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

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January 10, 1978

Pasted Am-32 to DPR-52

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

Tennessee Valley Authority
ATTN: Mr. Godwin Williams, Jr.
Manager of Power
818 Power Building
Chattanooga, Tennessee 37201

Gentlemen:

This is in response to applications for amendments dated January 12, May 11, July 8, September 23, 26, 27, October 28, November 16, December 13, 1977, and January 3, 1978.

Amendment No. 35 to DPR-33 changes the Technical Specifications to incorporate the limiting conditions for operation associated with Cycle 2 operation of Browns Ferry Nuclear Plant , Unit 1. These changes involve a revised fuel cladding integrity safety limit for minimum critical power ratio (MCPR), revised operating limit MCPR's for both 7x7 and 8x8 fuel assemblies, the addition of linear heat generation rate (LHGR) limits for the 8x8 fuel, revised limits for the maximum average planar linear heat generation rate (MAPLHGR) for the 7x7 and 8x8 fuel assemblies, and reduced limits for scram insertion times. The revised limits for the MAPLHGR result from your reanalysis of the Emergency Core Cooling System performance in response to the Commission's Order of March 11, 1977. We have found your reanalysis to be acceptable. Effective upon issuance of this amendment, the Commission's Order for Modification of License dated March 11, 1977, relative to Facility Operating License No. DPR-33, is terminated. In addition, a restriction on power operation during the initial startup for Cycle 2 has been imposed until sufficient high temperature recirculation has taken place to ensure disintegration of a rubber shoecover that had fallen into the Unit 1 vessel during the refueling outage.

Amendment Nos. 35 to DPR-33, 32 to DPR-52, and 9 to DPR-68 change the Technical Specifications for each of the Browns Ferry Nuclear Plant Units to clarify the operability requirements of the Rod Worth Minimizer and the Rod Sequence Control System during scram time testing, delete the Annual Operating Report requirements, add standards for qualifications of the Health Physics Supervisor, change the frequency of cycling fire protection system valves from quarterly to annually, and substitute revised, but equivalent, terms in the equations for the limiting settings on the Average Power Range Monitors' scram and rod block setpoints.

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

Sincerely,

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
Amendment No. 35 to DPR-33
Amendment No. 32 to DPR-52
Amendment No. 9 to DPR-68
Safety Evaluation
Notice

cc w/enclosures:
See next page

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. 32
License No. DPR-52

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

   A. The applications for amendments by Tennessee Valley Authority (the licensee) dated January 12, May 11, July 8, September 23, 26, 27, October 28, November 16, December 13, 1977, and January 3, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility License No. DPR-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. 32, 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

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance January 10, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 32

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

Remove the following pages and replace with identically numbered pages:

5/6	123/124
9/10	129/130
15/16	133/134
19/20	181/182
21/22	315/316
23/24	327/328
31/32	331/332
47/48	349/350
73/74	351/352

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

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All Reactor Building ventilation system automatic isolation valves are operable or deactivated in the isolated position.

Q. Operating Cycle - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

R. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the unit after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

S. Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation and the traversing in-core probe is not defined as a core alteration.

T. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

U. Thermal Parameters

1. Minimum Critical Power Ratio (MCPR) - Minimum Critical Power Ratio (MCPR) is the value of the critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power.

2. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

3. Core Maximum Fraction of Limiting Power Density (CMFLPD) - The highest ratio, for all fuel types in the core, of the maximum fuel rod power density (kW/ft) for a given fuel type to the limiting fuel rod power density (kW/ft) for that fuel type.

4. Average Planar Linear Heat Generation Rate (APLHGR) - The Average Planar Heat Generation Rate is applicable to a specific planar height and is equal to the sum of the linear heat generation rates for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

1.0 DEFINITIONS (Cont'd)

V. Instrumentation

1. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors.

2. Channel - A channel is an arrangement of a sensor and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.

3. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument channel response, alarm and/or initiating action.

4. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

5. Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit to insure all components are operable per design intent. Where practicable, action will go to completion; i.e., pumps will be started and valves operated.

6. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

7. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

8. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

9. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

B. Core Thermal Power Limit (Reactor Pressure ≤800 psia)

When the reactor pressure is less than or equal to 800 psia,

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

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

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

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

(Note: These settings assume operation within the basic thermal hydraulic design criteria. These criteria are LHGR 18.5 kw/ft and MCPR ≥ (1.25 if <8000 MWD/T; 1.29 otherwise). If it is determined that either of these design criteria is being violated during operation, action shall be initiated within 15 minutes to restore operation within prescribed limits. Surveillance requirements for APRM scram setpoint are given in specification 4.1.B.

2. APRM--When the reactor mode switch is in the STARTUP POSITION, the APRM scram shall be set at less than or equal to 15% of rated power.

3. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.

B. APRM Rod Block Trip Setting

The APRM Rod block trip setting shall be:

1.1 FUEL CLADDING INTEGRITY

or core coolant flow is less than 10% of rated, the core thermal power shall not exceed 823 MWt (about 25% of rated thermal power).

C. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 17.7 in. above the top of the normal active fuel zone.

2.1 FUEL CLADDING INTEGRITY

$$S_{RB} \leq (0.66W + 42\%)$$

where:

S_{RB} = Rod block setting is percent of rated thermal power (3293 MWt)

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

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

$$S_{RB} \leq (0.66W + 42\%) \frac{FRP}{CMFLPD}$$

C. Scram and isoluation--reactor low water ≥ 538 in. above vessel zero leve

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

E. Scram--turbine control valve

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

2. Loss of control oil pressure ≥ 550 psig

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

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

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

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 performation 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 1.05. MCPR >1.05 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possiblity of clad failure. Since boiling transition is not a directly observable parameter, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables, i.e., normal plant operation presented on Figure 2.1.1 by the nominal expected flow control line. The Safety Limit (MCPR of 1.05) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR > 1.25); 1.25 if core average exposure is ≥ 8000 MWD/T 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 1.05 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.

The MCPR value used in the ECCS performance evaluation (1.18) is less limiting than the MCPR for operation (1.25); 1.29 if core average exposure is ≥ 8000 MWD/T.

.1.1 BASES

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of MCPR = 1.05 would not produce boiling transition. Thus, although it is not required to establish the safety limit additional margin exists between the safety limit and the actual occurence of loss of cladding integrity.

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

If reactor pressure should ever exceed 1400 psia during normal power operating (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.
In addition to the boiling transition limit (MCPR = 1.05) operation is constrained to a maximum LHGR of 18.5 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 Core Maximum Fraction of Limiting Power Density exceeds the Fraction of Rated Thermal Power, operation is permitted only at less than 100% of rated power and only with reduced APRM scram settings as required by specification 2.1.A.1.
At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flow will always be greater than 4.56 psi. Analyses show that with a flow of $28X10^3$ 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 $28x10^3$ lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

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

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 comparisions 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 1.25 (1.29 if core average exposure is $\geq$ 8000 MWD/T) 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.

Steady-state operation without forced recirculation will not be permitted for more than 12 hours. and the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

2.1 BASES

In summary

1. The licensed maximum power level is 3,293 MWt.

2. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

3. The abnormal operational transients were analyzed to a power level of 3440 MWT.

4. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

The bases for individual set points are discussed below:

A. Neutron Flux Scram

1. APRM High Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during transients induced by disturbances, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses reported in Section 14 of the Final Safety Analysis Report demonstrated that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage. Therefore, use of a flow-biased scram provides even additional margin. Figure 2.1.2 shows the flow biased scram as a function of core flow.

An increase in the APRM scram setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM setting was selected because it provides adequate margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibili. of unnecessary scrams.

3.B Control Rods

b. During the shutdown procedure no rod movement is permitted between the testing performed above 20% power and the reinstatement of the RSCS restraints at or above 20% power. Alignment of rod groups shall be accomplished prior to performing the tests.

c. Whenever the reactor is in the startup or run modes below 20% rated power the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.

A second licensed operator may not be used in leiu of the RWM during scram time testing in the startup or run modes below 20 percent of rated thermal power.

d. If Specifications 3.3.B.3.a through .c cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 20% rated power, it shall be brought to a shutdown condition immediately.

4.3.B Control Rods

a. The capability of the RSCS to properly fulfill its function shall be verified by the following tests:

Sequence portion - Select a sequence and attempt to withdraw a rod in the remaining sequences. Move one rod in a sequence and select the remaining sequences and attempt to move a rod in each. Repeat for all sequences.

Group notch portion - For each of the six comparator circuits go through test initiate; comparator inhibit; verify; reset. On seventh attempt test is allowed to continue until completion is indicated by illumination of test complete light.

b. The capability of the Rod Worth Minimizer (RWM) shall be verified by the following checks:

1. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified before reactor startup or shutdown.

2. The RWM computer on line diagnostic test shall be successfully performed.

3. Prior to startup, proper annunciation of the selection error of at least one out-of-sequence control rod shall be verified.

4. Prior to startup, the rod block function of the RWM shall be verified by moving an out-of-sequence control rod.

5. Prior to obtaining 20% rated power during rod insertion at shutdown, verify the latching of the proper rod group and proper annunciation after insert errors.

