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

NOV 18 1978

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Tennessee Valley Authority
ATTN: Mr. N. B. Hughes
Manager of Power
830 Power Building
Chattanooga, Tennessee 37401

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 45, 41, and 18 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. These amendments consist of changes to the Technical Specifications in response to your request of August 3, 1978 (TVA BFNP TS 113) as supplemented by letter dated October 20, 1978.

Amendment No. 18 changes the Technical Specifications to incorporate the limiting conditions for operation associated with the initial 2000 megawatt days per tonne of fuel exposure during the second fuel cycle for Unit No. 3. As agreed with your staff, TVA will submit a reanalysis of transients for the end of cycle 2 to evaluate operation of Unit No. 3 beyond 2000 MWD/t fuel exposure. These amendments also change the Technical Specifications to incorporate minor changes in the arrangements for leak testing certain primary containment isolation and check valves.

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

78120500 H

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Enclosures:

1. Amendment No. 45 to DPR-33
2. Amendment No. 41 to DPR-52
3. Amendment No. 18 to DPR-68
4. Safety Evaluation
5. Notice

OFFICE	cc w/enclosures See next page	DOR:ORB-3 SSheppard	DOR:ORB-3 RJCClark/esp	OELD Reis	DOR:ORB-3 Tippolito
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DATE		11/20/78	11/18/78	11/18/78	11/18/78



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

NOVEMBER 18 1978

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

Tennessee Valley Authority
ATTN: Mr. N. B. Hughes
Manager of Power
830 Power Building
Chattanooga, Tennessee 37401

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 45, 41, and 18 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. These amendments consist of changes to the Technical Specifications in response to your request of August 3, 1978 (TVA BFNP TS 113) as supplemented by letter dated October 20, 1978.

Amendment No. 18 changes the Technical Specifications to incorporate the limiting conditions for operation associated with the initial 2000 megawatt days per tonne of fuel exposure during the second fuel cycle for Unit No. 3. As agreed with your staff, TVA will submit a reanalysis of transients for the end of cycle 2 to evaluate operation of Unit No. 3 beyond 2000 MWD/t fuel exposure. These amendments also change the Technical Specifications to incorporate minor changes in the arrangements for leak testing certain primary containment isolation and check valves.

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Sincerely,

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

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Enclosures:

1. Amendment No. 45 to DPR-33
2. Amendment No. 41 to DPR-52
3. Amendment No. 18 to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures:
See next page

P

Tennessee Valley Authority

- 2 -

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 45
License No. DPR-33

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendments by Tennessee Valley Authority (the licensee) dated August 3, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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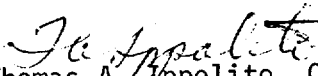
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility License No. DPR-33 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 45, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION


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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: **NOVEMBER 18 1978**

ATTACHMENT TO LICENSE AMENDMENT NO. 45

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

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

260/261
264/265

2. Marginal lines indicate revised area. Overleaf pages are provided for convenience.

TABLE 3.7.D (Continued)

<u>Valves</u>	<u>Valve Identification</u>	<u>Test Medium</u>	<u>Test Method</u>
69-1	RWCU Supply	Water ⁽²⁾	Applied between 69-1, 69-500 and 10-505
69-2	RWCU Supply	Water ⁽²⁾	Applied between 69-2, 69-500 and 10-505
71-2	RCIC Steam Supply	Air ⁽¹⁾	Applied between 71-2 and 71-3
71-3	RCIC Steam Supply	Air ⁽¹⁾	Applied between 71-2 and 71-3
71-39	RCIC Pump Discharge	Water ⁽²⁾	Applied between 3-66, 3-568, 69-579, 71-39, and 85-576
73-2	HPCI Steam Supply	Air ⁽¹⁾	Applied between 73-2 and 73-3
73-3	HPCI Steam Supply	Air ⁽¹⁾	Applied between 73-2 and 73-3
73-44	HPCI Pump Discharge	Water ⁽²⁾	Applied between 3-67, 3-554, and 73-44
74-47	RHR Shutdown Suction	Water ⁽²⁾	Applied between 74-47, 74-754, 74-49, and 74-661
74-48	RHR Shutdown Suction	Water ⁽²⁾	Applied between 74-48, 74-661, and 74-49.
74-53	RHR LPCI Discharge	Water ⁽²⁾	Applied between 74-53 and 74-55
74-57	RHR Suppression Chamber Spray	Water ⁽²⁾	Applied between 74-57, 75-58, and 74-59
74-58	RHR Suppression Chamber Spray	Water ⁽²⁾	Applied between 74-57, 74-58, and 74-59
74-60	RHR Drywell Spray	Water ⁽²⁾	Applied between 74-60 and 74-61
74-61	RHR Drywell Spray	Water ⁽²⁾	Applied between 74-60 and 74-61
74-67	RHR LPCI Discharge	Water ⁽²⁾	Applied between 74-67 and 74-69
74-71	RHR Suppression Chamber Spray	Water ⁽²⁾	Applied between 74-71, 74-72, and 74-73
74-72	RHR Suppression Chamber Spray	Water ⁽²⁾	Applied between 74-71, 74-72, and 74-73
74-74	RHR Drywell Spray	Water ⁽²⁾	Applied between 74-74 and 74-75

TABLE 3.7.D (Continued)

<u>Valves</u>	<u>Valve Identification</u>	<u>Test Medium</u>	<u>Test Method</u>
14-75	RHR Drywell Spray	Water ⁽²⁾	Applied between 74-74 and 74-75
14-77	RHR Head Spray	Water ⁽²⁾	Applied between 74-77 and 74-78
14-78	RHR Head Spray	Water ⁽²⁾	Applied between 74-77 and 74-78
14-661/662	RHR Shutdown Suction	Water ⁽²⁾	Applied between 74-660 and 74-661/662
15-25	Core Spray Discharge	Water ⁽²⁾	Applied between 75-25 and 75-27
15-53	Core Spray Discharge	Water ⁽²⁾	Applied between 75-53 and 75-55
15-57	Core Spray to Auxiliary Boilers	Water ⁽²⁾	Applied between 75-57 and 75-58
15-58	Core Spray To Auxiliary Boilers	Water ⁽²⁾	Applied between 75-57 and 75-58
17	Drywell/Suppression Chamber Nitrogen Purge Inlet	Nitrogen ⁽¹⁾	Applied between 76-17, 76-18, 76-19
76-18	Drywell Nitrogen Purge Inlet	Nitrogen ⁽¹⁾	Applied between 76-17, 76-18, 76-19
76-19	Suppression Chamber Purge Inlet	Nitrogen ⁽¹⁾	Applied between 76-17, 76-18, 76-19
76-24	Drywell/Suppression Chamber Nitrogen Purge Inlet	Air ⁽¹⁾	Applied between 64-17, 64-18, 64-19, and 76-24
77-2A	Drywell Floor Drain Sump	Water ⁽²⁾	Applied between 77-2A and 77-2B
77-2B	Drywell Floor Drain Sump	Water ⁽²⁾	Applied between 77-2A and 77-2B
77-15A	Drywell Equipment Drain Sump	Water ⁽²⁾	Applied between 77-15A and 77-15B
77-15B	Drywell Equipment Drain Sump	Water ⁽²⁾	Applied between 77-15A and 77-15B
90-254A	Radiation Monitor Suction	Air ⁽¹⁾	Applied between 90-254A, 90-254B, and 90-255
90-254B	Radiation Monitor Suction	Air ⁽²⁾	Applied between 90-254A, 90-254B, and 90-255
-255	Radiation Monitor Suction	Air ⁽²⁾	Applied between 90-254A, 90-254B, and 90-255

**TABLE 3.7.G
CHECK VALVES ON DRYWELL INFLUENT LINES**

<u>Valves</u>	<u>Valve Identification</u>	<u>Test Medium</u>	<u>Test Method</u>
3-554	Feedwater	Water	Applied between 3-67, 3-554, and 73-45.
3-558	Feedwater	Water	Applied between 3-67 and 3-558
3-568	Feedwater	Water	Applied between 3-66, 3-568, 71-40, 69-579, and 85-576
3-572	Feedwater	Water	Applied between 3-66 and 3-572
63-525	Standby Liquid Control Discharge	Water	Applied between 63-525 and 63-527
63-526	Standby Liquid Control Discharge	Water	Applied between 63-526 and 63-527
69-579	RWCU Return	Water	Applied between 3-66, 3-568, 69-579, 71-40, and 85-576
71-40	RCIC Pump Discharge	Water	Applied between 3-66, 3-568, 69-579, 71-40, and 85-576.
73-45	HPCI Pump Discharge	Water	Applied between 3-67, 3-554 and 73-45
74-54	RHR LPCI Discharge	Water	Applied between 74-54 and 74-55
74-68	RHR LPCI Discharge	Water	Applied between 74-68 and 74-69
75-26	Core Spray Discharge	Water	Applied between 75-26 and 75-27
75-54	Core Spray Discharge	Water	Applied between 75-54 and 75-55
85-576	CKD Hydraulic Return	Water	Applied between 3-66, 3-568, 71-40, 69-579, and 85-576.



