunction with low reactor water level, high drywell pressure and LPCI or CSS pumps running initiates ADS.
2	Instrument Channel - RHR Discharge Pressure	100 ± 10 psig	A	1. Below trip setting defers ADS actuation.

Table 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimun No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel CSS Pump Discharge Pressure	185 ± 10 psig	A	1. Below trip setting defers ADS actuation.
1(3)	Core Spray Sparger to Reactor Pressure Vessel d/p	2 psid ± 0.4	A	1. Alarm to detect core spray sparger pipe break.
1	RHR (LPCI) Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.

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Table 3.2.B

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2(2)	Instrument Channel - Reactor High Water Level	≤583" above vessel zero.	A	1. Above trip setting trips HPCI turbine.
1	Instrument Channel - HPCI Turbine Steam Line High Flow	≤ 90 psi (7)	A	1. Above trip setting isolates HPCI system and trips HPCI turbine.
4(4)	Instrument Channel - HPCI Steam Line Space High Temperature	≤200°F.	A	1. Above trip setting isolates HPCI system and trips HPCI turbine.
1	Core Spray System Logic	N/A	B	1. Includes testing auto initiation inhibit to Core Spray Systems in other units.
1	RCIC System (Initiating) Logic	N/A	B	1. Includes Group 7 valves. Refer to Table 3.7.A for list of valves.
1	RCIC System (Isolation) Logic	N/A	B	1. Includes Group 5 valves. Refer to Table 3.7.A for list of valves.
1(16)	ADS Logic	N/A	A	
1	RHR (LPCI) System (Initiation)	N/A	B	

10. Only one trip system for each cooler fan.

11. In only two of the four 4160 V shutdown boards. See note 13.

12. In only one of the four 4160 V shutdown boards. See note 13.

13. An emergency 4160 V shutdown board is considered a trip system.

14. RHRSW pump would be inoperable. Refer to section 4.5.C for the requirements of a RHRSW pump being inoperable.

15. The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level (≥ 378" above vessel zero) originating in the core spray system trip system.

16. The ADS circuitry is capable of accomplishing its protective action with one operable trip system. Therefore one trip system may be taken out of service for functional testing and calibration for a period not to exceed 8 hours.

17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds 2 consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if 1 RPT system is inoperable for more than 72 consecutive hours, an orderly power reduction shall be initiated and the reactor power shall be less than 85% within 4 hours.

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TABLE 4.2.B
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
Instrument Channel Reactor Low Pressure (PS-3-74A & B) (PS-68-95) (PS-68-96)	(1)	once/3 months	none
Instrument Channel Reactor Low Pressure (PS-68-93 & 94)	(1)	once/3 months	none
Core Spray Auto Sequencing Timers (Normal Power)	(4)	once/operating cycle	none
Core Spray Auto Sequencing Timers (Diesel Power)	(4)	once/operating cycle	none
LPCI Auto Sequencing Timers (Normal Power)	(4)	once/operating cycle	none
LPCI Auto Sequencing Timers (Diesel Power)	(4)	once/operating cycle	none
RHRSW A1, B3, C1, D3 Timers (Normal Power)	(4)	once/operating cycle	none
RHRSW A1, B3, C1, D3 Timers (Diesel Power)	(4)	once/operating cycle	none

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TABLE 4.2.B
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
ADS Timer	(4)	once/operating cycle	none
Instrument Channel RHR Pump Discharge Pressure	(1)	once/3 months	none
Instrument Channel Core Spray Pump Discharge Pressure	(1)	once/3 months	none
Core Spray Sparger to RPV d/p	(1)	once/3 months	once/day
Trip System Bus Power Monitor	once/operating cycle	N/A	none
Instrument Channel Condensate Storage Tank Low Level	(1)	once/3 months	none

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TABLE 4.2.B
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
LPCI (Containment Spray) Logic	once/6 months	(6)	N/A
Core Spray Loop A Discharge Pressure (PI-75-20)	N/A	once/6 months	once/day
Core Spray Loop B Discharge Pressure (PI-75-48)	N/A	once/6 months	once/day
RHR Loop A Discharge Pressure (PI-74-51)	N/A	once/6 months	once/day
RHR Loop B Discharge Pressure (PI-74-65)	N/A	once/6 months	once/day
Instrument Channel - RHR Start	Tested during functional test of RHR pump (refer to section 4.5.B).	N/A	N/A
Instrument Channel - Thermostat (RHR Area Cooler Fan)	once/month	once/6 months	N/A
Instrument Channel - Core Spray A or C Start	Tested during functional test of core spray (refer to section 4.5.A).	N/A	N/A
Instrument Channel - Core Spray B or D start	Tested during functional test of core spray (refer	N/A	N/A

TABLE 4.2.B
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
	to section 4.5.A).		
Instrument Channel - Thermostat (Core Spray Area Cooler Fan)	once/ month	once/6 months	N/A
RHR Area Cooler Fan Logic	Tested during functional test of instrument channels, RHR motor start and thermostat (RHR area cooler fan). No other test required.	N/A	N/A
Core Spray Area Cooler Fan Logic	Tested during logic system functional test of instrument channels, core spray motor start and thermostat (core spray area cooler fan). No other test required.	N/A	N/A
Instrument Channel - Core Spray Motors A or D Start	Tested during functional test of core spray pump (refer to section 4.5.A).	N/A	N/A
Instrument Channel - Core Spray Motors B or C Start	Tested during functional test of core spray pump (refer to section 4.5.A).	N/A	N/A
RPT initiate logic	once/month	N/A	N/A
RPT breaker	once/operating cycle	N/A	N/A

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

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

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

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

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

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. General Electric Supplemental Reload Licensing Submittal for BFNP unit 3 Reload 2, NEDO-24199, July 1979.

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

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

1. The RHRS shall be operable:

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

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

2. With the reactor vessel pressure less than 105 psig, the RHR may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain operable) for a period not to exceed 24 hours while being drained of

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

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

1.	a. Simulated Automatic Actuation Test	Once/ Operating Cycle
	b. Pump Operability	Once/ month
	c. Motor Operated valve operability	Once/ month
	d. Pump Flow Rate	Once/3 Months
	e. Testable check valve	Once/ operating cycle

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

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

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are operable.

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

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

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

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

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

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

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

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

4. No additional surveillance required.

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

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

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

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

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

11. When there is irradiated fuel in the reactor and the reactor vessel pressure is greater than atmospheric, unit 2 RHR pumps B and D with associated heat exchangers and valves must be operable and capable of supplying cross-connect capability except as specified in specification 3.5.B.12 below.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

second operable access path for the same phase of the mode (drywell sprays, suppression chamber sprays and suppression pool cooling) shall be demonstrated to be operable daily thereafter until the second path is returned to normal service.

8. No additional surveillance required.

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

10. No additional surveillance required.

11. The B and D RHR pumps on unit 2 which supply cross-connect capability shall be operable monthly when the cross-connect capability is required.

12. When it is determined that one RHR pump or associated heat exchanger located on the unit cross-connection in the

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

(Note: Because cross-connect capability is not a short term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

12. If one RHR pump or associated heat exchanger located on the unit cross-connection in unit 2 is inoperable for any reason (including valve inoperability, pipe break, etc.), the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pump and associated diesel generator are operable.

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

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

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

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

13. No additional surveillance required.

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

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 1.28 for 8x8 fuel, and 1.25 for 8x8R fuel, and 1.26 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.

adequate core cooling. With due regard for this margin, the allowable repair time of 7 days was chosen.

Should one RHR pump (LPCI mode) become inoperable, only 3 RHR pumps (LPCI mode) and the core spray system are available. Since this leaves only one RHR pump (LPCI mode) in reserve, which along with the remaining 2 RHR pumps (LPCI mode) and core spray system is demonstrated to be operable immediately and daily thereafter, a 7 day repair period is justified.

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

Should one RHR pump (containment cooling mode) become inoperable, a complement of three full capacity containment heat removal systems is still available. Any two of the remaining pumps/heat exchanger combinations would provide more than adequate containment cooling for any abnormal or post accident situation. Because of the availability of equipment in access of normal redundance requirements, which is demonstrated to be operable immediately and with specified subsequent performance, a 30-day repair period is justified.

Should two RHR pumps (containment cooling mode) become inoperable, a full heat removal system is still available. The remaining pump/heat exchanger combinations would provide adequate containment cooling for any abnormal post accident situation. Because of the availability of a full complement of heat removal equipment, which is demonstrated to be operable immediately and with specified performance, a 7-day repair period is justified.