3.3.B Control Rods

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. During operation with limiting control rod patterns, as determined by the designated qualified personnel, either:

   a. Both RBM channels shall be operable:
   or

   b. Control rod withdrawal shall be blocked.

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	5.0

4.3.B Control Rods

c. When required, the presence of a second licensed operator to verify the following of the correct rod program shall be verified.

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and at least once per 24 hours thereafter.

C. Scram Insertion Times

*1. After each refueling outage all operable rods shall be scram time tested from the fully withdrawn position with the nuclear system pressure above 950 psig (with saturation temperature). This testing shall be completed prior to exceeding 40% power. Below 20% power, only rods in those sequences (A_{12} and A_{34} or B_{12} and B_{34}) which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram time tested. The sequence restraints imposed upo the control rods in the 100-50 percent rod density groups to the preset power level may be removed by use of the individual bypass switches associated with those control rods which are fully or partially withdrawn and are not within the 100-50 percent rod densi groups. In order to bypass a rod, the actual rod axial position must known; and the rod must be in the correct in-sequence position.

TABLE 3.2
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum No. Operable Per Trip Sys (5)	Function	Trip Level Setting
2(1)	APRM Upscale (Flow Bias)	$\leq$ 0.66W + 42% (2)
2(1)	APRM Upscale (Startup Mode) (8)	$\leq$ 12%
2(1)	APRM Downscale (9)	$\geq$ 3%
2(1)	APRM Inoperative	(10_b)
1(7)	RBM Upscale (Flow Bias)	$\leq$ 0.66W + 41% (2)
1(7)	RBM Downscale (9)	$\geq$ 3%
1(7)	RBM Inoperative	(10_c)
3(1)	IRM Upscale (8)	$\leq$ 108/125 of full scale
3(1)	IRM Downscale (3)(8)	$\geq$ 5/125 of full scale
3(1)	IRM Detector not in Startup Position (8)	(11)
3(1)	IRM Inoperative (8)	(10^a)
2(1)(6)	SRM Upscale (8)	$\leq 1 \times 10^5$ counts/sec.
2(1)(6)	SRM Downscale (4)(8)	$\geq$ 3 counts/sec.
2(1)(6)	SRM Detector not in Startup Position (4)(8)	(11)
2(1)(6)	SRM Inoperative (8)	(10a)
2(1)	Flow Bias Comparator	$\leq$ 10% difference in recirculation flows
2(1)	Flow Bias Upscale	$\leq$ 110% recirculation flow
1(1) 2(1)	Rod Block Logic RSCS Restraint (PS-85-61A & PS-85-61B)	N/A 147 psig turbine first stage pressure (approximately 30% power)

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM, IRM, and APRM (Startup mode), blocks need not be operable in "Run" mode, and the APRM (Flow biased) and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition last longer than seven days, the system with the inoperable channel shall be tripped. If the first column cannot be met for both trip systems, both trip systems shall be tripped.

2. W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (3293 MWt). A ratio of FRP/CMFLPD <1.0 is permitted at reduced power. See Specification 2.1 for APRM control rod block setpoint.

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is ≥ 100 cps and IRM above range 2.

5. One instrument channel; i.e., one APRM or IRM or RBM, per trip system may be bypassed except only one of four SRM may be bypassed.

6. IRM channels A, E, C, G all in range 8 bypasses SRM channels A & C functions.

   IRM channels B, F, D, H all in range 8 bypasses SRM channels B & D functions.

7. The trip is bypassed when the reactor power is ≤ 30%.

8. This function is bypassed when the mode switch is placed in Run.

9. This function is only active when the mode switch is in Run. This function is automatically bypassed when the IRM instrumentation is operable and not high.

10. The inoperative trips are produced by the following functions:

    a. SRM and IRM

       (1) Local "operate-calibrate" switch not in operate.

       (2) Power supply voltage low.

       (3) Circuit boards not in circuit.

    b. APRM

       (1) Local "operate-calibrate" switch not in operate.

       (2) Less than 14 LPRM inputs.

       (3) Circuit boards not in circuit.

The frequency of calibration of the APRM Flow Biasing Network has been established as each refueling outage. There are several instruments which must be calibrated and it will take several hours to perform the calibration of the entire network. While the calibration is being performed, a zero flow signal will be sent to half of the APRM's resulting in a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Network, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month; i.e., in the period of a month a drift of .4% would occur and thus providing for adequate margin. For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.A and 4.1.B indicates that two instrument channels have not been included in the latter table. These are: mode switch in shutdown and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable, i.e., the switch is either on or off.

The ratio of Core Maximum Fraction of Limiting Power Density (MFLPD) to Fraction of Rated Power (FRP) shall be checked out once per day to determine if the APRM scram requires adjustment. This will normally be done by checking the LPRM readings. Only a small number of control rods are moved daily

during steady-state operation and thus the ratio is not expected to change significantly.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. This is compensated for in the APRM system by calibrating every 7 days using heat balance data and by calibrating individual LPRM's every 1000 effective full-power hours using TIP traverse data.

3.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective

To assure the operability of the reactor protection system.

Specification

When there is fuel in the vessel, the setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.A.

4.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.A and 4.1.B respectively.

B. Daily during reactor power operation at greater than or equal to 25% thermal power, the ratio of Fraction of Rated Power (FRP) to Core Maximum Fraction of Limiting Power Density (CMFLPD) shall be checked and the scram and APRM Rod Block settings given by equations in specifications 2.1.A.1 and 2.1.B shall be calculated.

C. When it is determined that a channel is failed in the unsafe condition, the other RPS channel that monitor the same variable shall be functionally tested immediately before the trip system containing the failure is tripped. The trip system containing the unsafe failure may be untripped for short periods of time to allow functional testing of the other trip system. The trip system may be in the untripped position for no more than eight hours per functional test period for this testing.

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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 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 14.5 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 1.05 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 scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of $\leq$ 10 percent of valve closure from full open, the resultant increase in bundle power is limited such that MCPR remains above 1.05 even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first stage pressure. Actuation of the relief valves limits pressure to well below the safety valve setting.

E. Turbine Control Valve Scram

1. Fast Closure Scram

The reactor protection system initiates a scram within 30 Msec after the control valves start to close. This setting and the fact that control valve closure time is approximately twice as long as that for the stop valves means that resulting transients, while similar, are less severe than for stop-valve closure. No fuel damage occurs, and reactor system pressure does not exceed the relief valve set point, which is approximately 280 psi below the safety limit.

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 scram set at 10 percent of valve closure, neutron flux does not increase.

2.1 BASES

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 limting transients show that no scram adjustment is required to assure MCPR >1.05 when the transient is initiated from MCPR > 1.25 (1.29 if core average exposure is ≥ 8000 MWD/T).

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 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

3. IRM Flux Scram Trip Setting (Continued)

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. In addition, the APRM 15% scram prevents 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, 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 1.05. 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 1.05. This rod block trip setting, which is automatically varried 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

The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 20 percent, these devices force adherence to acceptable rod patterns. Above 20 percent of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 20 percent of rated power are imposed by power distribution requirements, as defined in Sections 3.5.I, 3.5.J, 4.5.I, and 4.5.J of these technical specifications. Power level for automatic bypass of the RSCS function is sensed by first stage turbine pressure.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It

3. The Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to pre-specified sequences. All patterns associated with these sequences have the characteristic that, assuming the worst single deviation from the sequence, the drop of any control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in any pellet average enthalpy in excess of 280 calories per gram. An enthalpy of 280 calories per gram is well below the level at which rapid fuel dispersal could occur (i.e., 425 calories per gram). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Sections 3.6.6, 7.7.A, 7.16.5.3, and 14.6.2 of the FSAR and NEDO-10527 and supplements thereto.

In performing the function described above, the RWM and RSCS are not required to impose any restrictions at core power levels in excess of 20 percent of rated. Material in the cited reference shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 20 percent, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize individual control rod worth.

At power levels below 20 percent of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calorie per gram rod drop limit. In this range the RWM and the RSCS constrain the control rod sequences and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. Ref. Section 7.16.5.3 of the FSAR. They serve as a backup to procedure control of control rod sequences, which limit the maximum reactivity worth of control rods. In the event that the Rod Worth Minimizer is out of service, when required, a second licensed operator can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is back up by independent procedural controls to assure conformance.

* Because it is allowable by bypass certain rods in the RSCS during scram time testing below 20 percent of rated power in the startup or run modes, a second licensed operator is not an acceptable substitute for the RWM during this testing.

(b). Annual Operating Report

A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,* e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate Regional Office, to be submitted no later than the tenth of each month following the calendar month covered by the report. A narrative summary of operating experience shall be submitted in the above schedule.

2. Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

a. Prompt Notification With Written Followup. The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the appropriate Regional Office, or his designee no later than the first working day following the event, with a written followup report within two weeks. The written followup report shall include, as a minimum, a complete copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

Note: Instrument drift discovered as a result of testing need not be reported under this item but may be reportable under items 2.a(5), 2.a(6), or 2.b(1) below.