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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 41
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendments by Tennessee Valley Authority (the licensee) dated August 3, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 41, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: **NOVEMBER 18 1978**

ATTACHMENT TO LICENSE AMENDMENT NO. 41

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

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

259/260
263/264

2. Marginal lines indicate revised area. Overleaf pages are provided for convenience.

TABLE 3.7.D (Continued)

<u>Valves</u>	<u>Valve Identification</u>	<u>Test Medium</u>	<u>Test Method</u>
43-28B	RHR Suppression Chamber Sample Lines	Water ⁽²⁾	Applied between 74-226 and 43-28B
43-29A	RHR Suppression Chamber Sample Lines	Water ⁽²⁾	Applied between 74-227 and 43-29A
43-29B	RHR Suppression Chamber Sample Lines	Water ⁽²⁾	Applied between 74-227 and 43-29B
64-17	Drywell and Suppression Chamber air purge inlet	Air ⁽¹⁾	Applied between 64-17, 64-18, 64-19, and 76-24
64-18	Drywell air purge inlet	Air ⁽¹⁾	Applied between 64-17, 64-18, 64-19, and 76-24
64-19	Suppression Chamber air purge inlet	Air ⁽¹⁾	Applied between 64-17, 64-18, 64-19, and 76-24
64-20	Suppression Chamber vacuum relief	Air ⁽¹⁾	Applied between 64-20 and 64-(ck)
64-(ck)	Suppression Chamber vacuum relief	Air ⁽¹⁾	Applied between 64-20 and 64-(ck)
64-21	Suppression Chamber vacuum relief	Air ⁽¹⁾	Applied between 64-21 and 64-(ck)
64-(ck)	Suppression Chamber vacuum relief	Air ⁽¹⁾	Applied between 64-21 and 64-(ck)
64-29	Drywell main exhaust	Air ⁽¹⁾	Applied between 64-29, 64-30, 64-32, 64-33 and 84-19
64-30	Drywell main exhaust	Air ⁽¹⁾	Applied between 64-29, 64-30, 64-32, 64-33 and 84-19
64-31	Drywell exhaust to Standby	Air ⁽¹⁾	Applied between 64-31, 64-141, 84-20 and 64-140
64-32	Suppression Chamber Main Exhaust	Air ⁽¹⁾	Applied between 64-32, 64-33, 64-29, 64-30 and 84-19
64-33	Suppression Chamber Main Exhaust	Air ⁽¹⁾	Applied between 64-32, 64-33, 64-29, 64-30 and 84-19
64-34	Suppression Chamber to Standby Gas Treatment	Air ⁽¹⁾	Applied between 64-34, 64-141 and 64-139

TABLE 3.7.E
 SUPPRESSION CHAMBER INFLUENT LINES
 TOP-CHECK GLOBE ISOLATION VALVES

<u>Valves</u>	<u>Valve Identification</u>	<u>Test Medium</u>	<u>Test Method</u>
71-14	RCIC Turbine Exhaust	Water	Apply between 71-14 and 71-580
71-32	RCIC Vacuum pump Discharge	Water	Apply between 71-32 and 71-592
73-23	HPCI Turbine Exhaust	Water	Apply between 73-23 and 73-603
73-24	HPCI Turbine Exhaust Drain	Water	Apply between 73-24 and 73-609

TABLE 3.7.F
 CHECK VALVES ON SUPPRESSION CHAMBER INFLUENT LINES

<u>Valves</u>	<u>Valve Identification</u>	<u>Test Medium</u>	<u>Test Method</u>
71-580	RCIC Turbine Exhaust	Water	Apply between 71-14 and 71-580
71-592	RCIC Vacuum Pump Discharge	Water	Apply between 71-32 and 71-592
73-603	HPCI Turbine Exhaust	Water	Apply between 73-23 and 73-603
73-609	HPCI Exhaust Drain	Water	Apply between 73-24 and 73-609



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. 18
License No. DPR-68

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendments by Tennessee Valley Authority (the licensee) dated August 3, 1978, as supplemented by letter dated October 20, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility License No. DPR-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. 18, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: NOVEMBER 18 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 18

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

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

vii	18	27	97	161	181
viii	20	28	110	165	182
9	21	29	111	166	192
10	22	30	123	167	224
15	23	34	134	175	225
16	24	72	136	176	273
17	26	75	143	178	281
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2. Marginal lines indicate revised area.

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1.1 FUEL CLADDING INTEGRITYApplicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specifications

- A. Reactor Pressure > 800 psia and Core Flow > 10% of Rated.

When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit.

2.1 FUEL CLADDING INTEGRITYApplicability

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

Objective

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

Specification

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

- a. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$$S \leq (0.66W + 54\%)$$

where:

S = Setting in percent of rated thermal power (3293 Mwt)

1.1 FUEL CLADDING INTEGRITY2.1 FUEL CLADDING INTEGRITY

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

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

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

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

(NOTE: These settings assume operation within the basic thermal hydraulic design criteria. These criteria are $LHGR \leq 13.4 \text{ kW/ft}$ and $MCPR$ within limits of specification 3.5.K.

1.1 BASES: FUEL CLADDING INTEGRITY SAFETY LIMIT

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

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

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

uncertainties employed in deriving the safety limit are provided at the beginning of each fuel cycle.

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

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

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

In addition to the boiling transition limit ($M CPR = 1.07$) operation is constrained to a maximum LHGR of 13.4 kW/ft. This limit is reached when the Core Maximum Fraction of Limiting Power Density equals 1.0 ($CMFLPD = 1.0$). For the case where $CMFLPD$ exceeds the Fraction of Rated Thermal Power, operation is permitted only at less than 100% of rated power and only with reduced APRM scram settings as required by specification 2.1.A.1.

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

For the fuel in the core during periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If water level

should drop below the top of the fuel during this time, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation. As long as the fuel remains covered with water, sufficient cooling is available to prevent fuel clad perforation.

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 1, NEDO-24128, June 1978

2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Browns Ferry Nuclear Plant have been analyzed throughout the spectrum of planned operating conditions up to the design thermal power condition of 3440 Mwt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the FSAR. In addition, 3293 Mwt is the licensed maximum power level of Browns Ferry Nuclear Plant, and this represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity has been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

For analyses of the thermal consequences of the transients a MCPR of *** is conservatively assumed to exist prior to initiation of the transients. This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

*** See Section 3.5.K.

because it provides adequate margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of CMFLPD and FRP. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the CMFLPD exceeds FRP.

Analyses of the limiting transients show that no scram adjustment is required to assure $MCPR > 1.07$ when the transient is initiated from $MCPR > ***$.

2. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer and the Rod Sequence Control System. Worth of individual rods is very low in a uniform rod pattern. Thus, all of possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

3. IRM-Flux Scram Trip Setting

The IRM System consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a

*** See Section 3.5.K.

5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be at 120 divisions for that range; likewise, if the instrument was on range 5, the scram setting would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. A scram at 120 divisions on the IRM instruments remains in effect as long as the reactor is in the startup mode. The APRM 15 percent scram will prevent higher power operation without being in the run mode.