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

Since the RHRS is filled with low quality water during power operation, it is planned that the system be filled with demineralized (condensate) water before using the shutdown cooling function of the RHR system. Since it is desirable to 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 the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting value for MAPLHGR is shown in Tables 3.5.I-1, -2, -3. The analyses supporting these limiting values is presented in NEDO-24127 and NEDO-24194.

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 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

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. General Electric Supplemental Reload Licensing Submittal for BFNP Unit 3 Reload 2, NEDO-24199.

TABLE 3.5.I-1

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3

Fuel Type: Inital Core - Type 2

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)
200	11.4
1,000	11.6
5,000	12.0
10,000	12.2
15,000	12.3
20,000	12.1
25,000	11.3
30,000	10.2

TABLE 3.5.I-2

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3

Fuel Type: Initial Core - Type 1

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.3
5,000	11.8
10,000	12.1
15,000	12.3
20,000	12.1
25,000	11.3
30,000	10.2

TABLE 3.5.I-3

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3

FUEL TYPES: 8DRB265L and P8DRB265L

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)
200	11.6
1,000	11.6
5,000	12.1
10,000	12.1
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7

The values in this table are conservative for both prepressurized and non-pressurized fuel.

3.6 PRIMARY SYSTEM BOUNDARY

F. Jet Pump Flow Mismatch

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

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

4.6 PRIMARY SYSTEM BOUNDARY

F. Jet Pump Flow Mismatch

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

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

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

G. Structural Integrity

1. The structural integrity of the primary system shall be maintained at the level required by the original acceptance standards throughout the life of the plant. The reactor shall be maintained in a cold shutdown condition until each indication of a defect has been investigated and evaluated.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

G. Structural Integrity

1. Table 4.6.A together with supplementary notes, specifies the inservice inspection surveillance requirements of the reactor coolant system as follows:

   a. areas to be inspected

   b. percent of areas to be inspected during the inspection interval

   c. inspection frequency

   d. methods used for inspection

2. Evaluation of inservice inspections will be made to the acceptance standards specified for the original equipment.

3. The inspection interval shall be 10 years.

4. Additional inspections shall be performed on certain circumferential pipe welds as listed to provide additional protection against pipe whip, which could damage auxiliary and control systems.

   Feedwater- GFW-9, KFW-13, GFW-12, GFW-26, KFW-31, GFW-29, KFW-39, GFW-15, KFW-38, and GFW-32

3.6/4.6 BASES

9 Target Rock And 2 Crosby Valves

To meet the safety design basis, thirteen safety-relief valves have been installed on unit 3 with a total capacity of 81.08% 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 of 1293 psig if a neutron flux scram is assumed This results in an 82 psig margin of the code allowable overpressure limit of 1375 psig.

To meet the operational design basis, the total safety-relief capacity of 81.08% of nuclear boiler rated has been divided into 66.88% 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 Reference 5 on page 29. This analysis shows that the 11 relief valves limit pressure at the safety valves to 1218 psig, well below the setting of the safety valves. Therefore, the safety valves will not open. This analysis shows that peak system pressure is limited to 1243 psig which is 132 psig below the allowed vessel overpressure of 1375 psig.

11 Target Rock Valves Only

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 of 1280 psig if a neutron flux scram is assumed This results in an 95 psig margin of 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 Reference 5 on page 29. This analysis shows that the 11 relief valves limit pressure at the safety valves to 1206 psig, well below the setting of the safety valves. Therefore, the safety valves will not open. This analysis shows that peak system pressure is limited to 1232 psig which is 143 psig 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)

*This is plus zero (+ 0 psi), minus 2% (- 22 psi) for Crosby valves

3.6/4.6 BASES

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

3.6.F/4.6.F Jet Pump Flow Mismatch

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

ECCS performance during reactor operation with one recirculation loop out of service has not been analyzed. Therefore, sustained reactor operation under such conditions is not permitted.

3.6.G/4.6.G Structural Integrity

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

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

NOTES FOR TABLE 3.7.A

Key: 0 = Open
C = Closed
SC = Stays Closed
GC = Goes Closed

Note: Isolation groupings are as follows:

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

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

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

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

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

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

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

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

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

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

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

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

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

1. Reactor Vessel Low Water Level (470")

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

1. High Drywell Pressure

TABLE 3.7.D (Continued)
PRIMARY CONTAINMENT ISOLATION VALVES

Valves	Valve Identification	Test Medium
84-8D	Containment Atmospheric Dilution	Air
84-19	Containment Atmospheric Dilution	Air
76-49	Containment Atmospheric Monitor	Air
76-50	Containment Atmospheric Monitor	Air
76-51	Containment Atmospheric Monitor	Air
76-52	Containment Atmospheric Monitor	Air
76-53	Containment Atmospheric Monitor	Air
76-54	Containment Atmospheric Monitor	Air
76-55	Containment Atmospheric Monitor	Air
76-56	Containment Atmospheric Monitor	Air
76-57	Containment Atmospheric Monitor	Air
76-58	Containment Atmospheric Monitor	Air
76-59	Containment Atmospheric Monitor	Air
76-60	Containment Atmospheric Monitor	Air
76-61	Containment Atmospheric Monitor	Air
76-62	Containment Atmospheric Monitor	Air
76-63	Containment Atmospheric Monitor	Air
76-64	Containment Atmospheric Monitor	Air
76-65	Containment Atmospheric Monitor	Air
76-67	Containment Atmospheric Monitor	Air
76-68	Containment Atmospheric Monitor	Air
76-215	Containment Atmospheric Monitor	Air
76-217	Containment Atmospheric Monitor	Air
76-220	Containment Atmospheric Monitor	Air
76-222	Containment Atmospheric Monitor	Air
76-225	Containment Atmospheric Monitor	Air
76-226	Containment Atmospheric Monitor	Air
76-229	Containment Atmospheric Monitor	Air
76-230	Containment Atmospheric Monitor	Air
76-237	Containment Atmospheric Monitor	Air

TABLE 3.7.G
CHECK VALVES ON DRYWELL INFLUENT LINES

Valves	Valve Identification	Test Medium
3-554	Feedwater	Air
3-558	Feedwater	Air
3-568	Feedwater	Air
3-572	Feedwater	Air
63-525	Standby Liquid Control Discharge	Air
63-526	Standby Liquid Control Discharge	Air
69-579	RWCU Return (Feedwater Line B)	Air
69-624	RWCU Return (Feedwater Line A)	Air
71-40	RCIC Pump Discharge	Air
73-45	HPCI Pump Discharge	Air
85-576	CRD Hydraulic Return	Air

3.7.D/4.7.D <u>Primary Containment Isolation Valves</u>

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

<u>Group 1</u> - process lines are isolated by reactor vessel low water level (470") in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in group 1 are also closed when process instrumentation detects excessive main steam line flow, high radiation, low pressure, or main steam space high temperature.

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

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

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

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

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

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

3.9 AUXILIARY ELECTRICAL SYSTEM

2. Three unit 3 diesel generators shall be operable.

4.9 AUXILIARY ELECTRICAL SYSTEM

d. Each diesel generator shall be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.

e. Once a month a sample of diesel fuel shall be checked for quality. The quality shall be within the acceptable limits specified in Table 1 of the latest revision to ASTM D975 and logged.

2. D.C. Power System - Unit Batteries (250-Volt) and Diesel Generator Batteries (125-Volt) and Shutdown Board Battery (250-Volt)

   a. Every week the specific gravity and the voltage of the pilot cell, and temperature of an adjacent cell and overall battery voltage shall be measured and logged.

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.

3.9 AUXILIARY ELECTRICAL SYSTEM

4. From and after the date that the 250-Volt Shutdown board batteries or one of the three 250-Volt unit batteries and/or its associated battery board is found to be inoperable for any reason, continued reactor operation is permissible during the succeeding seven days. Except for routine surveillance testing, the NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the plans to return the failed component to an operable state.

5. When one division of the Logic System is inoperable, continued reactor operation is permissible under this condition for seven days, provided the CSCS requirements listed in Specification 3.9.B.2 are satisfied. The NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the plans to return the failed component to an operable state.

4.9 AUXILIARY ELECTRICAL SYSTEM

3.9 BASES

The objective of this specification is to assure an adequate source of electrical power to operate facilities to cool the unit during shutdown and to operate the engineered safeguards following an accident. There are three sources of alternating current electrical energy available, namely, the 161-kV transmission system, the nuclear generating units, and the diesel generators.

The 161-kV offsite power supply consists of two lines which are fed from different sections of the TVA 161-kV grid. In the normal mode of operation, the 161-kV system is operating and four diesel generators are operational. If one diesel generator is out of service, there normally remain the 161-kV sources, and the other three diesel generators. For a diesel generator to be considered operable its associated 125 V battery must be operable.