(2) Operation of the unit or affected systems when any parameter or operation subject to a limiting condition is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

Note: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the technical specifications, the limiting condition for operation is not considered to have been violated and need not be reported under this item but it may be reportable under item 2.b(2) below.

(3) Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

6.0 ADMINISTRATIVE CONTROLS

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6.0 ADMINISTRATIVE CONTROLS

6.7 Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.

1. Routine Reports

   a. Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

   Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commerical power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

A. The plant superintendent has on-site responsibility for the safe operation of the facility and shall report to the Chief, Nuclear Generation Branch. In the absence of the plant superintendent, the assistant superintendnet will assume his responsibilities.

B. The portion of TVA management which relates to the operation of the plant is shown in Figure 6.1-1.

C. The functional organization for the operation of the station shall be as shown in Figure 6.1-2.

D. Shift manning requirements shall, as a minimum, be as described in section 6.8.

E. Qualifications of the Browns Ferry Nuclear Plant management and operating staff shall meet the minimum acceptable levels as described in ANSI - N18.1, Selection and Training of Nuclear Power Plant Personnel, dated March 8, 1971. The qualifications of the Health Physics Supervisor will meet or exceed the minimum acceptable levels as described in Regulatory Guide 1.8, Revision 1, dated Sept. 1975.

F. Retraining and replacement training of station personnel shall be in accordance with ANSI - N18.1, Selection and Training of Nuclear Power Plant Personnel, dated March 8, 1971. The minimum frequency of the retraining program shall be every two years.

G. An Industrial Security Program shall be maintained for the life of the plant.

H. Responsibilities of a post-fire overall restoration coordinator will consist of duties as described in section 6.9.

I. The Safety Engineer shall have the following qualifications:

a. Must have a sound understanding and thorough technical knowledge of safety and fire protection practices, procedures, standards, and other codes relating to electrical utility operations. Must be able to read and understand engineering drawings. Must possess an analytical ability for problem solving and data analysis. Must be able to communicate well both orally and in writing and must be able to write investigative reports and prepare written procedures. Must have the ability to secure the cooperation of management, employees and groups in the implementation of safety programs. Must be able to conduct safety presentations for supervisors and employees.

b. Should have experience in safety engineering work at this level or have 3 years experience in safety and/or fire protection engineering. It is desirable that the incumbent be a graduate of an accredited college or university with a degree in inductrial, mechanical, electrical, or safety engineering or fire protection engineering.

B. The k_{eff} of the spent fuel storage pool shall be less than or equal to 0.90 for normal conditions and 0.95 for abnormal conditions (Sections 10.3 of the FSAR).

5.6 SEISMIC DESIGN

The station class I structures and systems have been designed to withstand a design basis earthquake with ground acceleration of 0.2g. The operational basis earthquake used in the plant design assumed a ground acceleration of 0.1g (see Section 2.5 of the FSAR).

circuit was selected because it contained 2 out of 3 detector logic, the most complicated CO_2 circuit logic. Calculations were based on failure rates for wires, connections, and circuit components as shown in Appendix III of WASH-1400. Failure rates were considered for the following circuit components:

1. Open circuit
2. Short to ground
3. Short to power
4. Timing motor failure to start
5. Relay failure to energize
6. Normally open contact failure to close
7. Normally open or normally closed contact short
8. Normally closed contact opening
9. Timing switch failure to transfer

The calculated probabilities (Pf) for no undetected failure of the circuits occurring were as follows, based on the specified test frequency.

AREA	TEST FREQUENCY	Pf
Spreading Room B	One Month	0.975287
HPCI Water Fog	Six Months	0.977175
Standby Diesel Gen Room A CO_2	Six Months	0.957595

The worst case of the three areas considered is Spreading Room B. The probability of undected failure is approximately 1/40, which means that one undetected failure will occur on the average every 40 months over an extended period of time and that the failure could exist up to one month. The frequency of testing is thus much greater than the frequency of failure and produces circuits with adequate reliability.

2. Circuits checks by initiation of end of the line or end of the branch detectors will more thoroughly test the parallel curcuits than testing on a rotating detector basis. This test is not a detector test, but is a test to simulate the effect of electrical supervision as defined in the NFPA code.*

3. Testing of circuits which actuate CO_2 , water, or ventilation systems requires disabling the automatic feature of the fire protection system for the area. A surveillance program which disabled these circuits monthly would significantly reduce the ability of these circuits to provide fire suppression.

*Ref: NFPA Code 72D-9, paragraph 1111, Code 72D-15, paragraph 1312 for definition of Class A systems, and Code 72A-18, Article 240.

The fire protection [illegible] flow and pressure to an individual load listed on Table 3.11.A while maintaining a design raw service water load of 1132 gpm.

4.11 BASES

Periodic testing of both the High Pressure Fire System and the CO_2 Fire Protection System will provide positive indication of their operability. If only one of the pumps supplying the High Pressure Fire System is operable, the pump that is operable will be checked immediately and daily thereafter to demonstrate operability. If the CO_2 Fire Protection System becomes inoperable in the cable spreading room, one 125-pound (or larger) fire extinguishere will be placed at each entrance to the cable spreading room.

Annual testing of automatic valves and control devices is in accordance with NFPA code Vol. II, 1975, section 15, paragraph 6015. More frequent testing would require excessive automatic system inoperability, since there are a large number of automatic valves installed and various portions of the system must be isolated during an extended period of time during this test.

Wet fire header flushing, spray header inspection for blockage, and nozzle inspection for blockage will prevent, detect, and remove buildup of sludge or other material to ensure continued operability. System flushes in conjunction with the semiannual addition of biocide to the Raw Cooling Water System will help prevent the growth of crustaceans which could reduce nozzle discharge.

Semiannual tests of heat and smoke detectors are in accordance with the NFPA code.

With the exception of continuous strip heat detectors panels, all non-class A supervised detector circuits which provide alarm only are hardwired through conduits and/or cable trays from the detector to the main control room alarm panels with no active components between. Non-class A circuits also actuate the HPCI water-fog system, the CO_2 system in the diesel generator buildings, and isolate ventilation in shutdown board rooms. The test frequency and methods specified are justified for the following reasons:

1. An analysis was made of worst-case fire detection circuits at Browns Ferry to determine the probability of no undetected failure of the circuits occurring between system test times as specified in the surveillance requirements. A circuit is defined as the wire connections and components that affect transmission of an alarm signal between the fire detectors and the control room annunciator. Three circuits were analyzed which were representative of an alarm-only circuit, a water-fog circuit, and a CO_2 circuit. The spreading room B smoke detector was selected as the worst-case alarm-only circuit because it had the largest number of wires and connections in a single circuit. The HPCI water-fog circuit was selected for analysis because it is the only water-fog circuit in the area of applicability for technical specifications. The Standby Diesel Generator Room A CO_2

3.11 FIRE PROTECTION SYSTEMS

4.11 FIRE PROTECTION SYSTEMS

checked to be 2664 gpm at 250 feet head	
e. Spray header and nozzle inspection for blockage	Once/year
f. System flush in conjunction with semi-annual addition of biocide to the Raw Cooling Water System	Twice/year
g. Building hydraulic performance verification	Once/3 years
h. Yard loop and cooling tower loop hydraulic performance verification	Once/year

LIMITING CONDITIONS FOR OPERATION

3.11 FIRE PROTECTION SYSTEMS

Applicability:

Applies to the operating status of the high pressure water, and CO_2 fire protection systems for the reactor building, diesel generator buildings, control bay, intake pumping station, cable tunnel to the intake pumping station, and the fixed spray system for cable trays along the south wall of the turbine building, elevation 586.

Objective:

To assure availability of Fire Protection Systems.

Specification:

A. High Pressure Fire Protection System

1. The High Pressure Fire Protection System shall have:

a. Two (2) high pressure fire pumps operable and aligned to the high pressure fire header.

b. Automatic initiation logic operable.

SURVEILLANCE REQUIREMENTS

4.11 FIRE PROTECTION SYSTEMS

Applicability:

Applies to the surveillance requirements of the high pressure water, and CO_2 fire protection systems for the reactor building, diesel generator buildings, control bay, intake pumping station, cable tunnel to the intake pumping station, and the fixed spray system for cable trays along the south wall of the turbine building, elevation 586 when the corresponding limiting conditions for operation are in effect.

Objective:

To verify the operability of the Fire Protection Systems.

Specification:

A. High Pressure Fire Protection System

1. High Pressure Fire Protection System Testing:

Item	Frequency
a. Simulated automatic and manual actuation of high pressure pumps and automatic valve operability	Once/year
b. Pump Operability	Once/month
c. Deleted	
d. Pump capability	Once/3 years

LIMITING CONDITIONS FOR OPERATION

3.6.C Coolant Leakage

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be shutdown in the Cold Condition within 24 hours.

D. Safety and Relief Valves

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

E. Jet Pumps

1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be shutdown in the Cold Condition within 24 hours.

SURVEILLANCE REQUIREMENT

4.6.C Coolant Leakage

D. 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.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the startup or run modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

3.6.E Jet Pumps

3.6.F Jet Pump Flow Mismatch

1. When both recirculation pumps are in steady state operation, the speed of the faster pump shall be maintained within 122% the speed of the slower pump when core power is 80% or more of rated power or 135% the speed of the slower pump when core power is below 80% of rated power.