The IRM scram provides protection for changes which occur both locally and over the entire core.

The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any safety limit is exceeded. For the case of a single control rod withdrawal error this transient has been analyzed in paragraph 7.5.5.4 of the FSAR. In order to ensure that the IRM provides adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Quarter rod density is illustrated in paragraph 7.5.5 of the FSAR. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

B. APRM Control Rod Block

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond

a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than 1.07. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during the steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the CMFLPD exceeds FRP thus preserving the APRM rod block safety margin.

C. Reactor Water Low Level Scram and Isolation
(Except Main Steamlines)

The set point for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR subsection N14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than 1.07 in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 31 inches below the normal operating range and is thus adequate to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure, neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2).

E. Turbine Control Valve Scram

1. Fast Closure Scram

This turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection coincident with failures of the turbine bypass valves. The Reactor Protection System initiates a scram when fast closure of the control valves is initiated by the fast acting solenoid valves and in less than 30 milliseconds after the start of control valve fast closure. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic control

oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50% greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

2. Scram on loss of control oil pressure

The turbine hydraulic control system operates using high pressure oil. There are several points in this oil system where a loss of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the generator load rejection scram, since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and high reactor pressure scrams. However, to provide the same margins as provided for the generator load rejection scram on fast closure of the turbine control valves, a scram has been added to the reactor protection system, which senses failure of control oil pressure to the turbine control system. This is an anticipatory scram and results in reactor shutdown before any significant increase in pressure or neutron flux occurs. The transient response is very similar to that resulting from the generator load rejection.

F. Main Condenser Low Vacuum Scram

To protect the main condenser against overpressure, a loss of condenser vacuum initiates automatic closure of the turbine stop valves and turbine bypass valves. To anticipate the transient and automatic scram resulting from the closure of the turbine stop valves, low condenser vacuum initiates a scram. The low vacuum scram set point is selected to initiate a scram before the closure of the turbine stop valves is initiated.

G. & H. Main Steam Line Isolation on Low Pressure and Main Steam Line Isolation Scram

The low pressure isolation of the main steam lines at 825 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the reactor mode switch be in the STARTUP

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

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

These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram set point and initiation set points. Transient analyses reported in Section N14 of the FSAR demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

L. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, Feb., 1973.
2. General Electric Supplemental Reload Licensing Submittal for BFN Unit 3 Reload 1, NEDO-24128, June 1978

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 limits from being exceeded.

Objective

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

Specification

The limiting safety system settings shall be as specified below:

<u>Protective Action</u>	<u>Limiting Safety System Setting</u>
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	1105 psig + 11 psi (4 valves)
	1115 psig + 11 psi (4 valves)

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

1.2 REACTOR COOLANT SYSTEM
INTEGRITY

2.2 REACTOR COOLANT SYSTEM
INTEGRITY

1125 psig |
± 11 psi
(3 valves)

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

1.2 EASES

REACTOR COOLANT SYSTEM INTEGRITY

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

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

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

The current cycle's safety analysis concerning the most severe abnormal operational transient resulting directly in a reactor coolant system pressure increase is given in Reference 5. The reactor vessel pressure code limit of 1,375 psig given in subsection 4.2 of the safety analysis report is well above the peak pressure produced by the overpressure transient described above. Thus, the pressure safety limit applicable to power operation is well above the peak pressure that can result due to reasonably expected overpressure transients.

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

The safety limit of 1,375 psig actually applies to any point in the reactor vessel; however, because of the static water head, the highest pressure point will occur at the bottom of the vessel. Because the pressure is not monitored at this point, it cannot be directly determined if this safety limit has been violated. Also, because of the potentially varying head level and flow pressure drops, an equivalent pressure cannot be a priori determined for a pressure monitor higher in the vessel. Therefore, following any transient that is severe enough to cause concern that this safety limit was violated, a calculation will be performed using all available information to determine if the safety limit was violated.

REFERENCES

1. Plant Safety Analysis (BFNP FSAR Section N14.0)
2. ASME Boiler and Pressure Vessel Code Section III
3. USAS Piping Code, Section B31.1
4. Reactor Vessel and Appurtenances Mechanical Design (BFNP FSAR Subsection 4.2)
5. General Electric Supplemental Reload Licensing Submittal for BFNP Unit 3 Reload 1, NEDO-24128, June 1978

2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

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

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

To meet the safety design basis, thirteen safety-relief valves have been installed on each unit with a total capacity of 84.2% of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1277 psig if a neutron flux scram is assumed. This results in a 98 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 1203 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 1229 psig which is 146 psig below the allowed vessel overpressure of 1375 psig.

NOTES FOR TABLE 3.1.A

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable instrument channels per trip system cannot be met for both trip systems, the appropriate actions listed below shall be taken.
 - A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.
 - B. Reduce power level to IRM range and place mode switch in the Startup/Hot Standby position within 8 hours.
 - C. Reduce turbine load and close main steam line isolation valves within 8 hours.
 - D. Reduce power to less than 30% of rated.
2. Scram discharge volume high bypass may be used in shutdown or refuel to bypass scram discharge volume scram with control rod block for reactor protection system reset.
3. Bypassed if reactor pressure < 1055 psig and mode switch not in run.
4. Bypassed when turbine first stage pressure is less than 154 psig.
5. IRM's are bypassed when APRM's are onscale and the reactor mode switch is in the run position.
6. The design permits closure of any two lines without a scram being initiated.
7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:
 - A. Mode switch in shutdown
 - B. Manual scram
 - C. High flux IRM
 - D. Scram discharge volume high level
 - E. APRM 15% scram
8. Not required to be operable when primary containment integrity is not required.
9. Not required if all main steamlines are isolated.
10. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.
11. The APRM downscale trip function is only active when the reactor mode switch is in run.

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
1(10)	Instrument Channel - Thermostat (RHR Area Cooler Fan)	≤100°F	A	1. Above trip setting starts RHR area cooler fans.
2(10)	Instrument Channel - Core Spray A or C Start	N/A	A	1. Starts Core Spray area cooler fan when Core Spray motor starts
2(10)	Instrument Channel - Core Spray B or D	N/A	A	1. Starts Core Spray area cooler fan when Core Spray motor starts
1(10)	Instrument Channel - Thermostat (Core Spray Area Cooler Fan)	≤100°F	A	1. Above trip setting starts Core Spray area cooler fans
1(10)	RHR Area Cooler Fan Logic	N/A	A	
1(10)	Core Spray Area Cooler Fan Logic	N/A	A	
72 1(11)	Instrument Channel - Core Spray Motors A, B, C, or D Start	N/A	A	1. Starts RHRSW pumps A1, C1, B3, and D3
1(12)	Instrument Channel - Core Spray Loop 1 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A1, C1, B3, and D3
1(12)	Instrument Channel - Core Spray Loop 2 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A1, C1, B3, and D3
1	RPT logic	N/A	(17)	1. Trips recirculation pumps on turbine control valve fast closure or stop valve closure > 30% power.

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. During monthly functional testing, both RPT systems may be put in test mode for a period not to exceed 8 hours. If both RPT systems are inoperable or if one RPT system is inoperable for more than 30 consecutive days, the reactor power shall be reduced to below 30% power within 24 hours.

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

<u>Function</u>	<u>Functional Test</u>	<u>Calibration</u>	<u>Instrument Check</u>
	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 thermo- stat (core spray area cooler fan). No other test required.	N/A	N/A
97 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

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

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic, and all sensors are required to be operable.

High temperature in the vicinity of the HPCI equipment is sensed by 4 sets of 4 bimetallic temperature switches. The 16 temperature switches are arranged in 2 trip systems with 8 temperature switches in each trip system.

The HPCI trip settings of 90 psi for high flow and 200°F for high temperature are such that core uncover is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of 450" water for high flow and 200°F for temperature are based on the same criteria as the HPCI.