The minimum fuel oil requirement of 103,300 gallons is sufficient for 7 days of full load operation of 3 diesels and is conservatively based on availability of a replenishment supply.

Offsite auxiliary power for Browns Ferry Nuclear Plant Unit 3 is supplied from two sources: the unit station transformers from the main generator or the 161-kV transmission system through the cooling tower transformers. If a cooling tower transformer is lost, the unit can continue to operate since the station transformer is in service, the other cooling tower transformer is available, and four diesel generators are operational.

A 4-kV shutdown board is allowed to be out of operation for a brief period to allow for maintenance and testing, providing all remaining 4-kV shutdown boards and associated diesel generators CS, RHR, (LPCI and Containment Cooling) Systems supplied by the remaining 4-kV shutdown boards, and all emergency 480 V power boards are operable.

There are five 250-volt d-c battery systems each of which consists of a battery, battery charger, and distribution equipment. Three of these systems provide power for unit control functions, operative power for unit motor loads, and alternative drive power for a 115-volt a-c unit preferred motor-generator set. One 250-volt d-c system provides power for common plant and transmission system control functions, drive power for a 115-volt a-c plant preferred motor-generator set, and emergency drive power for certain unit large motor loads. The fifth battery system delivers control power to a 4-kV shutdown board.

The 250-Volt dc 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

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

Browns Ferry units 1, 2, and 3 are 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 8x8 assemblies having 63 fuel rods each, and 8x8 R (and P8x8R) assemblies having 62 fuel rods each. The number of each type in the core is given in the most recent reload amendment topical report.

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 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 the fuel in the new-fuel storage facilitity shall be such that k_{eff}, for dry conditions,

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 28 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 6, 1979 (TVA BFNP TS 127), and supplemented by two letters dated September 26, 1979 and letters dated October 10, 1979 and October 25, 1979, 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 amendment and revised Technical Specifications were to: (1) incorporate the limiting conditions for operation associated with the third fuel cycle, (2) reflect facility modifications made during the current refueling outage to eliminate the low pressure coolant injection (LPCI) loop selection logic, (3) add a check valve in the reactor water cleanup (RWCU) system piping as a result of rerouting this piping so that the return flow is distributed equally among the feedwater lines and (4) reflect replacement of two of the eleven safety-relief valves with valves set to relieve at 1150 psig rather than 1125 psig.

Browns Ferry Unit No. 3 (BF-3) shutdown for refueling on August 24, 1979. Besides routine maintenance and equipment overhaul, several significant modifications were completed, including main steam relief valve (MSRV) tailpipe routing, core spray piping modifications, feedwater sparger modifications and LPCI modifications. Because of these modifications, all of the fuel was removed from the reactor vessel and stored in the spent fuel pool (SFP) while the work was in progress.

1.1 Reload

The initial core loading for Browns Ferry Unit No. 3 consisted of 764 of the single water rod 8 X 8 fuel assemblies, each containing 63 fuel rods. During the first refueling in September 1978, 208 of the fuel assemblies were replaced with 8 X 8 fuel assemblies containing 62 fuel rods in each. During the present refueling outage, an additional 144 of the initial fuel bundles were replaced with P 8 X 8 fuel assemblies, each containing 62 fuel rods. The prepressurized fuel assemblies (P 8 X8R) are essentially identical from a core physics standpoint to the two water rod fuel assemblies (8 X 8R) except that they are prepressurized with about three rather than one atmospheres of helium to minimize fuel clad interaction. Our evaluation of the P 8 X 8 R 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

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heluim 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 6, 1979, and supplemented by letter(2) dated October 25, 1979, a supplemental reload licensing document(3) prepared by General Electric Company (G.E.) for TVA and proposed changes to the BF-3 Technical Specifications(4).

1.2 LPCI Modification

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 (the current 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 modifies 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.

By their letter(1) dated August 6, 1979, TVA submitted proposed changes to the Technical Specifications(4) associated with the above modifications. Since this modification constitutes a change to the Emergency Core Cooling System (ECCS), TVA by letter(5) dated September 26, 1979 also transmitted a revised "Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3." The modifications to the BF-3

ECCS make it functionally identical to the ECCS system currently installed at Browns Ferry Units 1 and 2. The purpose of the changes is to upgrade overall performance of the BF-3 ECCS by assuring delivery of LPCI injection flow to the core in the event of a postulated break in the suction side of the recirculation system piping. By improving ECCS performance for this currently limiting break, additional margin to the 2200°F peak cladding temperature limit can be achieved. Our review of the BF-3 Loss of Coolant Accident reanalysis results together with those Technical Specifications required to implement the analysis results and assumptions is contained within this safety evaluation.

1.3 Modification of Reactor Water Cleanup System Piping

In the past, cracks have been detected in some BWR piping systems. The staff's investigation and evaluation of the causes of these cracks and recommended actions to minimize cracking potential has been reported in NUREG-0313[7] a revision to which was issued in October 1979, and NUREG-0531[8]. The cracks have generally been attributed to stress corrosion cracking. For this to occur, two elements must be present - a corrosive environment and stress. High purity water is corrosive to any metal. Since the concentration of ions such as iron, chromium and nickel in demineralized water is below the solubility limit and the water is not buffered, the water tends to dissolve or corrode the metal surface. This condition can be aggravated by crevices (such as might exist at fittings or welds) since there is the potential for oxygen concentration cells, and by other conditions in the piping systems (such as stagnant flow conditions). The other causitive element - stress - can result from residual stresses left in the piping during manufacture, stresses induced during fabrication (particularly stresses created by weld joints) and stresses created by operating conditions, such as those caused by thermal shock, vibration, water hammer, etc. The objective is to reduce either the stresses or the corrosivity of the environment - and preferably both - below the threshold required to initiate and propagate stress corrosion cracking. One of the facility modifications recommended by the staff and by the General Electric Company[9] is to modify the Reactor Water Cleanup (RWCU) System return piping so that the return flow is distributed equally among the feedwater lines. TVA performed this modification on BF-3 during the current refueling outage. This modification allows feedwater to be mixed with the higher temperature RWCU return water at low flow rates thereby lessening the thermal cycling on the feedwater nozzle and the consequent thermal fatigue. Because this modification entailed the addition of a check valve, by letter[10] dated September 26, 1979, TVA requested a change to the Technical Specifications to revise Table 3.7.G to include the required check valve. (This letter is separate from the letter of the same date in reference 5.)

1.4 Replacement of Two Safety-Relief Valves

Prior to the refueling outage, BF-3 had 11 Target Rock safety-relief valves. Four of these valves were set to relieve at 1105 psig, 4 were set to relieve at 1115 psig and 3 at 1125 psig. There have been some problems noted with the Target Rock valves as discussed in I&E Circular No. 79-18 [11], and I&E Bulletin 74-4 and IE Bulletin 74-4a[12] During the current refueling outage, TVA has installed two 6R10 Crosby Safety Relief Valves (SRVs) at BF-3 to obtain performance experience with these valves for possible future use at Browns Ferry, Hartsville and Phipps Bend. At the latter plants, the safety-relief valves will also be grouped as at Browns Ferry with respect to set-point pressure; however, whereas the highest setpoint at Browns Ferry is presently 1125 psig, at Hartsville and Phipps Bend the lowest setpoints will be 1165 psig. To obtain experience at a more proto-typical pressure, TVA proposed that the two replacement Crosby relief valves be set at 1150 psig. The two Crosby SRVs set at 1150 psig will replace two Target Rock valves set at 1125 psig in locations G and H which are not automatic depressurization system (ADS) locations.

The Crosby SRV is a simple, direct-acting, spring-loaded valve with an external pneumatic piston. Safety valve action occurs when the inlet pressure forces exceed the spring load and force the valve disc off of its seat.

For manual actuation, the external pneumatic piston is capable of opening the valve against the force of the spring at any steam pressure down to 0 psig. The pneumatic operator is so arranged that if it malfunctioned it would not prevent the valve disc from lifting if steam inlet pressure reached the spring set pressure.

Since the Target Rock valves on Browns Ferry Unit No. 3 have had their throats enlarged to provide increased capacity, the capacity of each of the two Crosby replacement valves is 94.3% of each of the modified Target Rock valves when compared at the same inlet pressure.

By letter[13] dated October 10, 1979, TVA submitted proposed changes to the Technical Specifications associated with replacement of 2 of the 11 safety-relief valves and a revised analysis[14] for the limiting transients to evaluate the impact of using the 2 Crosby SRVs set at 1150 psig in place of 2 of the high set (1125 psig) Target Rock SRVs.