2. If specification 3.6.F.1 cannot be met, one recirculation pump shall be tripped.

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

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

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

G. Structural Integrity

1. The structural integrity of the primary system shall be

4.6.E Jet Pumps

b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

2. Whenever there is recirculation flow with the reactor in the Startup or Run Mode and one recirculation pump is operating with the equalizer valve closed, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

F. Jet Pump Flow Mismatch

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

G. Structural Integrity

1. Table 4.6.A together with supplementary notes, specifies the

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10% of the control rods at 16-week intervals is adequate for determining the operability of the control rod system yet is not so frequent as to cause excessive wear on the control rod system components.

The numerical values assigned to the predicted scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on Browns Ferry Nuclear Plant.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

In the analytical treatment of the transients, 390 milliseconds are allowed between a neutron sensor reaching the scram point and the start of negative reactivity insertion. This is adequate and conservative when compared to the typically observed time delay of about 270 milliseconds. Approximately 70 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid power supply voltage goes to zero an approximately 200 milliseconds later, control rod motion begins. The 200 milliseconds are included in the allowable scram insertion times specified in Specification 3.3.C.

* In order to perform scram time testing as required by specification 4.3.C.1, the relaxation of certain restraints in the rod sequence control system is required. Individual rod bypass switches may be used as described in specification 4.3.C.1.

The position of any rod bypassed must be known to be in accordance with rod withdrawal sequence. Bypassing of rods in the manner described in specification 4.3.C.1 will allow the subsequent withdrawal of any rod scrammed in the 100 percent to 50 percent rod density groups; however, it will maintain group notch control over all rods in the 50 percent density to preset power level range. In addition, RSCS will prevent movement of rods in the 50 percent density to preset power level range until the scrammed rod has been withdrawn.

3.3/4.4 BASES:

D. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% ΔK. Deviations in core reactivity greater than 1% ΔK are not expected and require thorough evaluation. One percent reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 35 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 32 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 9 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

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

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

1.0 Introduction

The Tennessee Valley Authority (licensee or TVA) has proposed to reload and operate Browns Ferry Unit 1 (B.F.#1) with 168 8x8 (144 8D274L and 24 8D274H) reload fuel assemblies with 80 mil channels. The enrichment of each new 8x8 reload fuel assembly is 2.74 wt. % U-235. The balance of the 596 element core will consist of irradiated 7x7 fuel assemblies previously loaded in the initial core (Cycle 1). All Cycle 2 reload and irradiated assemblies except 7 will have two 9/32-inch holes drilled in each lower tie plate, with the 1-inch bypass flow holes in the core support plate plugged. The 9/32" holes in the fuel assembly lower fuel tie plates permit cooling water to flow into the bypass region between fuel assemblies to cool the in-core nuclear instrumentation and the plugging of 1" bypass flow holes was done to eliminate in-core vibrations.(11)

As noted above, Cycle 2 reload will contain 7 assemblies without the 9/32-inch holes drilled in the lower tie plate. Original B.F.#1 plans were to have all Cycle 2 assembly lower tie plates drilled. However, six of the drilled assemblies were found to be leaking fission products and the other assembly was mechanically damaged. Because of B.F.#1 startup schedular demands, the 7 assemblies were replaced with non-drilled assemblies. B.F.#1 considered this eventuality in their safety analysis, such as their Loss-of-Coolant Accident Analysis and conservatively assumed that 20 assemblies were undrilled.

The reactor is expected to operate in the configuration just described at the licensed power level of 3293 MWt for approximately 12 months. In support of the reload application the licensee has provided the General Electirc (GE) BWR Reload 1 licensing submittal

for B.F.#1[1], proposed Technical Specification changes[2][3][3a], a Loss of Coolant Accident (LOCA) analysis report[3], an increased relief valve simmer margin evaluation[3a], and responses to our requests for additional information.[4]

The information presented in the licensing submittal closely follows the guidelines in Appendix A of the generic GE Topical Report NEDO-20360[5]. Although later supplements to this report are undergoing review by the NRC staff, portions of this topical have been found applicable for reactors containing 8x8 reload fuel and are acceptable to us when supplemented with information required by our status report[6]. The supplemental information provided by the licensee and our evaluation thereof are summarized in Section 2.0 of this Safety Evaluation Report (SER).

In addition to the changes being made to the Technical Specifications that are related to the loading of 8x8 assemblies into Unit 1 for Cycle 2 operation, there are certain changes being made to the Technical Specifications of all three Units. These changes involve: (1) a request to clarify the operability requirements of the Rod Worth Minimizer and the Rod Sequence Control System during scram time testing submitted by application dated January 12, 1977, (2) a request to add standards for qualifications of the Health Physics Supervisor submitted by application dated May 11, 1977, (3) a request to change and add certain fire protection Technical Specifications submitted by application dated September 23, 1977, (4) a request to delete annual operating report requirements and change the monthly reporting requirements submitted by application dated November 16, 1977, and (5) a request to substitute revised, but equivalent, terms in the equations for the limiting settings on the Average Power Range Monitors' scram and rod block setpoints submitted by application dated December 13, 1977. Our evaluation of these changes to the Technical Specifications are summarized in Section 3.0 of this SER.

2.0 Evaluation of B.F.#1 Reload For Cycle 2

2.1 Nuclear Characteristics

For Cycle 2 approximately 22% of the 764 fuel assemblies will be unirradiated, and 78% will have been irradiated for one cycle. As indicated by the loading diagram presented in Reference 1, these assemblies will be distributed such that the core is quarter core symmetrical.

The data in Reference 1 indicate that the nuclear characteristics of the Reload 1 core are within the envelope of those values used in the analysis of the previous core. The licensee therefore states that the total control system worth, temperature, and void dependent behavior of the reconstituted core will not differ significantly from those values previously reported for B.F.#1 The shutdown margin of the Cycle 2 core meets the Technical Specification requirement that the core be at least 0.38% Δk subcritical in the most reactive condition throughout the operating cycle with the most reactive rod fully withdrawn and with all the others fully inserted. For Cycle 2 the minimum shutdown margin has been calculated by the licensee to be 0.019 Δk and occurs at the beginning of cycle.

The information presented by the licensee in Reference 1 indicates that a boron concentration of 600 ppm in the moderator will bring the reactor subcritical by at least 0.03 Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria is met by the Standby Liquid Control System.

The Technical Specification requirement for the storage of fuel for B.F.#1 is that the effective multiplication factor of the fuel, for dry conditions, is less than 0.90 and flooded is less than 0.95. This is achieved if the uncontrolled k_∞ of a single fuel bundle is less than 1.30 at 65°C. The peak uncontrolled k_∞ of 8D274L and 8D274H have a maximum k_∞ of 1.238 and 1.216 respectively within the applicable exposure and temperature range. These are less than 1.30 so that storage requirements for B.F.#1 are met.

Based on a review of the information presented in the B.F.#1 licensing submittal[1] as supplemented by applicable portions of the generic 8x8 reload report[5] and our acceptance thereof[6], we have determined that the nuclear characteristics and performance of the Cycle 2 core are similar to those of Cycle 1 and are acceptable.

2.2 Mechanical Design

The reload fuel has the same mechanical configuration and fuel bundle enrichments as the 8D247L and 8D274H assemblies described in the generic 8x8 reload Topical Report (Reference 5) except that two 9/32 inch holes are drilled in the lower tie plate of each reload assembly to provide bypass flow. Also, the improved water rod design described in Section 3.1 of Reference 5 has been adopted.

The generic 8x8 reload Topical Report [5], supplements of which are under review, has been found acceptable for use for reactors containing 8x8 reload fuel, when supplemented with information required by our status report (Reference 6) on the GE generic report evaluation. On the basis of our review of the generic 8x8 reload Topical Report and the reload submittal we conclude that the mechanical design of the B.F.#1 Reload 1 is acceptable.

2.3 Thermal-Hydraulics

The generic 8x8 reload Topical Report[5] and GETAB[7] are referenced to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins. Application of the GETAB establishes:

(1) the fuel damage safety limit,

(2) the limiting conditions of operation (LCO) such that the safety limit is not exceeded for normal operation and anticipated transients, and

(3) the limiting conditions of operation such that the initial conditions assumed in the accident analyses are satisfied.

We have evaluated the B.F.#1 Cycle 2 thermal margins based on the GETAB report[7] and plant specific input information provided by the licensee. Our evaluation of these margins is reported herein.

2.3.1 Fuel Cladding Integrity Safety Limit - Minimum Critical Power Ratio (MCPR)

The fuel cladding safety limit MCPR has been increased from 1.05 to 1.06, based on the GETAB[7] statistical analysis, to assure that 99.9% of the fuel rods in the core will not experience boiling transition during abnormal operational transients[8]. This limit is applied for both core-wide and localized transients or perturbations to the expected Critical Power Ratio (CPR) distribution.