High temperature at the Reactor Cleanup System floor drain could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This time period is only 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control

rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

If the IRM channels are in the worst condition of allowed bypass, the scaling arrangement is such that for unbypassed IRM channels, a rod block signal is generated before the detected neutrons flux has increased by more than a factor of 10.

A downscale indication is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two post treatment off-gas radiation monitors are provided and, when their trip point is reached, cause an isolation of the off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip or both have a downscale trip.

Both instruments are required for trip but the instruments are set so that any instruments are set so that the instantaneous stack release rate limit given in Specification 3.8 is not exceeded.

Four radiation monitors are provided for each unit which initiate Primary Containment Isolation (Group 6 isolation valves) Reactor Building Isolation and operation of the Standby Gas Treatment System. These instrument channels monitor the radiation in the Reactor zone ventilation exhaust ducts and in the Refueling Zone.

3.3 REACTIVITY CONTROL

2. The control rod drive housing support system shall be in place during reactor power operation or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.
3. a. Whenever the reactor is in the startup or run modes below 20% rated power the Rod Sequence Control System (RSCS) shall be operable.

4.3 REACTIVITY CONTROL

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.
3. Prior to the start of control rod withdrawal at startup, and prior to attaining 20% rated power during rod insertion at shutdown, the capability of the Rod Sequence Control System (RSCS) and the Rod Worth Minimizer to properly fulfill their functions shall be verified by the following checks:

of two operable SRM's are provided as an added conservatism.

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

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

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.07.

Analysis of this transient shows that the negative reactivity rates resulting from the scram (FSAR Figure N3.6-9) with the average response of all the drives as given in the above specification, provide the required protection, and MCPR remains greater than 1.07.

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

*** See Section 3.5.K.

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 power level range. 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 1, NEDO-24128, June 1978

3.4 BASIS: STANDBY LIQUID CONTROL SYSTEM

- A. If no more than one operable control rod is withdrawn, the basic shutdown reactivity requirement for the core is satisfied and the Standby Liquid Control System is not required. Thus, the basic reactivity requirement for the core is the primary determinant of when the liquid control system is required.

The purpose of the liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron that produces a concentration greater than 600 ppm of boron in the reactor core in less than 125 minutes. The 600 ppm concentration in the reactor core is required to bring the reactor from full power to a subcritical condition, considering the hot to cold reactivity difference, xenon poisoning, etc. The time requirement for inserting the boron solution was selected to override the rate of reactivity insertion caused by cooldown of the reactor following the xenon poison peak.

The minimum limitation on the relief valve setting is intended to prevent the loss of liquid control solution via the lifting of a relief valve at too low a pressure. The upper limit on the relief valve settings provides system protection from overpressure.

- B. Only one of the two standby liquid control pumping loops is needed for operating the system. One inoperable pumping circuit does not immediately threaten shutdown capability, and reactor operation can continue while the circuit is being repaired. Assurance that the remaining system will perform its intended function and that the long-term average availability of the system is not reduced is obtained from a one-out-of-two system by an allowable equipment out-of-service time of one-third of the normal surveillance frequency. This method determines an equipment out-of-service time of ten days. Additional conservatism is introduced by reducing the allowable out-of-service time to seven days, and by increased testing of the operable redundant component.
- C. Level indication and alarm indicate whether the solution volume has changed, which might indicate a possible solution concentration change. The test interval has been established in consideration of these factors. Temperature and liquid level alarms for the system are annunciated in the control room.

3.5 CORE AND CONTAINMENT COOLING SYSTEMSG. Automatic Depressurization System (ADS)

1. Four of the six valves of the Automatic Depressurization System shall be operable:
 - (1) prior to a startup from a Cold Condition, or,
 - (2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except as specified in 3.5.G.2 and 3.5.G.3 below.
2. If three of the six ADS valves are known to be incapable of automatic operation, the reactor may remain in operation for a period not to exceed 7 days, provided the HPCI system is operable. (Note that the pressure relief function of these valves is assured by section 3.6.D of these specifications and that this specification only applies to the ADS function.) If more than three of the six ADS valves are known to be incapable of automatic operation, an immediate orderly shutdown shall be initiated, with the reactor in a hot shutdown condition in 6 hours and in a cold shutdown condition in the following 18 hours.

4.5 CORE AND CONTAINMENT COOLING SYSTEMSG. Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:
 - a. A simulated automatic actuation test shall be performed prior to startup after each refueling outage. Manual surveillance of the relief valves is covered, in 4.6.D.2.
2. When it is determined that more than two of the ADS valves are incapable of automatic operation, the HPCIS shall be demonstrated to be operable immediately and daily thereafter as long as Specification 3.5.G.2 applies.

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

I. Average Planar Linear Heat Generation Rate

During steady state power operation, the Maximum Average Planar Heat Generation Rate (MAPLHGR) for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Tables 3.5.I-1, -2, and -3.

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

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

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

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

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

J. Linear Heat Generation Rate (LHGR)

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

$$\text{LHGR}_{\text{max}} \leq \text{LHGR}_d [1 - (\Delta P/P)_{\text{max}} (L/LT)]$$

LHGR_d = Design LHGR = 13.4 kW/ft.

$(\Delta P/P)_{\text{max}}$ = Maximum power spiking penalty = 0.021

LT = Total core length - 12.2 feet*

L = Axial position above bottom of core

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

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

J. Linear Heat Generation Rate (LHGR)

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

*12.5 feet for 8X8R fuel

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

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

K. Minimum Critical Power Ratio (MCPR)

The MCPR operating limit is 1.24 for 8X8 fuel and 1.21 for 8X8R 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 $\geq 25\%$ rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.

*These limits apply for the initial 2000 MWD/t of cycle 2. OLMCPR's for operation in excess of 2000 MWD/t fuel exposure in cycle 2 will be determined by a reanalysis of transients for the EOC.

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taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS.

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

H. Maintenance of Filled Discharge Pipe

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

The core spray and RHR system discharge piping high point vent is visually checked for water flow once a month prior to

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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 $\pm 20^\circ\text{F}$ 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 dated June 1978.

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

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logged and reported quarterly. It must be recognized that there is always an action which would return any of the parameters (MAPLHGR, LHGR, or MCPR) to within prescribed limits, namely power reduction. Under most circumstances, this will not be the only alternative.

M. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-10735, August 1973.
2. Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (USA Regulatory Staff).
3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.
4. General Electric Supplemental Reload Licensing Submittal for BFN Unit 3 Reload 1, NEDO-24128, June 1978

Table 3.5.I-1

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3Fuel Type: Initial Core - Type 2

<u>Average Planar Exposure (MWd/t)</u>	<u>MAPLHGR (kW/ft)</u>	<u>PCT (°F)</u>	<u>Oxidation Fraction</u>
200	11.4	1893	0.009
1000	11.6	1904	0.009
5000	12.0	1922	0.010
10,000	12.2	1900	0.009
15,000	12.3	1926	0.009
20,000	12.1	1928	0.009
25,000	11.3	1828	0.006
30,000	10.2	1700	0.004

Table 3.5.I-2

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3Fuel Type: Initial Core - Type 1

<u>Average Planar Exposure (MWd/t)</u>	<u>MAPLHGR (kW/ft)</u>	<u>PCT (°F)</u>	<u>Oxidation Fraction</u>
200	11.2	1889	0.009
1000	11.3	1887	0.009
5000	11.8	1897	0.009
10,000	12.1	1920	0.009
15,000	12.3	1949	0.010
20,000	12.1	1951	0.010
25,000	11.3	1852	0.007
30,000	10.2	1718	0.004

Table 3.5.I-3

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3

Fuel Type: 8DRB265

<u>Average Planar Exposure (MWd/t)</u>	<u>MAPLHGR (kW/ft)</u>	<u>PCT (°F)</u>	<u>Oxidation Fraction</u>
200	11.6	1947	0.011
1000	11.6	1941	0.011
5000	12.1	1963	0.011
10,000	12.1	1941	0.010
15,000	12.1	1956	0.011
20,000	11.9	1950	0.011
25,000	11.3	1888	0.009
30,000	10.7	1817	0.007

3.6 PRIMARY SYSTEM BOUNDARYD. Safety and Relief Valves

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

4.6 PRIMARY SYSTEM BOUNDARYD. Safety and Relief Valves

1. At least one safety valve and approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves (2 safety and 11 relief) will have been checked or replaced upon the completion of every second cycle.
2. Once during each operating cycle, each relief valve shall be manually opened until thermocouples downstream of the valve indicate steam is flowing from the valve.
3. The integrity of the relief/safety valve bellows shall be continuously monitored.
4. At least one relief valve shall be disassembled and inspected each operating cycle.