2.0 Discussion

2.1 Reload

This refueling (Reload 2) is the first 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 2 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.[15] Reference 15 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.

Our safety evaluations[16, 17] of GE's generic reload licensing topical report and report amendment concluded that the nuclear and mechanical design of P8x8R fuel used in this reload and GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, as applied to cores containing a mixture of fuel types, are acceptable. Our acceptance of the nuclear and mechanical design of the standard 8x8 (one water rod) fuel was expressed in the staff's evaluation[18] of the information in Reference 19.

As part of our evaluation[16] of Reference 15, we found the cycle-independent input data to be used for the reload transient and accident analyses for BF-3 to be acceptable. The supplementary cycle-dependent information and input data are provided in Reference 3, which follows the format and content of Appendix A of Reference 15.

As a result of the staff's generic evaluations[16,17] of a substantial number of safety considerations related to the use of P8x8R fuel in mixed core loadings with 8x8R and 8x8 fuel, only a limited number of additional review items are included in this evaluation. These include the plant and cycle-specific analysis input data and analysis results presented in Reference 3, and those items identified in Reference 16 as requiring special attention during BWR reload reviews.

2.2 LPCI Modification

The most severe pipe break locations for a boiling water reactor are in the recirculation system discharge and suction line piping. Large breaks occurring in these locations result in the most rapid reactor system depressurization rates and the earliest boiling transition times and uncovery times. For plants with LPCI loop-selection-logic a break in either the recirculation suction line or discharge line, when coupled with a postulated failure of the LPCI injection valve in the intact loop, results in no LPCI flow reaching the core. That is, since all flow is directed to the intact loop through a single injection point, failure (to open) of a single LPCI injection valve results in no LPCI flow reaching the core. Thus for these plants reflood times for recirculation line breaks and a postulated LPCI injection valve failure result in the longest hot node uncovery times since only the two operable core spray systems are available to provide core cooling and to reflood the core.

The worst break size, break location and single failure condition for a plant with LPCI loop selection logic is generally the complete severance of the largest (suction) line, with LPCI injection valve failure. For BF-3(with LPCI loop-selection-logic), the suction line break results[29] in the most rapid jet pump uncovery, boiling transition and hot node uncovery times, with the most delayed core reflooding time due to the unavailability of LPCI. Accordingly, for plants with LPCI loop selection logic, the suction break generally results in the highest peak cladding temperature and establishes the basis for the MAPLHGR limits for the plant.

In order to lessen the severity (PCT) of this limiting (suction) break, with assumed LPCI injection valve failure condition, the licensee modified the BF-3 ECC system during the second refueling outage. The LPCI modification consists of eliminating the LPCI loop-selection-logic system and permanently piping the discharge flow from two LPCI system pumps to one recirculation system discharge line and permanently piping the discharge flow from the other two LPCI system pumps to the second recirculation discharge line. Additionally, the modification will result in both recirculation line discharge valves closing after blowdown following a LOCA. These valves are located between the LPCI injection point on the recirculation discharge line and any potential break location on the suction line. The flow from the LPCI system pumps connected to the broken recirculation line is therefore isolated from any suction line break while the injection flow from the other system is also isolated because it is connected to the unbroken line (since the recirculation loop equalizer valve is locked closed). With this LPCI injection arrangement, only one LPCI loop can be disabled by any single failure and the largest (suction line) break can now derive credit for earlier reflooding due to the availability of at least one half of the LPCI system. The resulting faster core flooding and attendant reduced period of hot node uncovery reduces the PCT calculated for the suction line break to the extent that it potentially could become non-limiting relative to a recirculation discharge line break. At the same time with the subject LPCI modification, the discharge break consequences remain unchanged. All LPCI flow is still lost out the break for the LPCI system connected to the broken loop (since it cannot be isolated from the break by the recirculation discharge line isolation valve), while a postulated LPCI injection valve failure prevents LPCI flow from reaching the core via the intact recirculation loop. That is, as was the case with LPCI loop selection logic, no LPCI flow is available to flood the core. Therefore, although the discharge break is in a smaller diameter line than the suction line (and would normally be expected to yield a lower PCT), the lack of LPCI flow delays reflooding (relative to the suction break where LPCI flow from at least one system is now available) to the extent that this break location can become limiting. Accordingly, for BF-3 the net benefit of the proposed LPCI modifications is that the formerly limiting (in terms of PCT and MAPLHGR requirements) DBA suction line break becomes less severe and thereby improves overall ECCS performance over the spectrum of breaks and worst single failures.

3.0 Evaluation

3.1 Reload

3.1.1 Nuclear Characteristics

For Cycle 3, 144 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 and 8x8R fuel bundles exposed during the previous two 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 3, both the control rod system and the standby liquid control system will have an acceptable shutdown capability during Cycle 3.

3.1.2 Thermal-Hydraulics

3.1.2.1 Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 3, for BWR cores which reload with GE's 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 3 is unchanged from the SLMCPR previously approved for Cycle 2. The basis for this safety limit is addressed in Reference 15, while our generic approvals are given in References 16 and 17.

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 and 8x8R 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 15. Our acceptance of the cycle-independent values appears in Reference 16. 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 16. 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 3. Our evaluation[17] has also addressed the methods used to develop these supplementary input values.

3.1.2.2.2 Thermal-Hydraulic Methods

At the time we completed our evaluation of the generic methods, the acceptability of the GEXL critical power correlation[20], for use in connection with the retrofit fuel design, had not been adequately documented by GE. The staff found, however, that the then available 8x8R critical power test data was sufficient to support the acceptability of GE's 8x8R fuel design for BWR core reloads for one operating cycle. Accordingly, we stated[16] that future BWR core reload applications involving retrofit 8x8 fuel for a second operating cycle would have to include additional information which adequately justified the correlation for application to 8x8R fuel operating beyond one cycle. Since the Reload 2 licensing submittal[3] did not address this issue, we requested[21] that the licensee provide the required additional information. The licensee responded to our request by referencing information [22] furnished to the staff by GE which references a report[23] prepared by GE on this same subject.

Reference 23 provides the results of full scale critical power tests performed on 8x8R fuel bundles. The tests, which included both transient and steady-state simulations, followed the same approved procedures[20] used for the standard 8x8 (single water rod) and 7x7 (all fueled rods) fuel designs. The analysis of a total of 577 steady-state data points was performed using methods also previously approved by the staff. The data, involving nine test assemblies which spanned a range of local power peaking and flow conditions, showed according to GE, that the GEXL correlation was applicable to the 8x8R fuel if adjustment were made to the additive constants used in the formulation of the rod-by-rod R-factors. The local power peaking dependent R-factors are based on the new additive constants shown in Figure 3-1 of Reference 23 which were also used for the BF-3, Reload 1, 8x8R critical bundle power predictions. Using these new additive constants, GE performed a data analysis to assess the accuracy and precision of the GEXL correlation. The results of this analysis showed that the correlation fit provides for a mean predicted-to-measured critical power ratio of 0.9879 with a standard deviation of 0.0234.

When viewed over the range of its applicability (which is the same as the standard 8x8 fuel), the GEXL correlation is therefore somewhat conservatively biased while the statistical variation between the predicted and measured critical power is somewhat less than that associated with the standard 8x8 assembly[20], i.e., 2.34% vs 2.8%. Thus, when viewed over its range of applicability, the 8x8R GEXL correlation (with new additive constants) has somewhat better precision in predicting 8x8R critical bundle powers than the 7x7 and 8x8 GEXL formulations are for predicting 7x7 and 8x8 critical bundle powers respectively. Furthermore, from these results it may also be concluded that the 3.6% standard deviation and best estimate assumption of the GEXL correlation (which were actually used in the GETAB statistical analysis to derive the 1.07 safety limit MCPR) bound the statistical characteristics associated with the subject 8x8R GEXL correlation.

The additional information furnished by GE is also intended to be applicable to all BWR cores which contain 8x8R fuel. Accordingly, this information is also currently being generically reviewed by the staff. Although our evaluation is not yet complete, based on our review to date, we believe that for the range of testing, the 8x8R GEXL correlation has an acceptability and applicability which is equivalent to the 7x7 and 8x8 GEXL correlations previously approved by the staff. From our review of the subject data to date, we have also observed that for those critical power test conditions specifically representative of second cycle fuel operating at normal operating thermal-hydraulic state points, the correlation is somewhat nonconservative in its predictions. This observation focuses in on a correlation behavioral concern not explicitly addressed in the overall GETAB methods approved[23] for the 7x7 and 8x8 fuel types.