The uncertainties in core and system operating parameters and the GEXL correlation uncertainties expected for Cycle 2 operation of B.F.#1 are the same as those used for the original statistical analysis (Table 4-2 of Reference 5) on which the fuel cladding safety limit MCPR is based except for those increased changes due to a reload core. For example the standard deviation for the TIP readings uncertainty for the Cycle 2 core is 8.7% whereas the GETAB NEDO-10958 report shows 6.3%. The increase in uncertainty for the Cycle 2 core is a consequence of the increase in uncertainty in the measurement of power in a reload core. A TIP uncertainty of 6.3% would be applicable if this were the initial core. In both cases the TIP reading uncertainties are based on a symmetrical planar power distribution.

The bundle power distribution for Cycle 2 is expected to include fewer high power bundles than the distribution assumed for the original statistical analysis as is indicated by comparing Figures 4-1 and 4-2 in Reference 1 with Figure 4-2 of Reference 5. Therefore, it is conservative to apply the fuel cladding safety limit MCPR of 1.06 to Cycle 2 operation of B.F.#1.

2.3.2 Operating Limit MCPR

Various transients or perturbations to the CPR distribution could reduce the MCPR below the intended operating limit during Cycle 2 operation of B.F.#1. The limiting operational transients were analyzed by the licensee to determine which could potentially induce the largest reduction in MCPR.

The limiting operational transients evaluated were load rejection with failure of the bypass valves, turbine trip with failure of the bypass valves, loss of a 100°F feedwater heater, feedwater controller failure, and the control rod withdrawal error. Initial conditions and transient input parameters as specified in Table 4-3, Table 6-1 and Figure 6-1 of Reference 1 were assumed. For most of the parameters which vary with exposure, the limiting and most conservative value that would occur during the cycle were assumed. The exceptions to this are the local peaking factor and GEXL R-factor which are conservatively assumed to be those of fresh fuel.

We have reviewed the input to the transient calculations and the application of the analysis methods of Reference 5 and have determined that they provide appropriate conservatism for determination of the operating limit MCPR for B.F.#1 during Cycle 2.

The calculated reductions in CPR during each of the operational transients have been identified by the licensee in Reference 3a. The most limiting operational transients occurring at any time during Cycle 2 from rated conditions in the categories shown in Table 4-2 are: (1) a rod withdrawal error for the 7x7 fuel from BOC-2 to 3440 MWD/$_t$ with a ΔCPR of 0.24, (2) load rejection without bypass for 8x8 fuel from BOC-2 to 3440 MWD/$_t$ with a ΔCPR of 0.26 and (3) load rejection without bypass for 7x7 and 8x8 fuel from 3440 MWD/$_t$ to EOC with a ΔCPR of 0.28 and 0.38, respectively.*

Addition of these ΔCPR's to the safety limit MCPR would normally provide the minimum operating limit MCPR for each fuel type required to avoid violation of this safety limit, should these limiting transients occur. The licensee has therefore proposed MCPR operating limits of 1.30 and 1.32 for the 7x7 and 8x8 fuel types respectively from BOC-2 to 3440 MWD/$_t$ and 1.34 and 1.44 for the 7x7 and 8x8 fuel types respectively from 3440 MWD/$_t$ to EOC-2. However, the licensee reports in the reload submittal(1) that the most severe fuel loading error, consisting of a fresh 8x8 bundle loaded in a core position analyzed for a high burnup 7x7 assembly, results in a ΔCPR of 0.25 which exceeds the ΔCPR associated with the most limiting abnormal operational transient for 7x7 fuel from BOC-2 to 2440 MWD/$_t$. This fuel loading error could, therefore, decrease the MCPR below the safety limit MCPR (i.e., to 1.05) if the operating limit were based soley on the consideration of anticipated operational transients.

The staff has the fuel loading error under generic review. Until this issue is resolved, the staff, in the interim, requries that the operating limit MCPR proposed by the licensee be increased an additional .01 for 7x7 fuel from BOC-2 to 3440 MWD/$_t$ to account for the possibility of a fuel loading error.

Thus, based on the analyses of both the most severe abnormal operational transients add the fuel loading error, we require that the operating limit MCPR be 1.31 for 7x7 fuel from BOC-2 to 3440 MWD/$_t$ to avoid violating the safety limit in the event of a fuel loading error from rated conditions. The licensee has agreed to increase the operating limit MCPR to this value.

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* BOC- Beginning of Cycle
EOC- End of Cycle

2.3.3 Operating MCPR Limits For Less Than Rated Power And Flow

For the limiting transient of recirculation pump speed control failure at lower than rated power and flow conditions, the licensee will conform to the limiting conditions for operation stated in the Technical Specifications. This requires that for core flows less than the rated flow, the licensee maintain the MCPR greater than the minimum operating values. The minimum operating MCPR values for less than rated flow are the MCPR's for full rated flow (1.31 and 1.32 for the 7x7 and 8x8 fuel types respectively from BOC-2 to 3440 $MWD/_t$ and 1.34 and 1.44 for the 7x7 and 8x8 fuel types respectively from 3440 $MWD/^t$ to EOC-2), multiplied by the respective K_f factors appearing in Figure 3.5-2 of the Technical Specifications. The k_f factor curves were generically derived and assure that the most limiting transient occurring at less than rated flow will not exceed the safety limit MCPR of 1.06. We conclude that the calculated consequences of the anticipated operational transients do not violate the thermal limits of the fuel or the pressure limits of the reactor coolant boundary.

2.4 Accident Analysis

2.4.1 Fuel Loading Error

Fuel loading errors are discussed in Reference 2 for a fuel bundle placed in an improper location or rotated 180 degrees. For B.F.#1 the worst potential fuel loading error for Cycle 2 would result in a MCPR no less than 1.06 for an operating limit MCPR of 1.31 and a peak linear heat generation rate of 16.5 Kw/ft(1). The implications of the MCPR have been discussed previously and the peak LHGR is not large enough to cause fuel damage.

2.4.2 Control Rod Drop Accident

In Figures 6-1 through 6-3 of Reference 1 the licensee has shown that during Cycle 2 operation of B.F.#1 the magnitude of the Doppler coefficient as a function of fuel temperature and the magnitude of the reactivity insertion due to a dropped in-sequence control rod versus rod position are smaller than bounding curves of these quantities presented in Reference 5. Since the scram reactivity function for 20°C is outside of the bounding analysis, a specific analysis was performed by the licensee to verify that the consequences of a rod drop excursion from any in-sequence control rod would be below the design limit. The resultant peak enthalpy from the specific analysis is 161 cal/g for the 20°C case. The results of this analysis and the results of the scram reactivity function at 286°C for B.F.#1 being within the bound of the analysis for the generic reload are sufficient justification that no in-sequence rod drop accident will lead to peak fuel enthalpies greater than the 280 cal/gm design basis.

2.4.3 Fuel Handling Accident

The fuel handling accident was addressed in the original SER (6/26/72) prior to issuance of the operating license and in the staff's review of the generic 8x8 reload Topical Report. In the review of the generic 8x8 reload Topical Report, we stated the mechanical analysis should be better justified. However, our conclusion that the amount of fission products released from 8x8 fuel assemblies in a refueling accident would not be significantly greater than from the 7x7 fuel assemblies is not changed by this reload, and the conclusions of the SER (6/26/72) that the dose consequence of a fuel handling accident would be well within 10 CFR 100 guidelines are not changed.

2.4.4 ECCS Appendix K Analysis

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

In December of 1976, we were informed that certain input errors and computer code errors had been made in the evaluations that were provided under the requirements described above. An Order was issued to TVA on March 11, 1977, requiring that corrected revised calculations fully conforming to the requirements of 10 CFR 50.46 be provided for the Browns Ferry Nuclear Plant Unit 1 facility as soon as possible. Such corrected analyses were provided for the present reload in Reference 3. The corrected analyses included correction of all input errors previously made and correction of all computer code errors. The corrected analyses were performed using a calculational model which contains several model changes approved by the NRC staff in a Safety Evaluation issued April 12, 1977.[13] This Safety Evaluation is applicable to B.F.#1 and is incorporated by reference herein.

We have reviewed the corrected analyses submitted for the reload in Reference 3 along with a supplemental evaluation submitted in Reference 3a. We conclude that the B.F.#1 will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50 when: (1) it is operated in accordance with the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values given in Tables 3.5.I-1, -2, -3 and -4 of Reference 3a and (2) when it is operated at a MCPR equal to or greater than 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the Loss-of-Coolant Accident, as described elsewhere in this SER).

The analyses submitted in Reference 3 provide all information requested in our letter to GE on June 30, 1977, regarding number of breaks to be analyzed, documentation to be provided, etc. for the new analyses. These analyses for B.F.#1 reference the lead plant (James A. Fitzpatrick Nuclear Power Plant) analyses for BWR/4 plants with the low-pressure-coolant-injection system modification.

The staff's Safety Evaluation for Fitzpatrick is also lead plant evaluation and is incorporated by reference herein. This B.F.#1 ECCS evaluation considers only matters which differ from Fitzpatrick. The following description is proivded of particular features of the analyses which are different from the lead plant analyses and the reason underlying those differences. The break spectrum (i.e., peak clad temperature [PCT] vs. break size) for the lead plant[15] showed that the particular break producing the highest PCT for the lead plant was a recirculation pump discharge line break having an area approximately 80% as large as the largest discharge line break. However, the break spectrum for B.F.#1 showed that the particular break producing the highest PCT is the largest (100%) suction line break.