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limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the unit should be shut down to allow further investigation and corrective action.

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

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

REFERENCES

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

3.6.D/4.6.D Safety and Relief Valves

The safety and relief valves are required to be operable above the pressure (105 psig) at which the core spray system is not designed to deliver full flow. The pressure relief system 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.

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To meet the safety design basis, thirteen safety-relief valves have been installed on each unit with a total capacity of 34.2% 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 1277 psig if a neutron flux scram is assumed. This results in a 97 psig margin to the code allowable overpressure limit of 1375 psig.

To meet the operational design basis, the total safety-relief capacity of 34.2% 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 1203 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 1229 psig which is 146 psig below the allowed vessel overpressure of 1375 psig.

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

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

REFERENCES

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

TABLE 3.7.D
PRIMARY CONTAINMENT ISOLATION VALVES

Valves	Valve Identification	Test Medium	Test Method
71-2	RCIC Steam Supply	Air (1)	Applied between 71-2 and 71-3.
71-3	RCIC Steam Supply	Air (1)	Applied between 72-2 and 71-3
71-39	RCIC Pump Discharge	Water (2)	Applied between 3-66, 3-568, 71-39, 69-579, and 85-576.
73-2	HPCI Steam Supply	Air (1)	Applied between 73-2 and 73-3
73-3	HPCI Steam Supply	Air (1)	Applied between 73-2 and 73-3
73-44	HPCI Pump Discharge	Water (2)	Applied between 73-34, 73-35, and 73-44.
74-47	RHR Shutdown Suction	Water (2)	Applied between 74-47 and 74-49
74-48	RHR Shutdown Suction	Water (2)	Applied between 74-48 and 74-49
74-53	RHR LPCI Discharge	Water (2)	Applied between 74-53 and 74-55
74-57	RHR Suppression Chamber Spray	Water (2)	Applied between 74-57, 75-58, and 74-59.
74-58	RHR Suppression Chamber Spray	Water (2)	Applied between 74-57, 74-58, and 74-59
74-60	RHR Drywell Spray	Water (2)	Applied between 74-60 and 74-61
74-61	RHR Drywell Spray	Water (2)	Applied between 74-60 and 64-61
74-67	RHR LPCI Discharge	Water (2)	Applied between 74-67 and 74-69
74-71	RHR Suppression Chamber	Water (2)	Applied between 74-71, 74-72, and

TABLE 3.7.G
CHECK VALVES ON DRYWELL INFLUENT LINES

Valves	Valve Identification	Test Medium	Test Method
3-554	Feedwater	Water	Applied between 3-67 and 3-554. Valves 73-45, 73-44, 73-35, and 73-34 are used to form a water seal on 73-45.
3-558	Feedwater	Water	Applied between 3-67 and 3-558
3-568	Feedwater	Water	Applied between 3-66, 3-568, 71-40, 69-579, and 85-576.
3-572	Feedwater	Water	Applied between 3-66 and 3-572
63-525	Standby Liquid Control Discharge	Water	Applied between 63-525 and 63-527
63-526	Standby Liquid Control Discharge	Water	Applied between 63-526 and 63-527
69-579	RWCU Return	Water	Applied between 3-66, 3-568, 69-579, and 71-40, and 85-576.
71-40	RCIC Pump Discharge	Water	Applied between 3-66, 3-568, 69-579, and 71-40, and 85-576.
73-45	HPCI Pump Discharge	Water	Applied between 3-67, 3-559 and 73-45
74-54	RHR LPCI Discharge	Water	Applied between 74-54 and 74-55
74-68	RHR LPCI Discharge	Water	Applied between 74-68 and 74-69

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TABLE 3.7.G
CHECK VALVES ON DRYWELL INFLUENT LINES

Valves	Valve Identification	Test Medium	Test Method
75-26	Core Spray Discharge	Water	Applied between 75-26 and 75-27
75-54	Core Spray Discharge	Water	Applied between 75-54 and 75-55
85-573	CRD Hydraulic Return	Water	Applied between 85-573 and 85-577
85-576	CRD Hydraulic Return	Water	Applied between 85-576, 3-66, 3-568, 71-40, and 69-579.

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 core shall consist of 556 fuel assemblies of 63 fuel rods each and 208 fuel assemblies of 62 fuel rods each.
- B. The reactor core shall contain 185 cruciform-shaped control rods. The control material shall be boron carbide power (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 facility 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. 45 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 18 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS NOS. 1, 2 AND 3

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

1.0 Introduction

By letter dated August 3, 1978, and supplemented by letter dated October 20, 1978, the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. The proposed amendments and revised Technical Specifications would (1) incorporate the limiting conditions for operation associated with cycle 2 operation of Unit No. 3, and (2) incorporate minor changes to the leak rate testing valve lineups to reflect the current test program being conducted in accordance with the requirements of 10 CFR 50 Appendix J.

2.0 Discussion

Browns Ferry Unit No. 3 (BF-3) shutdown on September 8, 1978 for the first refueling of this unit. During the outage, 208 of the 764 fuel assemblies were replaced. Unit No. 3 was initially fueled with 8x8 fuel assemblies manufactured by the General Electric Company (GE).

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In support of the reload application, the licensee has provided the GE BWR Reload 1 licensing submittal for BF-3 (Reference 1), proposed Technical Specification changes (Reference 2), information on the BF-3 Loss of Coolant Accident (LOCA) analysis (Reference 3), and responses to NRC requests for additional information (Reference 4).

This reload involves loading of GE 8x8 fuel and GE8x8 retrofit (8x8R) fuel. The description of the nuclear and mechanical design of the 8x8 and 8x8R fuel is contained in GE's licensing topical report for BWR reloads (Reference 5). Reference 5 also contains a complete set of references to topical reports which describe GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, and information regarding the applicability of these methods to cores containing a mixture of 8x8 and 8x8R fuel.

Values for plant-specific data such as steady state operating pressure, core flow, safety and safety/relief valve setpoints, rated thermal power, rated steam flow, and other design parameters are provided in Reference 5. Additional plant and cycle dependent information is provided in the reload application (Reference 1), which closely follows the outline of Appendix A of Reference 5.

Reference 6 describes the staff's review, approval, and conditions of approval for the plant-specific data addressed in Reference 5. The above-mentioned plant-specific data have been used in the transient and accident analysis provided with the reload application.

Our safety evaluation (Reference 6) of the GE generic reload licensing topical report has also concluded that the nuclear and mechanical design of the 8x8R fuel, and GE's analytical methods for nuclear and thermal-hydraulic calculations as applied to mixed cores containing 8x8 and 8x8R fuel, are acceptable. Approval of the application of the analytical methods did not include plants incorporating a prompt recirculation pump trip (RPT).

Because of our review of a large number of generic considerations related to use of 8x8R fuel in mixed loadings, and on the basis of the evaluations which have been presented in Reference 6, only a limited number of additional areas of review have been included in this safety evaluation report. For evaluations of areas not specifically addressed in this safety evaluation report, the reader is referred to Reference 6.

3.0 Evaluation

3.1 Nuclear Characteristics

For Cycle 2 operation of BF-3, 208 fresh 8x8R fuel bundles of type 8DRB265 will be loaded into the core. (1) The remainder of the 764 fuel bundles in the core will be 8x8 fuel bundles of type 8D219 exposed during the previous cycle.

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

Based on the data provided in Sections 4 and 5 of Reference 1, both the control rod system and the standby liquid control system will have acceptable shutdown capability during Cycle 2.

