2/8/79

Docket No. 50-259

Distribution
Docket
ORB #3
Local PDR
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NRR Reading
BGrimes
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RClark
SSheppard
Attorney, OELD
OI&E (5)
BJones (4)
BScharf (10)
STSG
DEisenhut
RVollmer
ACRS (16)
OPA (CMIles)
DRoss
TERA
JRBuchanan
HDenton
RDiggs

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

Dear Mr. Parris:

The Commission has issued the enclosed Amendment No. 48 to Facility License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit No. 1. This amendment changes the Technical Specifications in response to your request of September 8, 1978 (TVA BFNP TS 115), as supplemented by your letters of October 5, 1978, November 30, 1978, December 5, 1978, December 14, 1978, January 8, 1979, January 9, 1979 and January 23, 1979.

By letter dated January 17, 1979, we issued Amendment No. 47 to Facility License No. DPR-33 which authorized you to startup and operate Browns Ferry Unit No. 1 in the third fuel cycle. As discussed in that letter and the accompanying safety evaluation, the operating limit minimum critical power ratios (OLMCPR) incorporated in Section 3.5.K of the Technical Specifications by Amendment No. 47 did not include credit for the end-of-cycle recirculation pump trip (EOC RPT) feature which you installed in Unit No. 1 during the refueling outage. The OLMCPRs in Amendment No. 47 were very conservative, bounding values, which as you note in your letter of January 23, 1979 have limited the facility output to approximately 86% of rated power. As a result of the meeting with TVA and General Electric Company representatives on January 17, 1979 and your letter of January 23, 1979, our concerns about the testability of the EOC RPT have been resolved. This amendment changes the OLMCPRs to those values justified by the analyses in the submittals referenced above.

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

Sincerely,

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

~~Enclosures and ccs:~~
See page 2

Correct as to form of FR Notice. No legal objection

OFFICE	ORB #3	ORB #3	~~OELD~~	ORB #3
SURNAME	SSheppard	RClark:mjf	Cutchin	TIppolito
DATE	2/9/79	2/7/79	2/8/79	/ /79

Enclosures:
1. Amendment No. 48 to DPR-33
2. Safety Evaluation
3. Notice

cc w/enclosures:
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Agency
Region IV Office
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Atlanta, Georgia 30308

Mr. Robert F. Sullivan
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Decatur, Alabama 35602

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. 48
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 8, 1978, as supplemented by letters dated October 5, 1978, November 30, 1978, December 5, 1978, December 14, 1978, January 8, 1979, January 9, 1979 and January 23, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 48, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 8, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 48

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:

69/70
71/72
101/102
159/160

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

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
50 1(3)	Core Spray Loop A Discharge Pressure (PI-75-20)	0 - 500 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(3)	Core Spray Loop B Discharge Pressure (PI-75-48)	0 - 500 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(3)	RHR Loop A Discharge Pressure (PI-74-51)	0 - 450 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(3)	RHR Loop B Discharge Pressure (PI-74-65)	0 - 450 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(10)	Instrument Channel - RHR Start	N/A	A	1. Starts RHR area cooler fan when respective RHR motor starts.
1(10)	Instrument Channel - Thermostat (RHR Area Cooler Fan)	$\leq$ 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)	$\leq$ 100°F	A	1. Above trip setting starts Core Spray area cooler fans

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
1(10)	RHR Area Cooler Fan Logic	N/A	A	
1(10)	Core Spray Area Cooler Fan Logic	N/A	A	
1(11)	Instrument Channel - Core Spray Motors A or D Start	N/A	A	1. Starts RHRSW pumps A3 & D1
1(11)	Instrument Channel - Core Spray Motors B or C Start	N/A	A	1. Starts RHRSW pumps B1 & C3
1(12)	Instrument Channel - Core Spray Loop 1 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A3 & C3
1(12)	Instrument Channel - Core Spray Loop 2 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps B1 & D1
1(13)	RHRSW Initiate Logic	N/A	(14)	
1	RPT logic	N/A	(17)	1. Trips recirculation pumps on turbine control valve fast closure or stop valve closure > 30% power.

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS System is required by section 3.5 to be operable, there shall be two operable trip systems except as noted. If a requirement of the first column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the first column reduced by more than one, action B shall be taken.