The SER for the lead plant[15] explains the reasons why the discharge break location is limiting for that plant. As explained more fully in that SER[15], the largest break in the largest pipe would normally be expected to be limiting (the largest pipe is the suction pipe). However, the LPCI modification (also explained more fully in the lead plant SER[15]) results in at least one loop of the LPCI system being available to help mitigate the consequences of suction pipe breaks even with the worst assumed single failure; but, due to certain piping and valve locations, with certain single failure assumptions, no LPCI system is available for the smaller, discharge line break. This results in a tradeoff or compensating effects situation where a larger, normally more severe break (suction line) has more ECCS available to mitigate its consequences, while a smaller, normally less severe break (discharge line) has less ECCS. The lead plant SER[15] states that in most cases this tradeoff results in the discharge break being limiting, as it is for Fitzpatrick.

For B.F.#1, the tradeoff had a different result with the largest suction break being slightly (23°F) more limiting than the worst discharge break analyzed. The reason for the difference between Fitzpatrick and B.F.#1 analysis results is best explained by the concept of an "effective break size," which is defined as the ratio of break area to primary system volume. The higher the "effective break size," the more severe are the consequences of the break (i.e., blowdown will be faster, flow decay and departure from nucleate boiling will be sooner, and core uncovery will be sooner, all of which contribute to higher PCT). Compared to Fitzpatrick, B.F.#1 has a smaller discharge line and a larger primary system volume, both of which combine to make the "effective" discharge break much smaller for B.F.#1 than for Fitzpatrick. On the other hand, the suction lines on the two plants are approximately the same size, and although the larger primary system volume of B.F.#1 makes the B.F.#1 "effective" suction line break somewhat smaller than Fitzpatrick's, the decrease is not as pronounced as for the discharge line break. Therefore , when one compares the break spectrum of the two plants, one would expect to see the discharge break relatively less severe (compared to the suction break) on B.F.#1. This shift is just large enough to cause the suction break to become limiting on B.F.#1.

In order to justify the above argument that the largest suction line break is limiting, it is necessary to determine that no discharge or suction break size that was not specifically analyzed could be more limiting than the discrete sizes that were specifically analyzed.

The same arguments presented in the lead plant SER(15) regarding PCT vs. discharge line break size also apply to B.F.#1. For B.F.#1 the maximum uncovered time interval peaks at 66% of the largest discharge break area. Since the uncovered time is a maximum, the highest PCT for a discharge line break, will be at or near that break size*. For the suction line break, the longest uncovered time interval occurs for a break equal to 100% of the largest suction line area, and since all other significant effects also tend to make the largest break limiting (i.e., earliest loss of nucleate boiling and uncovery time), it is clear that the "100%" suction line break is the most limiting suction line break.

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*Slight differences in "effective break size" and plant geometry (i.e., bypass area, bypass flow holes, etc.) caused this peak to occur at 80% of the largest discharge break area for Fitzpatrick, but the same arguments used in the Fitzpatrick SER apply to explain why the maximum PCT does not

TVA has presented results of PCT calculations specifically for B.F.#1 for the largest suction line break, largest discharge line break, and most limiting discharge line break. We agree, for the reasons stated above, that the most limiting break is the largest suction line break. This was used to generate the referenced MAPLHGR limits, which we therefore find acceptable as stated previously.

2.4.5 Steam Line Break Accident

Steam line break accidents which are postulated to occur inside containment are covered by the ECCS analysis discussed in section 2.4.4. The analysis of steam line break accidents occurring outside containment as presented by the licensee is acceptable based on our review and acceptance of the generic report NEDO-20360.[5,6]

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occur for the largest discharge line break for B.F.#1. The question arises on Fitzpatrick and on B.F.#1 as to whether or not the maximum discharge break PCT occurs precisely at the "80%" and "66%" discharge line break size respectively, for the two plants (i.e., has the worst break been found and analyzed). Since the "80%" break on Fitzpatrick was the most limiting break for that plant (with PCT = 2200°F) additional analyses were performed at slightly larger and slightly smaller breaks to more precisely locate the worst break size. In addition an added conservatism was included in the analyzed breaks to more precisely locate the worst break size and a shorter DNB time was assumed to add more conservatism into the calculation which would more than compensate for any slight error in precisely determining the exact size of the limiting break[15]. In the case of B.F.#1, these additional analyses and conservatisms were not included, since it is only necessary to show that no unanalysed discharge break could be more limiting than the worst (limiting) suction line break. The uncovered time period versus break area peaks very sharply at "66%", that is, any change to a slightly larger or smaller break area would cause a shift to a significantly shorter uncovered time which would over-compensate for any effects in the other directions due to the size change and result in a lower PCT. Moreover, if the highest PCT discharge line break size is slightly different from 66%, the 66% discharge break PCT is 2128°F, which is 23°F below the limiting (largest) suction line break's PCT of 2151°F. Any small inaccuracies in precisely determining the worst discharge break size couldnot cause more than a 2°F to 5°F shift in PCT, and the worst discharge break's PCT would still not become limiting (i.e., higher than 2151°F).

2.5 Overpressure Analysis

The licensee has presented analyses (one for the BOC-2 to 3440 MWD/t and one for 3440 MWD/t to EOC-2) to demonstrate that during the most severe overpressure event an adequate margin (99 psi and 81 psi respectively) exists between the peak vessel pressure and the ASME Code allowable vessel pressure which is 110% of the vessel design pressure[3a]. The analysed event, which produced the most severe overpressure, was the closure of all main steam line isolation valves (MSIV) with high flux scram and recirculation drive (pump) motor trip (ATWS DMT). ATWS DMT is trip of the recirculation pump on a high pressure signal. The input to the calculation is listed in Table 6-1 of Reference 1, and included end of cycle scram characteristics, void coefficient and Doppler coefficients. Furthermore, it has been demonstrated that should the MSIV transient be initiated at a reactor power slightly above the value assumed for the analysis (because of uncertainties in monitoring of power) there would still be an adequate margin to the ASME code pressure limit[4]. Similarly, should the transient be initiated at the maximum dome pressure allowed by the Technical Specifications rather than that assumed for the analysis there would be adequate margin to the pressure limit[4].

The effect on peak vessel pressure during an MSIV closure from the failure of a safety valve has been evaluated to be approximately 20 psi[1,9] so that the margin to the code limit is adequate for this circumstance also.

Based on the analysis and sensitivity studies submitted by the licensee the overpressure analysis for B.F.#1 for Cycle 2 has been found acceptable.

2.6 Thermal Hydraulic Stability Analyses

The thermal hydraulic stability analyses and results are described in References 5 and 1. The results of the Cycle 2 analyses show that the 7x7 and 8x8 channel hydrodynamic stability, at either rated power and flow conditions or at the low end of the flow control range, is within the operational design guide in terms of decay ratio.

Calculations were also performed by the licensee to assess the reactor power dynamic response at the two aforementioned reactor operating conditions. The results showed that the reactor core decay ratios at both conditions are well within the operational design guide decay ratio. We find these results to be acceptable.

We have expressed generic concerns regarding the least stable reactor condition allowed by Technical Specifications. This condition could be reached during an operational transient from high power where the plant sustains a trip of both recirculation pumps. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as fuel designs improve. Our concerns relate to both the consequences of operating at the ultimate decay ratio for the equilibrium core and the capacity of analytical methods to accurately predict decay ratios. The General Electric Company is addressing these concerns through meetings, Topical Reports and a test program.

Until this issue has been resolved generically, we have imposed a requirement on B.F.#1 which will restrict planned operations in the natural circulation flow mode[4]. The licensee has agreed to this Technical Specification limitation. The restriction will provide a significant increase in the reactor core stability margins during Cycle 2. On the basis of the foregoing, we find the thermal-hydraulic stability of B.F.#1 to be acceptable.

2.7 Recirculation Pump Startup From The Natural Circulation Operational Mode

During a recent BWR reload review [10] we raised a concern about recirculation pump startup from the natural circulation operational mode. Such pump startup could increase flow, collapse moderator voids, and subsequently result in a reactivity insertion transient. We note that the licensee identified[4] an analysis[16] made for a startup of an idle recirculation loop at power and flow conditions near natural circulation. However, the reported analysis does not adequately address our question on this matter and is still under review. Therefore, authorization to operate in this fashion would require additional analyses as to this accident sequence and its consequences. In the absence of this information, the licensee has agreed to have the Technical Specifications amended to restrict power operations in the natural circulation mode to reduce the potential for such an accident and to not allow startup of a recirculation pump from the natural circulation condition unless the temperature of the recirculation loop is within 75°F of the primary coolant water in the reactor vessel. We find these restrictive measures reduce the probability and consequences of this operation to an acceptably low level.

2.8 Physics Startup Testing

The licensee will conduct physics startup tests which, in addition to verifying the predicted shutdown margin, will test the incore monitoring instrumentation, the process computer programming and input, and the core loading. These tests will provide additional assurance that the B.F.#1 Cycle 2 core is loaded consistently with the reload licensing submittal, and that the uncertainties in monitoring power distributions are sufficiently small that the design basis safety limit MCPR of 1.06 is applicable.