3.2 Thermal-Hydraulics

3.2.1 Fuel Cladding Integrity Safety Limit

As stated in Reference 5, the minimum critical power ratio (MCPR) which may be allowed to result from core-wide or localized transients (or from undetected fuel loading errors) is 1.07. This limit has been imposed to assure that during transients 99.9% of the fuel rods will avoid transition boiling, and that transition boiling will not occur during steady state operation as a result of the worst possible fuel loading error.

The safety limit MCPR for BF-3 is being raised from 1.06 to 1.07 because the distribution of fuel rod power within the 8x8R fuel bundles is different from that of the 8x8 fuel. The reason for the difference is the presence of two rather than one water rods in 8x8R fuel. The issue has been addressed in Reference 6 and the 1.07 limit has been found acceptable for BWRs with uncertainties in flux monitoring and operational parameters no greater than those listed in Table 5-1 of Reference 5, for which the CPR distribution is within the bounds of Figures 5.2 and 5.2a of Reference 5. It has been shown in Section 5 of Reference 5 that these conditions are met for BF-3.

3.2.2 Operating Limit MCPR

Various transients or perturbations to the CPR distribution could reduce the CPR below the intended operating limit MCPR during Cycle 2 operation. The most limiting of these operational transients and the fuel loading error have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial power ratio (Δ CPR).

The transients evaluated were the limiting pressure and power increase transient (either turbine trip or load rejection without bypass, depending on which values have the faster closure time), the limiting coolant temperature decrease transient (loss of a feedwater heater), the feedwater controller failure transient, and the control rod withdrawal error transient. Initial conditions and transient input parameters as specified in Sections 6 and 7 of Reference 1 were assumed.

The calculated systems responses and Δ CPRs for the above listed operational transients and conditions have been analyzed by the licensee. Results were as follows:

	Δ CPR 8x8	Δ CPR 8x8R
Limiting Pressure and Power Increase Transient	.14	.14
Limiting Coolant Temperature Decrease Transient	.13	.13
Feedwater Controller Failure Transient	.09	.09
Rod Withdrawal Error	.17	.14
Fuel Loading Error, Rotated Bundle*	\leq .10	.10

*The misloaded bundle error is considered separately in Section 2.3.3

The above analyses include the effect of a recirculation pump trip (RPT) on turbine stop valve closure or throttle valve fast closure. This RPT feature inserts negative reactivity into the reactor due to the rapid flow decrease and resultant increased voiding. Thus, the RPT helps shut down the reactor, effectively increasing the speed of turbine-initiated scrams.

The transient analyses described above were performed with the REDY code (Reference 7). A new improved code, ODYN, has been developed by GE. The ODYN code, which uses a more physically correct model of the plant, generally predicts smaller Δ CPRs than the REDY code when the transient under study is fairly severe. However, as transient severity is lessened, ODYN predicts a greater Δ CPR than REDY (Reference 8, p. 1). Both codes are run with conservative input values, but ODYN should be a better predictor of plant behavior once these input values are specified.

GE has stated (Reference 8) that REDY can still be used because the limiting transient has a Δ CPR sufficiently large to be above the region where REDY is non-conservative with respect to ODYN. We have proceeded on this basis in approving reloads thus far.

The addition of the RPT feature to BF-3 has significantly reduced the Δ CPR associated with the limiting pressure and power increase transient. (TVA has provided no data, but we estimate a reduction in Δ CPR by roughly a factor of two based upon p. 12 of Reference 8.) This improvement has brought the BF-3 Cycle 2 transient analysis into the region where GE's assertion (Reference 8) is no longer valid. Thus, the degree of conservatism of the BF-3 Cycle 2 transient analysis must be re-evaluated.

Approximately six to eight weeks are required to reanalyze the operational transients for cycle 2 operation of Unit 3 with the ODYN code at a cost of \$85,000 to \$120,000. NRC has not as yet approved the ODYN code. However, the staff had requested that TVA supply an ODYN licensing basis reanalysis of the transients to compare these results with those obtained by the accepted REDY code. Initially (reference 4), TVA's position was that this reanalysis was unwarranted until such time as the ODYN code was approved by NRC.

The limited data available to the staff indicates that calculations which include axial effects and detailed steam line modeling are likely to predict more severe results than those obtained by the point kinetics REDY calculations. This possible lack of conservatism in the REDY calculations is of concern only for the end of the fuel cycle (EOC). It is known that transient severity is greatest at end-of-cycle, generally increasing by 0.06 or more in a Δ CPR during the last 2000 megawatt days per tonne (MWD/t) of fuel exposure in the cycle (section 5.2.2.5, reference 5). The transients for the Unit 3 cycle 2 reload were calculated for the EOC conditions, which are the most severe conditions. Thus, there is considerable extra conservatism in the calculated operating limit minimum critical power ratio (OLMCPR) at the beginning of the cycle. The only staff concern is the degree of conservatism at the end of the cycle.

To resolve the staff's concern, TVA has agreed to reanalyze the transients at the end of cycle 2. The total cycle is estimated to result in 5415 MWD/t exposure to the fuel. As noted above, the only concern is with the later part of the cycle. The OLMCPRs proposed by TVA as a result of the REDY analysis are conservative for at least the initial 2000 MWD/t exposure during the fuel cycle. Therefore, the staff has proposed, and the licensee has accepted, that the proposed OLMCPRs of 1.24 for 8x8 fuel and 1.21 for 8x8R fuel will apply for the first 2000 MWD/t exposure in cycle 2; that is, from the beginning of the cycle (BOC) to BOC + 2000 MWD/t. During this period, TVA will submit a reanalysis and the staff will reevaluate the OLMPCRs for the balance of the cycle.

3.2.3 Thermal-Hydraulic Improvement Features

3.2.3.1 Prompt Recirculation Pump Trip

The prompt recirculation pump trip feature was described in Reference 9. The system uses line breakers between the motor-generator sets and the pump motors. This location provides the rapid reduction in pump speed necessary for the feature to be effective during the transient discussed in Section 2.2.2. The system is designed to be of quality consistent with the reactor protection system. The RPT design was reviewed and accepted for Cycle 2 of Browns Ferry Unit 2 (Reference 10). The design remains acceptable.

3.2.3.2 Simmer Margin

The licensee has proposed changes to the Technical Specifications which will increase the capacity (by installing larger valves) of the safety/relief valves from 78.7% to 84.2% of nuclear boiler rated (NBR) steam flow, and also increase the setpoints of the relief valves. (The safety valve capacity and setpoints were not changed.) The transient, overpressure, and LOCA analyses performed for the Cycle 2 analysis assumed this change.

The criterion for simmer margin is that only relief valves open during anticipated transients. Safety valves should not open under these conditions.

The analysis of the limiting pressure and power increase transient, which is the worst case for anticipated pressure events, predicted a pressure of 1203 psig at the safety valves, which is well below the 1250 psig safety valve setpoint. Moreover, peak pressures calculated with the REDY code have always been greater than those calculated using ODYN (Reference 8), and thus the concerns outlined in Section 3.2.2 do not apply here. Therefore, we find these changes to be acceptable.

3.3 Accident Analysis

3.3.1 ECCS Appendix K Analysis

Input data and results for the ECCS analysis have been given in Reference 1, 3, and 11. The information presented fulfills the requirements for such analyses outlined in Reference 6.

We have reviewed the analyses and information submitted for the reload and conclude that the BF-3 plant will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when (1) it is operated within the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values given in Tables 3.5.I-1, -2, and -3 of Reference 2, and (2) it is operated at a Minimum Critical Power Ratio (MCPR) greater than or equal to 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the Loss of Coolant Accident, as described in Section 3.2.2).

3.3.2 Control Rod Drop Accident

For BF-3 Cycle 2, the generic scram reactivity curve (cold and hot) and the accident reactivity insertion curve (cold) do not satisfy the requirements for the bounding analyses described in Reference 5. Therefore, it was necessary for the licensee to perform plant and cycle specific analyses for the control rod drop accident for hot and cold startup conditions. The results of these analyses indicate that the peak fuel enthalpy for these events would be less than or equal to 280 calorics/gram, which is acceptable.