   Action:

   A. Repair in 24 hours. If the function is not operable in 24 hours, take action B.

   B. Declare the system or component inoperable.

   C. Immediately take action B until power is verified on the trip system.

   D. No action required, indicators are considered redundant.

2. In only one trip system.

3. Not considered in a trip system.

4. Requires one channel from each physical location (there are 4 locations) in the steam line space.

5. With diesel power, each RHRS pump is scheduled to start immediately and each CSS pump is sequenced to start about 7 sec later.

6. With normal power, one CSS and one RHRS pump is scheduled to start instantaneously, one CSS and one RHRS pump is sequenced to start after about 7 sec with similar pumps starting after about 14 sec and 21 sec, at which time the full complement of CSS and RHRS pumps would be operating.

7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of H_2O corresponds to 300% of rated steam flow at 1140 psia and 210% at 165 psia. The HPCIS setting of 90 psi corresponds to 225% of rated flow at 1140 psia and 160% at 165 psia.

8. Note 1 does not apply to this item.

9. The head tank is designed to assure that the discharge piping from the CS and RHR pumps are full. The pressure shall be maintained at or above the values listed in 3.5.1, which ensures water in the discharge piping and up to the head tank.

NOTES FOR TABLE 3.2.B (Continued)

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 ($\geq$ 378" above vessel zero) originating in the core spray system trip system.

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

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

TABLE 4.[illegible].B (Continued)

Function	Functional Test	Calibration	Instrument Check
RHR Area Cooler Fan Logic	Tested during functional test of instrument channels, RHR motor start and thermostat (RHR area cooler fan). No other test required.	N/A	N/A
Core Spray Area Cooler Fan Logic	Tested during logic system functional test of instrument channels, core spray motor start and thermostat (core spray area cooler fan). No other test required.	N/A	N/A
Instrument Channel - Core Spray Motors A or D Start	Tested during functional test of core spray pump (refer to section 4.5.A).	N/A	N/A
Instrument Channel - Core Spray Motors B or C Start	Tested during functional test of core spray pump (refer to section 4.5,A).	N/A	N/A
Instrument Channel - Core Spray Loop 1 Accident Signal	Tested during logic system functional test of core spray system.	N/A	N/A
Instrument Channel - Core Spray Loop 2 Accident Signal	Tested during logic system functional test of core spray system.	N/A	N/A
RHRSW Initiate Logic	once/6 months	N/A	N/A
RPT initiate logic	once/month	N/A	N/A
RPT breaker	once/operating cycle	N/A	N/A

TABLE 4.2.C
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE ROD BLOCKS

Function	Functional Test	Calibration (17)	Instrument Check
APRM Upscale (Flow Bias)	(1) (13)	once/3 months	once/day (8)
APRM Upscale (Startup Mode)	(1) (13)	once/3 months	once/day (8)
APRM Downscale	(1) (13)	once/3 months	once/day (8)
APRM Inoperative	(1) (13)	N/A	once/day (8)
RBM Upscale (Flow Bias)	(1) (13)	once/6 months	once/day (8)
RBM Downscale	(1) (13)	once/6 months	once/day (8)
RBM Inoperative	(1) (13)	N/A	once/day (8)
IRM Upscale	(1)(2) (13)	once/3 months	once/day (8)
IRM Downscale	(1)(2) (13)	once/3 months	once/day (8)
IRM Detector not in Startup Position	(2) (once/operating cycle)	once/operating cycle (12)	N/A
IRM Inoperative	(1)(2) (13)	N/A	N/A
SRM Upscale	(1)(2) (13)	once/3 months	once/day (8)
SRM Downscale	(1)(2) (13)	once/3 months	once/day (8)
SRM Detector not in Startup Position	(2) (once/operating cycle)	once/operating cycle (12)	N/A
SRM Inoperative	(1)(2) (13)	N/A	N/A
Flow Bias Comparator	(1)(15)	once/operating cycle (20)	N/A
Flow Bias Upscale	(1)(15)	once/3 months	N/A
Rod Block Logic	(16)	N/A	N/A
RSCS Restraint	(1)	once/3 months	N/A

.5.H Maintenance of Filled Discharge Pipe

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

I. Average Planar Linear Heat Generation Rate

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

J. Linear Heat Generation Rate (LHGR)

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

4.5.H Maintenance of Filled Discharge Pipe

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

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

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

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

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

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

J. Linear Heat Generation Rate (LHGR)

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

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

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

$(\Delta P/P)_{max}$ = Maximum power spiking penalty
= 0.026 for 7X7 fuel
= 0.022 for 8X8 fuel

LT* = Total core length = 12.0 feet for 7X7 fuel
= 12.2 feet for 8X8 fuel

L = Axial position above bottom of core

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

K. Minimum Critical Power Ratio (MCPR)

The MCPR operating limit from the beginning of cycle 3 to the end of cycle 3 minus 2000 MWd/t is 1.20 for 7X7 fuel and 1.24 for 8X8 and 8X8R fuel; the limit from the end of cycle 3 minus 2000 MWd/t to the end of cycle 3 is 1.25 for 7X7 fuel and 1.30 for 8X8 and 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.