Because the Cycle 2 core is to have a quarter core mirror symmetric loading there will be differences between the exposure environments of the pairs of diagionally symmetric TIPs on which the TIP symmetry tests are to be made. These exposure differences are expected to produce a larger apparent TIP uncertainty than would result from geometrical and random noise effects alone. Because the criteria on the maximum uncertainty allowed before taking corrective action are based on geometrical and noise uncertainties only, the TIP symmetry test for B.F.#1 is expected to conservatively overestimate the conditions under which actions are required. The results of the tests will be available within 90 days of startup.

2.9 Rubber Shoe Cover Lost In Reactor Vessel

A rubber show cover fell into the Unit 1 vessel during the refueling outage for Cycle 2 reload. Extensive search activities were conducted by TVA over a three week period without success in finding the shoe cover. TVA had the General Electric Company run tests on identical shoe covers. These tests included heat-up in a water autoclave to greater than 500°F and flow tests with flows up to 50 percent of rated reactor core flow in a test flow loop that simulated the core entry flow path configuration.

We have reviewed the material submitted by the Tennessee Valley Authority (TVA) regarding the lost rubber shoe cover in the reactor vessel.[22, 23] Chemical effects, possible control rod interference, and potential flow blockage to a fuel assembly are the three areas of potential concern; these three subjects were addressed by TVA and are discussed below in that order.

The very small amount of material introduced by decomposition of the shoe represents an insignificant fraction of the total primary system inventory. The lack of florides and the insignificant amount of chlorides (1 to 2 grams) indicate that the material would have no significant effect on water chemistry or corrosion in the primary system.

The shoe cover could potentially lodge in a control blade guide tube, causing increased friction which would be detected during control blade motion tests. However, based on our knowledge of the large forces available to insert a control blade during a scram, and considering the relatively low strength of a rubber shoe cover (even a rubber shoe cover before high temperature weakens it as described below), we concur with the GE-TVA conclusion that the shoe cover could not significantly affect a reactor scram.

The potential for flow blockage to a fuel assembly required that certain procedures be followed as described below to disintegrate the shoe cover before reactor operation at powers where flow blockage could pose a safety hazard .

Autoclave tests have been conducted which demonstrate that this type of rubber shoe will lose tensile strength and structural integrity after exposure to 500°F water for more than 24 hours.[22, 23] Such autoclaved material has been tested in a flow loop at lower temperatures (less than 200°F) and was shown to rapidly disintegrate when flows approach 100 gpm, the equivalent of 50% of rated flow in the reactor. At flows in the range of 60 gpm, the equivalent of 30% of rated flow in the reactor, the autoclaved material was shown to break apart but at a much slower rate (the pressure drop across the "rubber blockage plane" decreased by approximately a factor of 2 in about 12 minutes at the equivalent of 30% flow.)[24]

Under startup conditions proposed by TVA and described below, the flow induced disintegration would occur at greater than 30% of rated flow and at temperatures above 500°F, not at the less-than-200°F conditions present in the test loop. Based on our own manipulation of autoclaved rubber samples at room temperature and at 212°F (under boiling water) we know that this material becomes much weaker as temperature is increased. NRC staff personnel who are familiar with physical properties and behavior of rubber, the TVA staff, and the Goodyear Tire and Rubber Company technical staff agree that this same trend would continue to higher temperature; i.e., that above 500°F the rubber would have less tensile strength and would disintegrate faster than at less-than-200°F.

Therefore we concur with the TVA staff that the rubber shoe would reach the weakened (autoclaved) condition and would subsequently disintegrate into pieces so tiny that they could not cause flow blockages having any safety significance after exposure to in-reactor temperatures above 500°F and flows in excess of 30%-of-rated flow for 60 hours.

During reactor startup, TVA proposes to expose the shoe to the above conditions (60 hours at 500°F-or-above temperature and 30%-of-rated or greater flow) before core power is allowed to exceed 5%. We concur that operating under these conditions for 60 hours poses no safety hazard for the following reasons. Flow reduction to less than 70% of the flow in an unblocked assembly could not be experimentally produced even by optimally placing the rubber material by hand to cause such blockage in the flow loop. Even if complete blockage of the inlet could nevertheless somehow be produced in the reactor, sufficient flow would enter the bundle through the "finger spring" path alone (other "leakage" paths also exist) to prevent departure-from-nucleate boiling from occurring at bundle powers below 0.6 MW.[22, 25] This corresponds to a core power below 5%, based on a study of worst power peaking that could occur during startup with the Browns Ferry Unit 1 rod-withdrawal sequence. Therefore, reactor operation below 5% power, until shoe cover disintegration occurs, poses no safety problem due to potential blockage from the shoe.

Following startup operation as above, TVA will increase power to allow feedwater pump operation so that inlet subcooling can be provided to the recirculation pumps. The pumps can then be run at 100% of rated flow, which will be maintained for at least 1 hour before core power is allowed to exceed 30%. This will assure removal of any remaining small amount of flow blockage (that somehow might unexpectedly survive the preceeding lower flows) before full core powers are reached.

Based on the above, we concur with TVA that full power operation of Browns Ferry Unit 1 following the startup procedures described will not pose a hazard to safe operation.

2.10 Technical Specification Changes For B.F.#1 Cycle 2

- The proposed Technical Specification changes(1), incorporate the Fuel Cladding Integrity Safety Limit MCPR and Operating Limit MCPR requirements for 7x7 and 8x8. The basis for these changes are addressed in Sections 2.3.1, 2.3.2 and 2.3.3.

- The licensee has proposed to incorporate fuel densification power spiking effects on the maximum LHGR equation for the reload 8x8 fuel. Until such time as removal of this penalty is approved generically, NRC is continuing to require a 2.2% penalty.

- The licensee has proposed changes to the Technical Specifications, to preclude or limit operation with natural circulation flow in the STARTUP and RUN modes of operation. The basis for this change is addressed in Section 2.7.

- The licensee has proposed new MAPLHGR values for Reload 1 fuel. The basis for this change is addressed in Section 2.4.4.

- The licensee has proposed 67B scram times in the Technical Specification. This change reduces the 90% insertion time. Changes in insertion time affect the most limiting operational transients. For these transients the first two seconds are critical. The Technical Specification for 50% insertion time is two seconds and since the 50% insertion time is not being changed the proposed 67B scram times has little or no effect on these transients.

- The licensee has proposed to add 13.4 KW/ft as the design LHGR for 8x8 fuel. The design LHGR was generically reviewed as part of Reference 5 and found to be acceptable by the NRC staff.

- The licensee has proposed startup limitations on power level and recirculation flow rates for certain time durations in order to ensure that a shoe cover lost in the reactor vessel is disintegrated. The basis for this change is addressed in Section 2.9.

We find the Technical Specification changes acceptable and consistent with the information in the B.F. Reload #1 licensing submittal.

3.0 Evaluation of Other Technical Specification Changes

3.1 Rod Worth Minimizer (RWM) and Rod Sequence Control System (RSCS)

TVA requested a change to the Technical Specifications for Units 1, 2 and 3 that would clarify the operability requirements of the RWM and the RSCS.(17) This change relates to a surveillance requirement of the Technical Specifications to test the insertion time for all operable control rods after each refueling outage. This testing is necessary to ensure that the control rods will insert within the time used for the transient analyses which demonstrate that the core safety limits will not be violated during those transients. In order to test some of the rods, the restraints imposed by the RSCS must be by-passed. The Standard Technical Specifications being issued for plants presently being licensed include such an allowance for by-pass. The RWM also has an allowance for inoperability below 20 percent power provided that a second operator verifies that the operator at the reactor console is following the control rod program.

The change proposed by TVA would include a restriction that prohibits the use of the second operator in lieu of the RWM during the scram time testing. The change also requires that the actual axial position of a bypassed rod must be known and the rod must be in the correct in-sequence position. These changes provide the proper commensurate requirements for rod movement control and we find the changes acceptable.

3.2 Health Physics Supervisor

TVA requested a change to the Technical Specifications for Units 1, 2 and 3 relating to the qualifications of the Health Physics Supervisor.(18) We had requested by letter dated March 9, 1977, that the Technical Specifications be modified to make it clear that the Health Physics Supervisor must meet the requirements set forth in Regulatory Guide 1.8, "Personnel Selection and Training" dated September 1975. This change clarifies the personnel qualification requirements in this respect, satisfies our request and is therefore acceptable.

3.3 Fire Protection Technical Specifications

TVA requested a change to the Technical Specifications for Units 1, 2 and 3 to modify the fire protection specifications.(19) We have not completed our review of all of the proposed changes. However, one change that they proposed would change the frequency of testing automatic valves and control devices from quarterly to annually. Annual testing of automatic valves and control devices is in accordance with NFPA Code Volume II, 1975, Section 15, paragraph 6015. More frequent testing would require more automatic system inoperability, since there are a large number of automatic valves installed and certain portions of the system must be isolated in order to perform the testing. The present Standard Technical Specifications for new plants require annual testing. Based on the foregoing, we find the proposed annual testing acceptable.