Again, this subject is being generically reviewed by the staff. However, until this review is complete, we believe that for Cycle 3 of BF-3, there is sufficient conservatism implicit in the generic determination of the 1.07 safety limit MCPR to offset a possible non-conservatism associated with this concern. That is, specifically, the generic GETAB statistical analysis assumed a 3.6% correlation uncertainty while GE's analysis of the 8x8R test data results in a 2.34% standard deviation. Additionally, the generic evaluation considered an all 8x8R equilibrium core, whereas the Cycle 3 BF-3 core involves 8x8, 8x8R and P8x8R fuel in a non-equilibrium condition. In view of these conservatisms (which are representative of a typical non-equilibrium 8x8R reload core) we believe that the overall thermal-hydraulic (GETAB) methods are adequate for establishing conservative MCPR operating limits for Cycle 3 of BF-3. However, as 8x8R equilibrium conditions are approached, this conservatism will diminish. In order that this conservatism not be substantially eroded, this issue should be addressed for the next reload cycle of BF-3.

3.1.2.2.3 Plant System Transient Simulation Methods

In the analysis of the load rejection with bypass failure and the feedwater controller failure transients, the licensee has taken credit for the beneficial effects of the prompt recirculation pump trip (RPT) as was the case in the previous operating cycle. The RPT feature has the effect of reducing the transient ΔCPR during reactor core pressurization events, by tripping breakers in the electrical circuit between the motor-generator sets and the recirculation pumps on closure of turbine stop or control valves. The prompt RPT immediately reduces core flow and thereby increases core voids. The rapid voiding provides negative reactivity which supplements scram negative reactivity. In this manner, the RPT reduces the thermal power rise during pressurization events. This RPT feature is a thermal margin improvement option which was not generically approved in our evaluation of the reference reload topical report.(17)

The CPR benefit associated with the prompt RPT was calculated with the REDY code.(24) The REDY code employs a two node steamline thermal-hydraulic model and a point kinetics neutronics model. Several pressurization tests(25) at the Peach Bottom Unit 2 boiling water reactor were intended to show the validity of these REDY models.

The experimental results showed, that the REDY steamline model did not accurately predict the pressurization rate which causes the reduction in CPR. Futhermore, the REDY point kinetics model could not simulate the transient axial reactivity in the core. GE immediately provided calculational comparisons of REDY to test results, and attempted to demonstrate that although REDY did not accurately model some transient effects, it did provide a conservative basis for current licensing calculations.

We agreed with GE's general conclusion that REDY provides a conservative calculation for the current licensing basis transients on operating reactors. However, we also recognized that REDY's inability to accurately predict pressurization rate and axial reactivity response, limits the simulation of RPT effects. The Peach Bottom tests demonstrated (the existence of inability of REDY to simulate) a pressure wave in the steam lines.(26,27) In addition, it was noted that the power rise associated with the pressurization was significantly greater in the upper portion of the core than in the lower portion.

Quantative comparison of the tests with REDY calculations indicated that the REDY model underpredicted the pressurization rate but overpredicted the core's response to pressurization effects. Thus, there are two discrepancies between REDY simulated effects and real transient's effects. One is non-conservative and the other is conservative. It is not possible to state from these comparisons alone which effect would predominate for a given transient.

After the analysis of the test results, comparisons were made between REDY simulations and simulations using detailed steamline modeling and a time-varying axial power distribution.(28) These comparisons, although limited, indicate a trend in which REDY-based calculations conservatively predicted Δ CPR for more severe transients but underpredict ΔCPR (for a given set of input parameters) for milder transients.(28) These calculations also showed that the ΔCPR benfits derived from the RPT feature may be overpredicted by REDY when compared with the predictions of the detailed steamline and core model.

In view of this information, we decided to take no action for three reasons: (1) operating limit MCPRs are always based upon the most severe transient for each fuel type, (2) these limiting transients were sufficiently severe to be in the range where REDY-based calculations are conservative, and (3) GE was developing a more sophisticated transient simulator to accurately predict the questioned phenomena.

However, with the addition of the RPT feature, the limiting pressure and power increase transient analyses generally predict a ΔCPR in the range where REDY is less conservative. We find that full credit for the RPT effect cannot be justifdied solely on a REDY analysis.

Two alternatives have been considered to resolve this issue. The first alternative is to provide additional justification for the proposed specification. The GE ODYN code has more nodes to model steamline dynamics than REDY and also has a one-dimensional axial core neutronics model. ODYN's development has been based on first principles and verified by the Peach Bottom tests. ODYN is currently under a staff review. When approved, ODYN will be used for calculating the ΔCPR for pressurization events such as the load rejection with bypass.

Until approval, we believe that ODYN could be used to simulate the RPT effects and, thereby, provide assurance of its ΔCPR benefit. During this time, we will accept the greater ΔCPR of the ODYN and REDY calculations. Once ODYN receives generic approval, we will accept an ODYN calculation. However, the licensee has been informed that we will evaluate any other justification which the licensee submits and all applicable calculations and data which become available to us through other channels. The other alternative is to conservatively bound the ΔCPR from the REDY calculation. With the RPT modeled, the available comparison of ODYN and REDY predictions shows a ΔCPR difference of about 0.02. This calculation is for a specific BWR which is different in plant size and core loading than the Browns Ferry Units. From this information we and the licensee have agreed that a conservative bound to the REDY calculation with RPT would be assured with a 0.03 ΔCPR increase for rapid pressurization transients.

3.1.2.2.4 Abnormal Operational Transient Analysis Results

The licensee reports in the reload supplement(3) that the most limiting event for the 8x8R and P8x8R fuel types is the load rejection without bypass. This transient results in a CPR reduction of 0.15 and 0.16 for the 8x8R and P8x8R fuel assemblies, respectively, as predicted by the REDY code. Since the load rejection without bypass transient is a pressurization type event the .03 increase is applicable to these results. For the standard 8x8 fuel type the control rod withdrawal event is reported to be most limiting, with a CPR reduction of 0.21. The next most severe event for the 8x8 fuel is the load rejection without bypass with a transient ΔCPR of 0.15 as predicted by REDY. Thus the control rod withdrawal remains limiting relative to the load rejection transient even when a 0.03 CPR adjustment is applied to the latter event. In response(2) to our concern(21) on this subject the licensee has proposed to increase the fuel dependent operating limits by .03, as appropriate, on an exposure dependent basis. Since the severity of pressurization events increase toward end of cycle, the licensee has proposed(2) to add a .03 penalty to the 8x8R and P8x8R REDY predictions(3) for exposures between EOC3-2000 Mwd/T and EOC3 for establishing the required operating limits. No penalty has been proposed for exposures between BOC3 and EOC3-2000 Mwd/T for these type fuels. From our review we have concluded that the licensee has not provided an adequate basis for the proposed operating limits from BOC3 to EOC3-2000 Mwd/T. That is the intermediate exposure operating limits were not developed using the methods described in Reference 15, nor were adequate alternative evaluation bases provided. This position has been discussed with the licensee and he agreed to accept a single fuel dependent operating limit based on the end-of-cycle REDY analysis with a .03 CPR penalty added. Accordingly, based on our review of the licensee's submitted calculated results and the .03 CPR adjustment applicable to REDY calculations for pressurization transients which model the beneficial effects of the RPT feature, the licensee will be required to meet the following MCPR operating limits:

Fuel Type	MCPR Operating Limit
8x8	1.28
8x8R	1.25
P8x8R	1.26

With BF-3 operated in accordance with the above MCPR operating limits, we agree that the 1.07 SLMCPR will not be violated even in the event of the most severe abnormal operational transients.

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 3 show that the fuel type-dependent and exposure-dependent safety limit LHGRs, shown in Table 2-3 of Reference 15, 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 Analysis

3.1.3.1 ECCS Appendix K Analysis

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License, implementing the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading... "the licensee shall submit a re-evaluation of ECCS performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR Part 50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation assumptions and results.

For Cycle 2, the licensee re-evaluated the adequacy of BF-3 ECCS performance in connection with the retrofit 8x8 reload fuel design. The methods used in this analysis were previously approved by the staff. For Reload 1, we reviewed the ECCS analysis results submitted by the licensee for the Cycle 2 reload fuel and concluded that BF-3 would be in conformance with all the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50 when operated in accordance with the 8x8R MAPLHGR versus Average Planar Exposure values which appeared in the proposed plant Technical Specifications. Except for prepressurization, the Reload 2 fuel is the same design as the Reload 1 fuel.