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.

* 12.5 feet for 8x8R fuel

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 48 TO FACILITY LICENSE NO. DPR-33

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-259

1.0 Introduction

By letter dated September 8, 1978, (TVA BFNP TS115), as supplemented by letters dated October 5, 1978, November 30, 1978, December 5, 1978, December 14, 1978, January 8, 1979 and January 9, 1979, the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit No. 1. In support of this reload application for Browns Ferry Unit No. 1 (BF-1), the licensee submitted a reload licensing document prepared by the General Electric Company (GE), a supplemental reload licensing document also prepared by GE and proposed changes to the Technical Specifications.

On January 17, 1979, we issued Amendment No. 47 to Facility License No. DPR-33 in response to the above submittals. The amendment authorized TVA to startup and operate BF-1 in the third fuel cycle. In the staff's safety evaluation accompanying this amendment, we evaluated the items requiring attention during reload reviews, including nuclear and mechanical design of the fuel, thermal-hydraulic, transient and accident analyses, startup testing programs, etc.; all staff concerns were satsifactorily resolved with the licensee. As discussed, below, the staff had reservations about the testability of the end-of-cycle recirculation pump trip (EOC RPT) feature which the licensee installed in BF-1 during the refueling outage. (The EOC RPT is different from and should not be confused with the recirculation pump trip feature which has been installed in several boiling water reactors to mitigate the consequences of anticipated transients without scram; the latter is commonly refered to as ATWS RPT). The staff's reservation about the testability of the EOC RPT was resolved by not including credit for the

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EOC RPT system in the operating limit minimum critical power ratios (OLMCPR) incorporated into the Technical Specifications with Amendment No. 47. (See section 3.2.2 of the accompanying safety evaluation). On January 17, 1979, a meeting was held with representatives of TVA and the General Electric Company (G.E.) to discuss the electrical design and testability of the EOC RPT. The information presented at this meeting resolved the staff's reservations on preoperational, startup and periodic testing of the EOC RPT system. This information was documented in TVA's letter of January 23, 1979.(1) Accordingly, this amendment revises the OLMCPRs to provide credit for the EOC RPT as justified by the analyses submitted by the letters referenced above.

2.0 Discussion

Section 14.5 of the Browns Ferry Nuclear Plant Final Safety Analysis Report (BFNP FSAR) discusses the analyses of abnormal operational transients. The events that could result in significant nuclear system pressure increases are those that result in a sudden reduction of steam flow while the reactor is operating at power. These possible events are: (1) generator trip, (2) loss of condenser vacuum, (3) turbine trip, (4) turbine bypass valve malfunction, (5) closure of main steam isolation valve and (6) pressure regulator malfunction.

During the refueling outage (November 26, 1978 to January 17, 1979), TVA installed an end-of-cycle RPT system (hereafter referred to as RPT system) in BF-1. This system provides automatic trip of both recirculation pumps after turbine trip or generator load rejection if reactor power is above approximately 30 percent of rated full load. The purpose of this trip is to reduce the peak reactor pressure and peak heat flux resulting from transients in which it is postulated that there is a coincident failure of the turbine bypass system. The recirculation pump trip signal results from either turbine control valve fast closure or turbine stop valve closure. Reactor scram is also initiated by these signals. Since the recirculation pump trip involves opening of circuit breakers between the motor-generator set and the pumps, the flow coastdown is more rapid than that resulting from loss of power to the motor-generator sets. The very rapid reduction in core flow following a recirculation pump trip early in these transients reduces the severity of the events because the immediate resultant increase in core voids provides negative reactivity which supplements the negative reactivity from control rod scram.