3.3.3 Fuel Loading Error

As discussed in Section 3.2.2, potential fuel loading errors involving misoriented bundles have been explicitly included in the calculation of the operating limit MCPR. Potential errors involving bundles loaded into incorrect positions have also been analyzed by a method which considers the initial MCPR of each bundle in the core, and the resultant MCPR was shown to be greater than 1.07. This GE method for analysis of misoriented and misloaded bundles has been reviewed and approved by the staff (Reference 12).

The analyses which have been performed for potential fuel loading errors for BF-3 Cycle 2 are acceptable for assuring that CPRs will not be below the safety limit MCPR of 1.07.

3.3.4 Overpressure Analysis

The overpressure analysis for the MSIV closure with high flux scram, which is the limiting overpressure event, has been performed in accordance with the requirements of Reference 6. As specified in Reference 8, the sensitivity of peak vessel pressure to failure of one safety valve has also been evaluated. We agree that there is sufficient margin between the peak calculated vessel pressure and the design limit pressure to allow for the failure of at least one valve. Therefore, the limiting overpressure event as analyzed by the licensee is considered acceptable.

2.3.5 ADS Out-of-Service Analysis

The automatic depressurization system (ADS) is provided to aid in vessel depressurization following a small break loss-of-coolant accident (LOCA). Thus, the ADS only affects the results of break analyses where depressurization through the break itself is relatively slow (small breaks), and operation of the ADS increases the depressurization rate, allowing low pressure systems (such as the core spray (CS) and the low pressure coolant injection (LPCI) systems) to reach higher flows sooner. This causes earlier reflood and lower calculated peak cladding temperature (PCT) results for the small break analyses. The more installed relief capacity (i.e., number of valves) in the ADS, the more pronounced is this effect.

Previous small-break analyses, in the small break size range where ADS has an appreciable effect (0 to approximately 0.5 ft²), took credit for operation of five of the six ADS valves (Reference 13). Maximum PCT in that break size range was around 1530°F, far below the larger (and limiting) break sizes whose PCTs are around but still below 2200°F.

Continuous reactor operation with only four of the six ADS valves operable is acceptable if the small breaks' PCTs do not exceed 2200°F for any fuel operating at the MAPLHGR limit.

The application for change in the Technical Specifications (Reference 3) contained a generic estimate of a 200°F PCT increase for small breaks in the range affected by ADS capacity (0 to 0.5 ft²). We have previously required substantiation of that estimate for Units 1 and 2 of Browns Ferry, which was provided in Reference 14 as discussed below. The results also apply to BF-3, as the three plants are similar except that BF-3 does not have the LPCI modification. The LPCI modification will have no effect on this analysis because loss of HPCI is the worst single failure.

- (1) The estimate of 200°F PCT increase was provided for the Browns Ferry plants by a generic ADS out-of-service analysis, which included calculations for a 251-inch inside diameter pressure vessel (Reference 14). BF-3 is within this category.
- (2) The generic estimate of 200°F PCT increase was confirmed for the ADS steam flow range appropriate for BF-3 (with four and five ADS valves operable) by the generic ADS out-of-service analysis, which included the BF-3 ADS' capacity range.
- (3) The model used for the generic ADS out-of-service analysis did not contain the latest model changes described in Reference 15. However, those model changes have not caused significant changes in the PCT results for the small break analyses of a smaller sized BWR/4 and an identically sized BWR/3 (Reference 14), and similarly the changes would not significantly affect small break PCT results for BF-3.

For other reasons, the model changes (Reference 15) allowed operation at slightly higher MAPLHGR limits. At these higher powers, small break PCT results could be as much as 40°F higher. Therefore, PCT for the worst small break with four of the six ADS valves operable would be approximately $1460^{\circ}\text{F} + 200^{\circ}\text{F} + 40^{\circ}\text{F} = 1700^{\circ}\text{F}$. This is considerably below 2200°F and is therefore acceptable.

We, therefore, conclude that the material presented and discussed above adequately supports the TVA request to operate continuously with four of the six ADS valves in service, and such operation is, therefore, acceptable.

3.3.6 Recirculation Pump Trip Failure

It is extremely unlikely that the RPT feature will fail. However, the consequences must be examined to see if they lie within the accident criteria.

The limiting pressure and power increase transient, with failure of the RPT feature, may result in fuel failure if all plant parameters are close to worst-case condition. Radioactive material could then be released through the feedwater pump turbines, steam jet air ejectors, and gland seals. (Most of this material would have to pass through the offgas system before release.) The

specific activity within the steam would have to be below the value which would trigger MSIV closure on high steam line activity. An incident which caused isolation on high activity would be bounded by the analysis of the steam line break in the plant FSAR. Since the high steam line radiation setpoint is required by the Technical Specifications to be no more than three times normal background, transients coupled with RPT failures leading to coolant activities greater than three times the Technical Specification maximum would fall into this category.

During the course of the limiting pressure and power increase transient, the increasing water level reaches the high level setpoint eight seconds into the transient, which trips the feedwater turbines. The water level then reaches a maximum and recedes. We estimate (by extrapolation of the data in Reference 1) that the level will drop to the low low setpoint after approximately 25 seconds. At this point, HPCI and RCIC initiate and the MSIV begin to close (Group I isolation). MSIV closure requires three to five additional seconds.

Failure of the RPT feature should not greatly affect the water level behavior except in the very early stages of the transient, when the void-sweeping effects are important. Once the MSIVs close, the radioactive releases will be bounded by the steam line break accident. Therefore, the important question is: how much steam flows through the feedwater turbines, steam jet air ejectors and gland seals in the 25 seconds before isolation?

At full power, the feedwater turbines on any LWR installation consume 2% or less of the main steam flow. The SJAEs and gland seals consume much less. Moreover, the feedwater turbines are tripped after eight seconds.

Clearly, assuming three times the maximum permissible coolant activity, 2% steam flow for eight seconds plus much less than 2% for 22 additional seconds will result in less release than 200% steam flow for five seconds at the maximum permissible coolant activity. The difference is greater than a factor of 5. Therefore, the 200%-five second assumptions of the steam line break analysis are bounding, and the consequences of RPT failure are acceptable.

Common mode failures must also be examined. The RPT feature operates off the same steam chest switches as the reactor scram on trip/fast closure. The licensee has referred (Reference 4) to probabilistic analyses submitted on other dockets. These analyses conclude that the probability of failure of the reactor scram is on the order of 10^{-6} per demand (Reference 16). The switches are only one contributor to this failure rate. Moreover, the RPT hardware is of similar quality to the reactor scram hardware (Reference 9). Therefore, it is concluded that the probability of simultaneous failure of the trip/fast closure scram and the RPT feature is much less than 10^{-6} per demand, and therefore need not be considered.

It is our judgement that all other simultaneous failures (e.g., caused by a seismic event) would necessitate failure of some equipment but not others in arrays which are of negligible probability.

3.4 Thermal Hydraulic Stability

The results of the thermal hydraulic stability analysis (Reference 1) show that the channel hydrodynamic and reactor core decay ratios at the natural circulation - 105% rod line interaction (which is the least stable physically attainable point of operation) are below the stability limit.

Because operation in the natural circulation mode at greater than 50% rated thermal power is prohibited by the Technical Specifications, there is added margin to the stability limit and this is acceptable.

3.5 Physics Startup Testing

The licensee will perform a series of physics startup tests and procedures to provide assurance that the conditions assumed for the transient and accident analysis calculations will be met during Cycle 2. The tests will check that the core is 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 licensee has stated (Reference 17) that the methods, criteria and reporting requirements for the test program will be, with two exceptions, identical to these accepted for Unit 2 (Reference 10).

The first exception involves the action to be taken in the event that the TIP asymmetry test indicates that the TIP instrumental uncertainty is in excess of that assumed in the development of the safety limit MCPR (Section 5 of Reference 5). Normally, an instrumental uncertainty higher than that assumed in the safety analyses would require additional safety margin, and thus some operating limit penalty.