In Reference 17, we stated that LOCA analyses previously performed and accepted for unpressurized 8x8 fuel are conservatively bounding for prepressurized fuel of that type (enrichment pattern). Accordingly we find it acceptable for the licensee to utilize the 8x8R MAPLHGR vs Average Planar Exposure technical specification limits for the reload P8x8R fuel in connection with showing compliance with the requirements of 10CFR50.46.

3.1.3.2 Control Rod Drop Accident

For Cycle 3, the key plant-specific and cycle-specific nuclear charteristics for the worst case control rod drop accident (CRDA) occurring during hot startup conditions are conservatively bounded by the values used in bounding CRDA analysis given in Reference 16.

The bounding analysis, which includes the adverse effects of fuel densification power spiking, shows that the peak fuel enthalpy will not exceed the 280 cal/gm design limit. Therefore, for Cycle 3 of BF-3, the peak fuel enthalpy associated with a CRDA from the hot startup condition will also be within the 280 cal/gm design limit.

For the worst case control rod drop accident occurring during cold startup conditions, however, not all of the key plant-specific and cycle-specific nuclear characteristics are within the values used in the generic CRDA analysis. That is, although the actual Cycle 3 Doppler coefficient and scram reactivity shape function conservatively fall within the values assumed in the bounding analysis, the accident reactivity shape function does not. Therefore, the licensee has performed a plant-specific control rod drop accident analysis applicable to BF-3 for Cycle 3. The results of this analysis, using the approved methods described in Reference 16, show that the positive reactivity insertion rate of the dropped rod is sufficiently compensated by Doppler feedback and scram reactivity effects to limit the peak energy deposition in the fuel to 278 cal/gm.

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 licensee has considered the effect of postulated fuel loading errors on bundle CPR. An analysis of the most severe fuel loading errors were performed using GE's revised analysis methods which have previously been reviewed and approved by the staff. The results show that the worst possible fuel bundle misloadings will not cause a violation of the 1.07 safety limit MCPR even when assuming the proposed OLMCPRs. These results include the application of a 0.02 penalty factor applied to the CPR results of the misoriented fuel bundle analysis, as required by our approval of the revised methods.

Thus, the required operating limit MCPRs will effectively preclude DNB related fuel failures caused by either fuel cladding overheating or cladding oxidation, which might otherwise occur because of a fuel loading error. These results are acceptable to the staff.

3.1.4 Overpressure Analysis

For Cycle 3, 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(15) 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 shows that the peak pressure at the bottom of the reactor vessel does not exceed 1300 psig for worst case end-of-cycle conditions, even when assuming the effects of one failed safety valve. This is acceptable to the staff.

3.1.5 Thermal-Hydraulic Stability

A thermal-hydraluic stability analysis was performed for this reload using the methods described in Reference 15. The results show that the fuel type 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 0.273 (8x8R/P8x8R), 0.383 (8x8) and 0.79 respectively. These predicted decay ratios are all well below the 1.0 Ultimate Performance Limit decay ratio proposed by GE.

The staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition. This condition could be reached during an operational transient from high power if the plant were to sustain a trip of both recirculation pumps without a reactor trip. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as reload fuel designs change. The staff concerns relate to both the consequences of operating with a decay ratio of 1.0 and the capability of the analytical methods to accurately predict decay ratios. The General Electric Company is addressing these staff concerns through meetings, topical reports and a stability test program. It is expected that the test results and data analysis, as presented in a final test report, will aid considerably in resolving the staff concerns.

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 3. On the basis of the foregoing, the staff considers the thermal-hydraulic stability of BF-3 during Cycle 3 to be acceptable.

3.1.6 Physics Startup Testing

The licensee will peform a series of physics startup tests and provide assurance that the conditions assumed for the transient and accident analysis calculations will be met during Cycle 3. The test will verify that the core has been loaded as intended, that the incore monitoring system is functioning as expected and that the process computer has been reprogrammed to properly reflect changes associated with the reload. The test program is consistent with that previously found acceptable for BF-3. We find this test program acceptable for Cycle 3.

3.1.7 Technical Specifications

The proposed Technical Specifications for Cycle 3 operation of BF-3 include revised operating limit minimum critical power ratios (OLMCPRs) for each fuel type in the core. As discussed in Section 3.1.2.2 herein, the fuel-dependent operating limit MCPRs proposed by the licensee have been adjusted, with his agreement, to account for possible excess end-of-cycle recirculation pump trip benefits calculated by the REDY Code for pressurization type transients. Thus the OLMCPRs agreed to by the licensee and the staff for the entire third cycle are 1.28, 1.25 and 1.26 for the 8x8, 8x8R and P8x8R fuel types respectively. These MCPR operating limits are acceptable. Additionally the licensee has proposed MAPLHGR vs Average Planar Exposure limits for the prepressurized reload 8x8R, which are the same as the unpressurized 8x8R fuel. As discussed in Section 3.1.3.1, this is acceptable.

3.2 LPCI Modification

3.2.1 Codes and Methods

The reanalysis of the Loss of Coolant Accident for Browns Ferry Unit No. 3 with LPCI modifications was performed using a General Electric evaluation model which is generally described in Reference 30. The model used for this analysis also includes previously approved[31] model changes[32,33] made to the REFLOOD and CHASTE computer codes. Additionally, other previously approved[31] model changes[34,35,36] which take into account the beneficial effects of alternate reflood flow paths (via holes drilled into the fuel assembly lower tie plates) have been included in the reanalysis. In summary therefore, the LOCA analysis of BF-3 with LPCI modifications was performed using approved calculational models and methods.

3.2.2 Analysis Results

3.2.2.1 Lead Plant Reference

In support of the BF-3 LOCA reanalysis, the licensee has referenced a previously approved[37] Loss-of-Coolant Accident anlaysis[38] performed for the James A. FitzPatrick Nuclear Power Plant (JAF). James A. FitzPatrick is the "lead plant" BWR/4 with LPCI loop-selection-logic removed. The lead plant reference provides detailed and expanded analysis results and documentation which justifies the extent to which break size, break location and single failure combinations must be considered when evaluating the LOCA consequences of specific BWR/4s with LPCI modifications.

The results from Reference 38 show that the most limiting breaks occur in the recirculation piping when failure of the LPCI injection valve is assumed. In particular the results for JAF show that the most limiting break location is the recirculation discharge line rather than the larger diameter recirculation suction line and is due to the effects of the LPCI modification associated with JAF, BF-3 and other BWR/4s. The basis for the discharge break being more limiting than the suction break is discussed in Section 2.2 herein. Furthermore, the break spectrum (i.e. peak cladding temperature vs break size)for the lead plant shows that a break in the discharge piping have a break area equal to approximately 80% of the area associated with the largest discharge line break is limiting. The reason the limiting break size is less than 100% of the maximum possible limiting location break area is provided in our safety evaluation[37] for the lead plant.

It should be noted, however, that for plants with LPCI modification, such as JAF and BF-3, the peak cladding temperature resulting from a recirculation discharge line break and the PCT resulting from a recirculation suction line break are very nearly the same. That is, small differences in reactor system design (e.g., flow areas inside the vessel, active fuel and bypass regions; exact pipe sizes; exact design of the LPCI system) determine which break location and break size is actually limiting for any particular plant. Based on our review of the lead plant reference we conclude that minor differences between the lead plant and BF-3 resulting in a change in worst break location or break size between these plants would not significantly effect our conclusions regarding the break spectrum, the worst break location (on the recirculation line piping as opposed to other pipes such as the feedwater or core spray lines) or worst single failure. Accordingly, we conclude that the James A. Fitzpatrick Loss of Coolant Accident analysis is an acceptable lead plant reference for BF-3.

3.2.2.2 Plant Specific Results

Supplementing the lead plant analysis, the licensee has submitted additional ECCS performance calculations[6] which specifically model the BF-3 plant with LPCI loop-selection-logic removed. These plant-specific analyses provide detailed results for the spectrum of postulated breaks occurring in the BF-3 recirculation suction and discharge piping with assumed LPCI injection valve failure. Similar to the lead plant analysis, the LOCA analysis performed for BF-3 shows that the most limiting break location is the recirculation discharge line. In the case of BF-3, the limiting break area (i.e. the design basis accident) is approximately 66% as large as the largest discharge line break assuming failure of the injection valve in the intact loop. Formerly, the BF-3 (with LPCI loop-selection-logic) LOCA analysis results(29) showed that the limiting break size , break location and single failure condition was the complete severence of the suction line piping and LPCI injection valve failure. The observed shift of the limiting break location from the suction line to the discharge line is not unexpected and is prinicply due to the LPCI modification described earlier. Complete severence of the recirculation suction line is now the second most limiting break size and location for BF-3.