3.0 Evaluation

3.1 Operating Limit Minimum Critical Power Ratio (OLMCPR)

Various transient events will reduce the MCPR from its normal operating value. To assure that the fuel cladding integrity safety limit MCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed by the licensee to determine which event results in the largest reduction in critical power ratio. Each of the events has been conservatively analyzed for each of the several fuel types (i.e., 7x7, 8x8, 8x8R) and for the full range of exposure through the cycle.

The methods used for these calculations, including cycle-independent initial conditions and transient input parameters are described in Reference 2. Our acceptance of the values used and related transient analysis methods appear in Reference 3. Supplemental cycle-dependent initial conditions and transient input parameters used in the analysis appear in the table in Section 6 and 7 of Reference 4. Our evaluation of the methods used to develop these supplementary transient input values have already been addressed and appear in Reference 3. The overall transient methodology, including cycle-independent transient analysis inputs, provides an adequately conservative basis for the determination of transient ΔCPRs. The transient events analyzed were load rejection without bypass, turbine trip without bypass, feedwater controller failure, loss of 100°F feedwater heating and control rod withdrawal error.

In the analysis of these reactor transients, the licensee has proposed to take credit for an EOC recirculation pump trip (RPT). This reduces the transient ΔCPR during reactor core pressurization events (e.g., load rejection, turbine trip) by tripping breakers in the electrical line between the motor-generator sets and the recirculation pumps on closure of turbine stop or control valves. The prompt RPT immediately reduces core flow and thereby increases core voids. The immediate voiding provides negative reactivity which supplements scram reactivity. In this manner, the RPT reduces the thermal power spiking during the pressurization events. This RPT feature is a margin improvement option which was not generically approved in our evaluation of the reference reload topical.[3]

The ΔCPR credit for the prompt RPT was calculated with the REDY code. The REDY code employs a two node steamline thermal hydraulic model and a point kinetics neutronics model. Several pressurization experiments at Peach Bottom Unit 2 (Reference 5) were designed to check the validity of these REDY models.

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

We agreed with GE's general conclusion that REDY provides a conservative calculation for the current licensing basis transients on operating reactors. However, we also recognized that REDY's inability to accurately predict pressurization rate and axial reactivity response limits simulation of effects of RPT. The Peach Bottom tests demonstrated the existence of a pressure wave phenomenon in the steam lines.(6,7) In addition, it was noted that the power rise associated with pressurization was significantly greater in the upper portion of the core than in the lower portion.

Quantitative comparison of the tests with REDY calculations indicated that the REDY model underpredicted the pressurization rate but overpredicted the core's response to pressurization effects. Thus, there are two discrepancies between REDY simulated effects and real transient's effects. One is non-conservative and the other is conservative. It is impossible to state from these comparisons which effect would predominate for a given transient.

After the analysis of the tests, comparisons were made between REDY simulations and simulations using detailed steamline modeling and a time-varying axial power distribution.(8) These comparisons, although rather limited, suggest a trend in which REDY-based calculations conservatively predicted ΔCPR for more severe transients but underpredict ΔCPR (for a given set of input parameters) for less severe transients.(8) These calculations also showed that the ΔCPR benefits for the RPT feature may be overpredicted by REDY as compared to the detailed steamline and core modeling predictions.

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

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

Two alternatives suggest themselves as means of resolution. The first alternative is to provide additional justification for the proposed specification. The GE ODYN code has more nodes to model steamline dynamics than REDY and also has a one-dimensional axial core neutronics model. ODYN's development has been based on first principles and verified by the Peach Bottom tests. ODYN is currently under a staff review that is to be complete within the next few months. ODYN will be used as the calculational model for pressurization events when it is approved.

Prior to its approval, we find that ODYN could be used to simulate the RPT effects and, thereby, provide assurance of the ΔCPR benefit. During this time, we will accept the greater ΔCPR of the ODYN and REDY calculations. Once ODYN receives generic approval, we will accept the ODYN calculated ΔCPR regardless.

We are not <u>requiring</u> an ODYN calculation, however. We have made it clear that we will evaluate <u>any</u> other justification which the licensee submits and all applicable calculations and data which become available to us through other channels.