TVA stated (Reference 17) that increased instrumental uncertainties will automatically penalize the operation of the plant in terms of MCPR, MAPLHGR, MLHGR and TPF by an amount greater than the penalty that would be calculated by a re-assessment of the safety limit assumptions. The reason this effect takes place is because (1) there are many locations in the core which run at powers very nearly equal to that of the peak power location, and (2) the operating limits are written in terms of maxims. Thus, even if the maximum location is read low due to instrumental uncertainty, there is a nearly unity probability that another location, almost as high in power, will be read high. Provided the peak location is accompanied by many other locations which are less in power by an amount which is much smaller than the instrumental uncertainty, the maximum value read by the incore instrumentation will automatically be conservative. Moreover, this automatic penalty rises in a nearly linear fashion as the instrumental uncertainty increases.

Since the instrumental uncertainty assumed in the safety analysis is combined statistically (i.e., RMS) with other allowances, the penalty calculated from the safety analysis rises less than linearly with increased instrumental uncertainty. Therefore, the automatic penalty discussed above is always greater than or equal to the appropriate safety penalty. Since the BF-3 Cycle 2 core meets all of the above criteria, we find this change to the startup test program to be acceptable.

The second exception involves the comparison of predicted vs. measured core power maps at high power, BOC conditions. The licensee has expressed difficulty in distinguishing power map discrepancies from instrumental noise and maintains that the balance of his testing program will detect any anomalies in the core (reference 17). Therefore, the licensee desires to eliminate this test.

After reviewing the licensee's core loading and past experience with power map uncertainties, we agree that this test is insufficiently sensitive to detect most postulated core anomalies. Moreover, examination of the presently available studies of the sensitivity of BWR core power maps to various perturbations indicates that there are not enough of these studies presently available to allow interpretation of core power maps discrepancies, even if such discrepancies could be unambiguously identified. Therefore, we find this second change to the startup program to be acceptable.

3.6 Rod Sequence Control System

Section 3.3.B.3.a of the present Technical Specifications for BR-3 contains a note which reads: "The Rod Sequence Control System (RSCS) has been evaluated only through the first refueling outage. A complete reevaluation is required prior to operations following the first refueling". As discussed in the introduction, BF-3 shutdown for the first refueling on September 8, 1978. BF-3 now has the Group Notch RSCS, as discussed in Reference 5 and accepted in Reference 6. Therefore, we find that the licensee's proposed deletion of the note in Section 3.3.B.3.a of the Technical Specifications is acceptable.

3.7 Primary Containment Isolation Valves

The surveillance requirements for testing primary containment integrity are specified in Section 4.7 of the Technical Specifications. Section 4.7.A.2.g states that local leak rate tests shall be performed on the primary containment testable penetrations and isolation valves at certain specific pressures and intervals. The testable penetrations and valves are listed in seven tables (3.7B thru 3.7H).

Table 3.7.D lists 105 primary containment isolation valves by number of the valve, the test medium to be used to test the specific valves (i.e., air or water) and the sections of lines to be tested for each valve (i.e., the test pressure will be applied, for example, between valves 74-48, 74-49 and 74-661). The inservice inspection and testing program for Browns Ferry has been under review by the staff and the licensee for the past two years (see TVA's submittals of May 25, 1977 and July 29, 1977, our letters of February 25, 1977 and August 8, 1978 and summary of meetings held August 15 and 16, 1978 between the staff and TVA on the ISI program). As a result of the continuing efforts to keep up with the Appendix J requirements, TVA has proposed to change the section of line to be tested for three of the 105 valves in Table 3.7.D (i.e., the hydrostatic test will be applied between different valves). The changes do not change the valves to be tested or the test medium to be used (water in all 3 cases). The changes are proposed to permit testing of more than one valve at a time.

Table 3.7.6 lists 15 check valves on drywell influent lines that are required to be tested. TVA proposes to delete the check valve that was listed for the control rod drive return line since it no longer exists in the plant; the CRD return line was rerouted and the penetration capped at the reactor vessel to reduce the potential for intergranular stress corrosion cracking. TVA also proposes to change the section of line to be tested for 6 of the check valves to eliminate testing each valve individually to reduce the initial test time. There are no proposed changes to the valves to be tested, other than for the CRD return line, and no change in the test medium.

The staff concludes that the proposed changes to the test procedures for the primary containment isolation and check valves are in accordance with 10 CFR 50, Appendix J, they do not in any way change the valves to be tested and that the proposed changes are acceptable.

4.0 Environmental Considerations

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

5.0 Conclusion

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

Date: NOVEMBER 18 1978

6.0 References

1. Supplemental Reload licensing submittal for Browns Ferry Nuclear Plant Unit 3 Reload 1, NEDO-24128, June, 1978, submitted as Enclosure 2 of letter, O. E. Gray III (TVA) to H. R. Denton (NRC), dated August 3, 1978.
2. Enclosure 1 of letter, O. E. Gray III (TVA) to H. R. Denton (NRC), dated August 3, 1978.
3. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3, NEDO-24127, June, 1978, submitted as Enclosure 2 of letter, O. E. Gray III (TVA) to H. R. Denton (NRC), dated August 3, 1978.
4. Letter, J. E. Gilleland (TVA) to Director of Nuclear Reactor Regulation (NRC), dated October 20, 1978.
5. General Electric Boiling Water Reactor Generic Reload Application, NEDE-24011-P, May, 1977.
6. Safety Evaluation for the General Electric Topical Report "Generic Reload Fuel Application," NEDE-24011-P, April, 1978, transmitted as enclosure of letter, D. G. Eisenhut (NRC) to R. Gridley (GE), dated May 12, 1978.
7. Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor, NEDO-10802, February, 1973.
8. Impact of One-Dimensional Transient Model on Plant Operations Limits, enclosure of letter, E. D. Fuller (GE) to U. S. Nuclear Regulatory Commission, dated June 26, 1978.
9. Basis for Installation of Recirculation Pump Trip System, NEDO-24119, April, 1978.
10. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 35 to Facility Operating License No. DPR-52, enclosed in letter, T. A. Ippolito (NRC) to TVA (Attn. N. B. Hughes), dated June 21, 1978.

11. Letter, Lee Liu (Iowa Electric Light & Power Co.) to Edson G. Case (NRC), Letter No. IE-77-1453, dated July 29, 1977.
12. Letter, D. G. Eisenhut (NRC) to R. Engel (GE), dated May 8, 1978.
13. Letter, J. Gilleland (TVA) to E. Case (NRC), dated October 28, 1977.
14. Letter, J. E. Gilleland (TVA) to Director of Nuclear Reactor Regulation (NRC), dated April 20, 1978.
15. Letter, K. Goller (NRC) to G. Sherwood (GE), SER for GE ECCS Evaluation Model, dated April 12, 1977.
16. 251 NSSS GESSAR, Attachment A to Response to Staff Question 222.22.
17. Minutes of meeting between NRC staff and TVA, held in Bethesda on September 26, 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSIONDOCKET NOS. 50-259, 50-260, AND 50-296TENNESSEE VALLEY AUTHORITYNOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY
OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 45 to Facility Operating License No. DPR-33, Amendment No. 41 to Facility Operating License No. DPR-52 and Amendment No. 18 to Facility Operating License No. DPR-68 issued to Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3, located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

Amendment No. 18 changes the Technical Specifications to incorporate the limiting conditions for operation associated with the initial 2000 megawatt days per tonne (MWD/t) fuel exposure during the second fuel cycle for Unit No. 3. The amendments also incorporate minor changes in the test setups to be used to test certain primary containment isolation and check valves.

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


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

For further details with respect to this action, see (1) the application for amendments dated August 3, 1978, as supplemented by letter dated October 20, 1978, (2) Amendment No. 45 to License No. DPR-33, Amendment No. 41 to License No. DPR-52, and Amendment No. 18 to License No. DPR-68, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 18th day of November 1978.

FOR THE NUCLEAR REGULATORY COMMISSION


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