The results for BF-3 have also been compared with the most recently accepted LOCA conformance calculations(39, 40) performed for Browns Ferry Units 1 and 2. Both units were analyzed with LPCI loop-selection-logic removed. The comparison shows that the limiting break size and location is different for BF-3 than the limiting break size and location for Units 1 and 2. For Units 1 and 2 complete severence of the larger diameter suction line piping is limiting while a break in the discharge line piping having a break area equal to 66% of the area associated with the complete severence of the discharge pipe was shown(39, 40) to be the second most limiting size and location.

The fact that the worst break size and location is different among these virtually identical LPCI-modified BWR/4s, can be traced to the different fuel types (including number of fuel assemblies with drilled lower tie plates) loaded in the respective cores. The cores of Browns Ferry Units 1 and 2 contain both 7x7 and 8x8 fuel types (and not all fuel assemblies drilled) while the BF-3 core contains only 8x8 fuel types (with all fuel assemblies drilled). On a system level, cores with 7x7 fuel tend to reflood up to the high power axial plane somewhat later due to the limiting effects of counter-current flow on the core spray contribution to vessel reflood rate. This can be seen for example by comparing the BF-1 and BF-3 dryout, uncovery and reflood times for the same discharge breaks. Although the dry out and uncovery times are about the same for the same breaks the reflood time is significantly later for BF-1 (some 7x7 fuel) than for BF-3 (all 8x8 fuel). The other important effect of fuel type relates to the differences in the amount of stored energy which can be removed by the time of boiling transition (loss of good heat transfer from fuel rod to coolant). For 7x7 fuel it takes approximately 25 seconds to remove the stored heat via nucleate boiling while it takes only about half this time for 8x8 fuels. Accordingly, the PCT of cores with 7x7 fuel are more sensitive to differences in boiling transition times associated with different break sizes and break locations. More heat will still be stored in the 7x7 fuel to heat up its cladding than in the 8x8 fuel to heat up its cladding once nucleate boiling heat transfer is lost. That is although the dryout times of the fuel in the BF-1 (7x7) and BF-3 (8x8) cores are about the same for a given break in the spectrum of pipe breaks, the cladding temperature at the time of hot node uncovery is substantially higher for the Unit 1 fuel than for the Unit 3 fuel due to the greater stored energy still contained in the 7x7 fuel at fuel dryout. In summary, therefore, the combined thermal and hydraulic effcts of fuel type on vessel reflood and cladding heat-up phenomenon results in a shift of the worst break location and size from the largest suction line break (BF-1 and BF-2) to an intermediate size break in the discharge line (BF-3).

The PCT for the limiting break size and location was conservatively calculated with the added conservatism of applying the 80% discharge break LAMB-SCAT Code (earlier boiling transition time) results to the 66% discharge break SAFE/REFLOOD Code (uncovery time) results. Thus any slight non-conservatism (at most 2°F to 5°F) due to the possibility that the PCT occurs slightly above or below the limiting break size is more than compensated for by this unrequired (extra) conservatism in the CHASTE (heatup) analysis. The CHASTE (fuel cladding heatup) reanalysis was performed for each of the initial and reload fuel types assuming the same respective tables of Maximum Average Planar Linear Heat Generation Rates (MAPLHGR) vs. Average Planar Exposure as those used in connection with the previously accepted ECCS conformance analysis[29] performed for BF-3 with LPCI loop-selection-logic. Accordingly, since ECCS performance is improved relative to the formerly limiting suction line break, the overall peak cladding temperature for the worst break location, break size, single failure, fuel type and exposure has been lowered. Formerly the licensee reported[29] a PCT of 1963°F for BF-3 with LPCI loop selection logic. With the LPCI modification the licensee now reports[6] a PCT of 1790°F. Additionally, operation of BF-3 at these MAPLHGR values results in a local cladding oxidation of less than 1% and a core wide metal-water reaction of .05% for the limiting break size with LPCI injection valve failure (i.e. the DBA). These calculated values also meet the requirements specified in 10CFR50.46.

With regard to small break consequences, the licensee states[6] that the generic results reported in Reference 41 are applicable to BF-3. The bounding analysis referenced provides the PCT for the worst size small break occurring in the recirculation discharge piping of a BWR/4 with LPCI modifications. The analysis assumes a direct current power source failure (worst single failure for a small break). For this assumed failure, Reference 41 indicates that one LPCI pump, one of the two core spray systems (i.e. two 50% capacity pumps) and the automatic depressurization system (ADS) are available to mitigate the accident. The generic analysis shows that PCT will be less than 2200° even when taking credit for only four of the six ADS valves.

The effects of a DC power source failure on the consequences of small and large breaks as reported in Reference 41 are also being generically reviewed by the staff. Although we have not yet completed our review of Reference 41, based on the systems stated to be available with a DC power source failure, we believe that there will not be changes to the generic study which could make the results of a plant-specific small break LOCA become more limiting than the worst large break LOCA.

3.2.3 Overall Evaluation of LPCI Modification

We have reviewed the analysis of emergency core cooling system performance submitted by TVA for BF-3 with the proposed LPCI-modifications and conclude that all of the requirements of 10 CFR 50.46 and Appendix K to 10CFR50.46 will be met when the reactor is operated in accordance with the MAPLHGR versus Average Planar Exposure values given in Tables 3.1.5-1 through 3.5.I-3 of Reference 3.

3.3 Modification of RWCU System Piping

As discussed in Section 1.3 of this safety evaluation, the reactor water cleanup (RWCU) System piping was modified so that the return flow is distributed equally among the feedwater lines. This modification, which has been recommended by NRC and GE, will allow the higher temperature RWCU return water to be mixed at low flow rates with the large volume of feedwater, thereby lessening the thermal cycling on the feedwater nozzle and the consequent thermal fatigue. The modification required the addition of a check valve. The change to the Technical Specifications is to add this check valve to Table 3.7.G, "Check Valves on Drywell Influent Lines" as one of the penetration and isolation valves which must be included in the containment leak test conducted each operating cycle. In partial response to the USNRC Office of Inspection and Enforcement's Bulletin 79-08, during this outage TVA added new additional hydrogen and oxygen sensing lines into the primary containment. These lines are isolatable by the usual inboard and outboard isolation valves and outboard block valves. Since these valves must also be included in the periodic containment leak test, they were added to Table 3.7.D "Primary Containment Isolation Valves". We conclude that these plant modifications improve plant safety and that the proposed changes to the Technical Specifications are appropriate and acceptable.

3.4 Replacement of Two Safety-Relief Valves

Raising the lift settings of the subject dual action SRVs affects those plant transients which result in an increase in reactor system pressure sufficient to cause safety/relief valve actuation of the highest pressure setpoint groups. Accordingly, the licensee has reanlayzed the most severe pressurization transients. The limiting events for Browns Ferry Unit 3 are generator load rejection with bypass system failure (LR w/o BP) and main steam isolation valve (MSIV) closure with indirect high flux scram (vessel overpressure protection analysis).

3.4.1 Abnormal Operation Transients

For BF-3 the largest change in bundle critical power ratio (CPR) for the retrofit 8x8R fuel types is caused by the load rejection without bypass pressurization event. This event, which is initiated by fast closure of the turbine control valves, causes a rapid collapse of moderator voids in the core. The collapse of the voids causes a significant addition of positive reactivity to the core, which results in a pronounced neutron flux spike, and a subsequent rise in core heat flux. Before core heat flux can rise substantially, the event is terminated by a reactor scram and prompt recirculation pump trip caused by a fast closure trip signals developed at the turbine control valves.

The licensee reanalyzed this event using methods which are the same as those used for the most recent BF-3 reload safety analysis.[3] For the revised thermal margin analyses, in addition to the assumed 25 psi increase in valve lift pressure, the safety/relief valves capacities of the two Crosby valves were modeled to reflect the somewhat lesser steam relief rate of these valves compared to the two Target Rock valves they will replace.

The reanalyses[14] shows that the proposed change in SRV setpoint would not result in a significant increase in the critical power ratio for the LWr/oBP event when compared with the most recent analyses[3]. Accordingly. the staff finds it acceptable to retain the present operating limit minimum CPRs and that the proposed SRV setpoint change is acceptable with regard to fuel thermal margin considerations.

The licensee also reanalyzed the load rejection without bypass event from the viewpoint of peak transient reactor system pressure. For calculating peak pressure the plant transient analysis used the same models and methods as for the fuel thermal margin analyses, including credit for the prompt recirculation pump trip feature. The results show that a peak transient pressure increases by approximately 12 psi with the proposed change setpoint change. However, the reanalysis shows that a margin greater than 25 to the lift pressure of code safety valves (1250 psig) is still available. This is acceptable.