Another alternative to an adequate ΔCPR for RPT effects is to conservatively bound the REDY calculation. In a previous reload safety evaluation for Browns Ferry Unit 3,[9] we found that an increase of 0.05 in ΔCPR for the limiting pressure and power increase transients or an increase of .07 in ΔCPR for the feedwater controller failure transient (whichever is more limiting) will bound the potential non-conservatism in the analyses. In the subsequent reload amendment,[10] operating limit MCPRs for only the initial 2000 MWd/t of Cycle 2 were provided, with the statement that MCPRs to end of cycle will be determined by reanalysis. This position was reached from the licensee's insistence to not implement the conservatively adjusted staff required MCPR and the licensee's preference to rely on a timely GE core specific ODYN reanalysis with staff review.

Since this amendment was issued, we have had several telephone conversations and meetings with TVA. From these discussions, we have concluded that the increase in ΔCPR may be somewhat reduced. This conclusion is based on the fact that the previously established increase was in part established from ODYN comparisons to REDY for measured scram times. These calculations would emphasize the core axial modeling differences between the codes and, thus, enhance the differences by virtue of the phenomenon modeling characteristics. With this, the ODYN-REDY comparison for the measured scram time can be somewhat de-emphasized and the ΔCPR differences between ODYN and REDY for the RPT can be considered to provide the primary indication of the effect of the "more sophisticated" modeling. The only available comparison of ODYN and REDY for the RPT shows a ΔCPR difference of about 0.02. This calculation is for a specific BWR which is different in plant size and core loading than the Browns Ferry Units. On these bases, we and the licensee have agreed that a conservative bound to the REDY calculation with RPT would be assured with a 0.03 ΔCPR increase for rapid pressurization transients.

Based on our composite review of the licensee's submittals and 0.03 MCPR increase, the most limiting abnormal operational transient for all fuel types and exposure intervals except for the 7x7 fuel from BOC to EOC-2000 MWd/t is the load rejection without bypass. For the 7x7 fuel from BOC to EOC-2000 MWd/t, the limiting transient is the loss of 100°F feedwater heating.

The operating limit MCPRs which the licensee has proposed(11) and which are acceptable to the staff are as follows:

	OPERATING LIMIT MCPR	
Fuel Type	EOC-2000 MWd/t	EOC
7x7	1.20	1.25
8x8	1.24	1.30
8x8R	1.24	1.30

Thus, when the reactor is operated in accordance with the above operating limit MCPRs the 1.07 SLMCPR will not be violated in the event of the most severe abnormal operational transient. This is acceptable to the staff.

3.2 Failure of Trip Inputs from Turbine Building to Reactor Protection System

During our review of the reactor protection system, we noted that the trip inputs for the recirculation pump trip and reactor scram following load rejection or turbine trip originate in the turbine building. The turbine building, as is the case of most boiling water reactor plants, is not seismically qualified; hence, its integrity and functions cannot be assured in the event of an earthquake.

For these reasons, the licensee was requested to analyze the consequences of a safe shutdown earthquake concurrent with the limiting transient event without taking credit for reactor scram or recirculation pump trip from the turbine building inputs. The licensee has referenced generically applicable analyses. We agree with the licensee that this analysis is applicable and on the basis of previous staff findings on this analysis,[12] we find the results acceptable.

3.3 Electrical Control and Instrumentation Aspects of EOC RPT

The design philosophy for the EOC RPT system is described in GE report NEDO-24119, "Basis for Installation of Recirculation Pump Trip System", Browns Ferry Nuclear Plant, April 1978[13]. This report provides the safety evaluation of the instrumentation and control aspects of the proposed modification. The design of the RPT system was evaluated against the criteria of IEEE Standard 279.

The RPT feature serves as an essential safety supplement to the scram system and, as such, is required to comply with IEEE Standard 279. Basically, the RPT feature consists of hydraulic pressure switches (sensors to detect the fast closure of the turbine control valves), position switches (sensors to detect closure of the turbine stop valves), relays, logic, and fast-acting circuit breakers (actuation devices). In order to satisfy the single failure criterion, the RPT logic feature consists of two almost-identical systems in a one-out-of-two configuration such that either is capable of operating independent circuit breakers in the supply circuit of each recirculation pump motor.

The operation of any RPT sensor (pressure switch or position switch from any of the four turbine control valves or any of the four turbine stop valves) causes an electromagnetic relay to de-energize. The relay contacts are combined with contacts from an Operating Bypass and contacts from a manual bypass switch to provide power to the breaker trip coils. The turbine valve sensors and turbine pressure sensors for the RPT feature are the same ones used for the scram system. The Operating Bypass disables the RPT system when the turbine first-stage pressure is below about 30%, as is done for the turbine inputs to the scram system. The manual bypass switch ("out-of-service") allows each RPT system to be disabled for maintenance purposes.

A fast closure sensor from each of two turbine control valves provides input to one RPT system; sensors from the other two turbine control valves provide inputs to the second RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one RPT system; sensors from the other two stop valves provide imputs to the other RPT system. For each RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of control valves and a 2-out-of-2 logic for the stop valves. The operation of either logic will actuate the RPT feature.

Information initially provided by GE and TVA indicated that the RPT feature should function when scram occurs for either of the two turbine transients. However, the criterion that was finally established by GE and TVA requires the RPT to function only if all four turbine stop valves or all four control valves close. That is, if only three of the stop valves close, scram will occur but RPT need not and might not. No single failure within the RPT will prevent the RPT from providing system-level protective action in accordance with the final criterion.

TVA has stated that the RPT equipment will be appropriately qualified as Class IE. We note that inputs to the RPT originate in the turbine building which is not a seismic category 1 structure. The input equipment is adequately qualified for the anticipated occurrences of the turbine. As discussed in Section 3.2 above, we have evaluated this aspect of the design with respect to the potential consequences of a safe shutdown earthquake and conclude that there will be no undue risk to the public health and safety from this postulated event.

The two systems of the RPT feature will be physically and electrically independent. There is one interconnection between the RPT and a non-safety system. When the RPT is tripped, auxiliary relay contacts feed the control circuits of the motor-generator sets to deenergize them. This interlock is adequately isolated such that no credible failure can prevent proper action of the RPT.

Although the purpose of the RPT is to mitigate a core-wide pressurization transient, the desired thermal margin advantage can be realized only if the initiating events are sensed on an anticipatory basis, rather than monitoring pressure directly. The use of pressure switches to sense the loss of hydraulic control fluid pressure to each control valve is adequate to anticipate fast closure of those valves. Similarly, position switches set at $\geq 90\%$ open will anticipate closure of the turbine stop valves. The RPT is not given credit for any other initiating events.

Each RPT system may be manually bypassed by use of a keyswitch ("cut-of-service") which is administratively controlled. Both the manual bypasses and the Operating Bypass ($< 30\%$ power) are annunciated automatically and distinctively in the control room.

Unlike the scram system, the RPT logic is not fail-safe (i.e., does not go to tripped state upon loss of electric power). The sensor relays will go to the tripped state on loss of power, but the RPT logic circuits are "power-on-to-trip". For a 4160V circuit breaker, electric power is required to operate the trip coil. For this design, the logic circuits and trip coils operate on 250 vdc. A total of four sets of Class 1E batteries can provide the 250 vdc for the RPT systems - 2 per system. An alarm is provided to indicate loss of power to either RPT logic. This departure from the fail-safe design of scram systems is acceptable.

Each RPT system will go to completion in that after the system causes the circuit breakers to trip, the operator is required to manually reset the breaker.

To be effective the RPT must be initiated virtually immediately. TVA has stated that their analysis shows that manual initiation of a prompt trip of the recirculation pumps at any reasonable point after the time when automatic action should have occurred will not have a significant improvement on the situation. The power to the motor-generator sets can be tripped manually from the control room. Therefore, provisions for manual initiation of the EOC RPT feature are unnecessary.

As described above, a fast closure sensor from each of two turbine control valves provides input to one RPT system; sensors from the other two turbine control valves provide inputs to the second RPT system. Similarly, position switches for each of two turbine stop valves provide input to one RPT system; sensors from the other two stop valves provide inputs to the other RPT system. The staff performed a failure mode analysis of the system. The most severe postulated failure would be the failure of one of the four turbine control valves or turbine stop valves to close (thus not providing an actuation signal to the one RPT division) with simultaneous failure of the bypass valves to open and simultaneous failure of the other RPT system. However, in this case, the open line to the turbine would in effect be acting as a bypass. The licensee was requested to verify that the RPT feature need not actuate for the condition where one of either the turbine stop or control valves does not close.

The licensee referenced calculation of load rejection with bypass and turbine trip with bypass as supportive evidence. We have reviewed the flow capacities of the bypass and open steamline configurations and have evaluated the dynamic pressure response associated with each configuration. We have concluded that the pressurization transient analyses with bypass provide a conservative estimate of the effect of failure of one or more turbine stop or control valve to close. The results of the pressurization transient with bypass and without RPT indicate that the transient is not limiting. Therefore, the design of the RPT system to respond for only the condition where all turbine stop or control valves close is acceptable.