3.4.2 Overpressurization Analysis

The licensee has also provided the results of a bounding overpressurization analysis, to demonstrate that an adequate margin exists to the ASME Code allowable pressure, with the proposed revised SRV settings. The ASME Code allows peak transient pressures up to 100% of vessel design pressure, i.e., 1375 psig. The most limiting event was taken to be the closure of all main steam isolation valves with a reactor trip on high neutron flux which was the same event analyzed for the most recent reload. The analysis conservatively assumed an initial reactor power of 104.5% and 100% core flow, an end-of-cycle scram reactivity insertion rate curve and all safety/relief valves operative. As for the load rejection without bypass, the Crosby valve characteristics modeled reflected the lower relief capacities and the higher opening pressure setpoint of these valves. The reanlaysis included a (ATWS) recirculation pump trip on high reactor pressure since the attendant flow reduction has the effect of increasing peak transient pressure. The results show that the substitution of the two Crosby valves for the Target-Rock valves increases the peak pressure at the bottom of the reactor vessel by approximately 13 psig leaving a margin of 82 psi to the 1375 psig Code allowable safety limit. Furthermore, a generic analysis,[42] showing the sensitivity of peak transient pressure to total relief capacity, when applied to Browns Ferry 3, shows that the failure-to-open of one SRV would cause pressure to increase by less than 20 psi. Therefore, the maximum transient reactor vessel pressure for MSIV closure at end-of-cycle assuming an indirect high neutron flux scram and one failed safety valve will still show ample margin to the pressure safety limit. These results are acceptable to the staff.

4.0 Environmental Considerations

We have determined that this 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 this 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 30, 1979

6.0 References

1. Tennessee Valley Authority letter (L. Mills) to USNRC (H. Denton) dated August 6, 1979.

2. Tennessee Valley Authority letter (L. Mills) to USNRC (H. Denton) dated October 25, 1979.

3. "Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Power Station Unit 3, Reload 2", NEDO-24199 dated June 1979.

4. "Proposed Changes to Browns Ferry Unit 3 Technical Specifications" submitted as Enclosure 1 to TVA letter (L. Mills) to USNRC (H. Denton) dated August 6, 1979.

5. Tennessee Valley Authority letter (L. Mills) to USNRC (H. Denton) dated September 26, 1979.

6. "Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3", NEDO-24194A dated July 1979.

7. "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, NUREG-0313, dated July 1977 and Revision 1 to NUREG-0313 dated October 1979.

8. "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of LWR Plants", NUREG-0531 dated February 1979.

9. General Electric Company Service Information Letter No. 208, Revision 1, "Minimizing Feedwater Nozzle Thermal Duty", dated October 1978.

10. Tennessee Valley Authority letter (L. M. Mills) to USNRC (H. R. Denton) dated September 26, 1979.

11. USNRC Office of Inspection and Enforcement, IE Circular No. 79-18 "Proper Installation of Target Rock Safety-Relief Valves.

12. USNRC Office of Inspection and Enforcement, IE Bulletin 74-4 and 74-4a, "Malfunction of Target Rock Safety Relief Valves" issued March 22, 1974.

13. Tennessee Valley Authority letter (L. M. Mills) to USNRC (H. R. Denton) dated October 10, 1979.

14. Revised Appendix A, "Analysis For Alternate Safety-Relief Valves" dated September 25, 1979 to NEDO-24199, "Browns Ferry Unit 3 Reload 2 Supplemental Licensing Submittal" dated June 1979.

15. "Generic Reload Fuel Application," NEDE-24011-P-A, dated August 1978 including Appendix D dated August 1978.

16. USNRC letter (D. Eisenhut) to General Electric (R. Gridley) dated May 12, 1978 transmitting "Safety Evaluation for the General Electric Topical Report Generic Reload Fuel Application (NEDE-244011-P) dated April 1978.

17. USNRC letter (T. Ippolito) to General Electric (R. Gridley) dated April 1979 transmitting "Safety Evaluation Supplement for an Amendment dated August 1978 to the General Electric Topical Report Generic Reload Fuel Application."

18. "Status Report on the Licensing Topical Report" 'General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel' NEDO-20360 Revision 1 and Supplement 1 by Division of Technical Review, ONRR, USNRC, April 1975.

19. "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," Revision 1 and Supplement 1 April 1974, NEDO-20360.

20. "General Electric BWR Thermal Analysis Basis (GETAB) Date, Correlation and Design Application," NEDO-10958, November 1973.

21. USNRC letter (T. Ippolito) to TVA (H. Parris) dated September 19, 1979.

22. General Electric letter (R. Engle) to USNRC (D. Eisenhut and R. Tedesco), dated March 30, 1979.

23. General Electric letter (R. Gridley) to USNRC (D. Eisenhut and D. Ross), dated October 5, 1978, transmitting "General Electric Information NEDE-24131, Basis for 8x8 Retrofit Fuel Thermal Analysis Application."

24. "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, February 1973.

25. "Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2," Carmichael, L. A. and Niemi, R. O., EPRI-NP-564, June 1978.

26. General Electric letter (R. Engel) to USNRC dated July 11, 1977.

27. General Electric letter (E. Fuller) to USNRC dated October 25, 1977.

28. "Input of One Dimensional Transient Model on Plant Operating Limits," enclosure of letter, E. D. Fuller (GE) to USNRC dated June 26, 1978.

29. "Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3," NEDO-24127, June 1978.

30. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K" NEDO-20566 submitted August 1974, and General Electric Refill Reflood Calculation (Supplement to the SAFE Code Description) transmitted to the USAEC by letter, G. L. Gyory (GE) to Victor Stello, Jr., (NRC) dated December 20, 1974.

31. Safety Evaluation for General Electric ECCS Evaluation Model Modification transmitted by letter from K. R. Goller (NRC) to G. G. Sherwood (GE), dated April 12, 1977.

32. General Electric letter (A. Levine) to USNRC (D. Ross) dated January 27, 1977 transmitting General Electric (GE) Loss-of-Coolant Accident (LOCA) Analysis Model Revisions-Core Heatup Code CHASTE05.

33. General Electric letter (A. Levine) to USNRC (D. Vassallo), dated March 14, 1977 transmitting Request for Approval for Use of Loss-of-Coolant Accident (LOCA) Evaluations Model Code REFLOOD05.

34. Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibrations, Supplement 1, NEDE-21156-1, September 1976.

35. Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibrations, Supplement 2, NEDE-21156-2, January 1977.

36. General Electric letter (R. Engel) to USNRC (V. Stello), Answers to NRC Questions on NEDE-21156-2 January 24, 1977.

37. USNRC letter (R. Reid) to PASNY (G. Berry) dated September 16, 1977 transmitting Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 30 to Facility Operating License No. DPR-59.

38. PASNY letter (G. Berry) to USNRC (R. Reid) dated July 29, 1977 transmitting Loss-of-Coolant Accident Analysis Report for James A. Fitzpatrick Nuclear Power Plant (Lead Plant), NEDO-21662-2, July 1977.

39. "Loss of Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 2" NEDO-24088-1, February 1978.

40. "Loss of Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 1" NEDO-24056, September 1977.

41. General Electric Letter (R. Engel) to USNRC (P. Check), "DC Power Source Failure for BWR/3 and 4," dated November 1, 1978.

42. GE letter((I. Stuart) to USNRC (V. Stello) "Code Overpressure Protection Analysis - Sensitivity of Peak Vessel Pressure to Valve Operability," dated December 23, 1975.

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. 28 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 third fuel cycle, (2) reflect facility modifications made during the current refueling outage to eliminate the low pressure coolant injection loop selection logic (an action which the Commission required to be accomplished and the design for which the Commission approved in Amendment No. 23 to Operating License No. DPR-68 dated May 11, 1979), (3) reflect rerouting of the reactor water cleanup system piping to reduce thermal cycling on the feedwater nozzles (and thus provide increased margin against the initiation and propagation of cracks in these nozzles), and (4) reflect replacement of two of the 11 safety-relief valves with valves of an improved design that will provide a slightly increased simmer margin (i.e., the two replacement valves will be set to relieve at 1150 psig rather than 1125 psig).

The application for the 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 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 application for amendment dated August 6, 1979, as supplemented by two letters dated September 26, 1979 and letters dated October 10, 1979 and October 25, 1979, (2) Amendment No. 28 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 Comisssion, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

FOR THE NUCLEAR REGULATORY COMMISSION

T. Ippolito

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

Dated at Bethesda, Maryland this 30th day of November 1979.