3.4 RPT Testability

Capability to check the RPT sensors and logic is provided by operating each valve one at a time. Lights across the relay contacts in the logic indicate proper operation at that point. The RPT systems do not need to be bypassed to conduct such tests. However, during the periodic testing of the scram logic, where two valves are operated simultaneously, the affected RPT system must be bypassed briefly to prevent RPT actions. Appropriate technical specifications cover this situation. The RPT circuit breakers will be functionally tested during refueling outages.

The RPT feature is required to interrupt the pump motor circuit within 175 milliseconds of the start of valve closure. Of this, 10 ms is allotted for system action and sensor response, 30 ms for logic response, and 135 ms for breaker action. Because the EOC RPT feature must function so promptly, we require that appropriate response time testing be conducted at each refueling shutdown. TVA is not in complete agreement with this staff requirement, but has committed (in a letter dated January 23, 1979, J. E. Gilleland to T. A. Ippolito) to submit their proposal on this matter no later than July 23, 1979 (well before the next refueling). The RPT system will be tested preoperational to verify response time.

The final design of the RPT feature for Browns Ferry Unit No. 1 adequately complies with IEEE Standard 279 for its stated purpose. All parts of the RPT feature are appropriately qualified to mitigate appropriate anticipated pressure transients from the turbine. We conclude, therefore, that the design is acceptable. The matter of periodic (refueling interval) response time testing is being deferred and must be completed prior to the next refueling outage for this unit.

4.0 Environmental Considerations

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

5.0 Conclusion

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

Dated: February 8, 1979

References:

1. Letter, TVA (Gilleland) to USNRC (Ippolito) dated January 23, 1979.

2. "Generic Reload Fuel Application," General Electric Report, NEDE-24011-P-3, dated March 1978.

3. USNRC letter (Eisenhut) to General Electric (Gridley) dated May 12, 1978, transmitting "Safety Evaluation for the General Electric Topical Report, 'Generic Reload Fuel Application,' (NEDE-24011-P)."

4. TVA letter (Gilleland) to USNRC (Denton) dated November 30, 1978, "Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Plant Unit 1 Reload 2," NEDO-24136, Rev. 1, November 1978.

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

6. Letter, R. Engel (GE) to Office of Nuclear Reactor Regulation (NRC), dated July 11, 1977.

7. Letter, E. D. Fuller (GE) to U. S. Nuclear Regulatory Commission, dated October 25, 1977.

8. "Impact of One-Dimensional Transient Model on Plant Operating Limits," enclosure of letter, E. D. Fuller (GE) to U. S. Nuclear Regulatory Commission, dated June 26, 1978.

9. Memo from P. S. Check (RSB-NRC) to T. A. Ippolito (ORB#3-NRC), "Browns Ferry 3 - Cycle 2 Reload (TACS #8026)," November 8, 1978.

10. NRC letter (Ippolito) to TVA (Hughes), Amendment Nos. 45, 41, and 18 to Facility License No. DPR-33, DPR-52, and DPR-68 for Browns Ferry Nuclear Plant Units Nos. 1, 2, and 3, dated November 18, 1978.

11. Letter, TVA (Gilleland) to USNRC (Denton) dated January 8, 1979, TVA BFNP TS 115.

12. Safety Evaluation Report related to operation of Edwin I. Hatch Nuclear Plant, Unit No. 2, Georgia Power Company, et. al., USNRC, Office of Nuclear Reactor Regulation, Docket No. 50-366, NUREG-0411, June 1978.

13. Letter, TVA (Gray) to USNRC (Case) dated July 18, 1978, TVA BFNP TS 111.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-259

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 48 to Facility Operating License No. DPR-33 issued to Tennessee Valley Authority (the licensee), which revised the Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit No. 1 (the facility) located in Limestone County, Alabama. The amendment is effective as of the date of issuance.

This amendment permits operation of Browns Ferry Unit No. 1 in Cycle No. 3 following the second refueling outage.

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

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

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For further details with respect to this action, see (1) the application for amendment dated September 8, 1978, as supplemented by letters dated October 5, 1978, November 30, 1978, December 5, 1978, December 14, 1978, January 8, 1979, January 9, 1979, and January 23, 1979, (2) Amendment No. 48 to License No. DPR-33, 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 8th day of February 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

T. Ippolito

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