3.4 Annual Operating Report

Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications," is the basis for reporting requirements found in Technical Specifications today. When these Technical Specifications were issued we requested that licensees use the formats in the guide for the Licensee Event Report (LER) and Monthly Operating Report. In some cases licensees' use of these formats was required by a reference to Regulatory Guide 1.16 in the Technical Specifications. After two years of experience with the reporting requirements identified in this guide we reviewed the scope of information licensees are required to submit in the LER, Annual Operating Report, Monthly Operating Report and Startup Report.

From our review of all licensee reports, we determined that much of the information found in the Annual Operating Report either is addressed in the LER's or Monthly Operating Report, which are submitted in a more timely manner, or could be included in these reports with only a slight augmentation of the information already supplied. Therefore we conclude that the Annual Operating Report could be deleted as a Technical Specification requirement if certain additional information were provided in the Monthly Operating Reports. As a result we sent letters during September 1977 to licensees informing them that a revised and improved format for Monthly Operating Reports was available and requested that they use it. Licensees were informed that if they agreed to use the revised format they should submit a change request to delete the requirement for an Annual Operating Report except that occupational exposure data must still be submitted.

By letter dated November 16, 1977, TVA requested a change to the Technical Specifications that would delete all but one of the four specified items in the Annual Operating Report. The report which tabulates occupational exposure on an annual basis is needed and therefore, the requirement to submit this information has been retained. We have determined that the failed fuel examination information does not need to be supplied routinely by licensees because this type of historical data can be obtained in a compiled form from fuel vendors when needed. The information concerning forced reductions in power and outages will be supplied in the revised Monthly Operating Reports and the narrative summary of operating experience will be provided on a monthly basis in the Monthly Operating Report rather than annually. The licensee has committed to use the revised Monthly Operating Report format beginning with their report for January 1978 as requested. We have concluded that all needed information will be provided and deletion of the Annual Operating Report is acceptable.

3.5 Core Maximum Fraction of Limiting Power Density (CMFLPD)

TVA proposed a change to the Technical Specifications for Units 1, 2 and 3 relating to the formula for the limiting settings on the Average Power Range Monitor's scram and rod block setpoints.(21) The change involves substituting an equivalent expression $\left(\frac{FRP}{CMFLPD}\right)$ for the existing expression $\left(\frac{DTPF}{MTPF}\right)$ in the formula, where:

FRP is the fraction of rated power
CMFLPD is the core maximum fraction of limiting power density
DTPF is the design value of the total peaking factor
MTPF is the existing maximum total peaking factor

Since Cycle 2 of Unit 1 includes both 7x7 and 8x8 fuel assemblies which have different design values of the total peaking factor, two formulas would be required for each setpoint with the more limiting result being applicable. The CMFLPD is the highest ratio, for all fuel types in the core, of the maximum fuel rod power density (Kw/ft) for a given fuel type to the limiting fuel rod power density (Kw/ft) for that fuel type. Therefore, a single formula with a unique solution is obtained. In addition, the process computer program for the Browns Ferry Plant already computes the CMFLPD and properly normalizes to the appropriate fuel type. We, therefore, find this change acceptable.

4.0 Conclusions

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.

4.2 Safety Considerations

For those matters discussed in Sections 2.3.1, 2.3.2, 2.3.3, and 2.4.4, we have concluded , based on the considerations discussed in those sections that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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.

For the remainder of the matters evaluated in the other Sections of this SER and their associated changes to the Technical Specifications, we have concluded that: (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: January 10, 1978

References

1. General Electric Boiling Water Reactor Reload No. 1 Licensing Amendment Browns Ferry Nuclear Plant Unit 1, NEDO-24020, May 1977.

2. Tennessee Valley Authority Application for Amendment of Facility Operating License No. DPR-33, TVA-BFNP-TS-86 dated July 8, 1977.

3. Browns Ferry Nuclear Plant Unit 1 Loss-of-Coolant Accident Analysis, NEDO-24056, TVA-BFNP-TS-94 dated September 26, 1977, and Supplement dated October 28, 1977.

3a. Browns Ferry Nuclear Unit 1 Increased Relief Valve Simmer Margin Evaluation, TVA-BSNP-TS-95 dated September 27, 1977.

4. Additional Information on Reload 1 for Browns Ferry Unit 1, October 25, 1977, November 7, 1977 and November 28, 1977.

5. "General Electric Reload Licensing Application for 8x8 Fuel," Revision 1, Supplement 3, September 1975, NEDO-20360.

6. Status Report on the Licensing Topical Report, "General Electric Boiling Water Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, United States Nuclear Regulatory Commission, April 1975.

7. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," NEDO-10958, Class 1, November 1973.

8. General Electric Letter (John A. Hinds) to U.S. Atomic Energy Commission (Walter Butler), "Responses to the Third Set of AEC Questions on the General Electric Licensing Topical Reports," NEDO-10958 and NEDE-10958, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," July 24, 1974.

9. General Electric Letter (Ivan F. Stuart) to U.S. Nuclear Regulatory Commission (Victor Stello) "Code Overpressure Protection Analysis - December 23, 1975.

10. "Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 33 to License No. DPR-49, Duane Arnold Energy Center (Docket No. 50-331)", dated May 6, 1977.

11. "Safety Evaluation Report on the Reactor Modification to Eliminate Significant In-Core Vibration in Operating Reactors with 1-Inch Bypass Holes in the Core Support Plant," by Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, February 1976.

12. Letter, Dennis L. Ziemann (NRC) to G. Carl Andognini (BECO), "Re: Pilgrim Station Unit 1, " dated March 11, 1977.

13. "Safety Evaluation for General Electric ECCS Evaluation Model Modifications," letter from K. R. Goller, (NRC) to G. G. Sherwood (GE), dated April 12, 1977.

14. Letter, Darrell G. Eisenhut (NRC) to E. D. Fuller (GE), Documentation of the Reanalysis Results for the Loss-of-Coolant Accident (LOCA) of Lead and Non-Lead Plants," June 30, 1977.

15. "Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 30 to Facility Operating License No. DPR-59, Power Authority of the State of New York, James A. Fitzpatrick Nuclear Power Plant, Docket No. 50-333", dated September 16, 1977.

16. "Analytical Methods of Plant Transient Evaluations for the GE BWR," NEDO-10802 dated February 1973 and Amendments 1 and 2 dated April 1975 and June 1975, respectively.

17. Tennessee Valley Authority Application for Amendment of Facility Operating Licenses DPR-33, DPR-52 and DPR-68 dated January 12, 1977, TVA-BFNP-TS-75.

18. Tennessee Valley Authority Application for Amendment of Facility Operaring Licenses DPR-33, DPR-52 and DPR-68 dated May 11, 1977, TVA-BFNP-TS-82.

19. Tennessee Valley Authority Application for Amendment of Facility Operating Licenses DPR-33, DPR-52 and DPR-68 dated September 23, 1977, TVA-BFNP-TS-93.

20. Tennessee Valley Authority Application for Amendment of Facility Operating Licenses DPR-33, DPR-52 and DPR-68 dated November 16, 1977, TVA-BFNP-TS-97.

21. Tennessee Valley Authority Application for Amendment of Facility Operating Licenses DPR-33, DPR-52 and DPR-68 dated December 13, 1977, TVA-BFNP-TS-98.

22. Tennessee Valley Authority Application for Amendment of Facility Operating Licenses DPR-33, DPR-52 and DPR-68 dated January 3, 1978, TVA-BFNP-TS-100.

23. Handout at December 29, 1977 meeting in Bethesda, including GE letter (A. L. Vost) to TVA (L. M. Mills) on December 22, 1977, transmitting GE Safety Analysis on Lost Rubber Shoe in Browns Ferry Unit 1 Vessel.

24. Verbal statement by TVA personnel at the December 29, 1977 TVA-NRC meeting in Bethesda.

25. Consequences of a Postulated Flow Blockage Incident in a BWR, GE, NEDO-10174, October 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 35 to Facility Operating License No. DPR-33, Amendment No. 32 to Facility Operating License No. DPR-52, and Amendment No. 9 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, Unit Nos. 1, 2 and 3, (the facility) located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

Amendment No. 35 to DPR-33 changes the Technical Specifications to incorporate the limiting conditions for operation associated with Cycle 2 operation of Browns Ferry Nuclear Plant Unit 1. These changes involve a revised fuel cladding integrity safety limit for minimum critical power ratio (MCPR), revised operating limit MCPR's for both 7x7 and 8x8 fuel assemblies, the addition of linear heat generation rate (LHGR) limits for the 8x8 fuel, revised limits for the maximum average planar linear heat generation rate (MAPLHGR) for the 7x7 and 8x8 fuel assemblies, and reduced limits for scram insertion times. The revised MAPLHGR limits are based on the results of a new evaluation of the Emergency Core Cooling System (ECCS) performance submitted in compliance with our Order for Modification of License dated March 11, 1977. This amendment terminates the March 11, 1977 Order. In addition a restriction on power operation during the initial

startup for Cycle 2 has been imposed until sufficient high temperature recirculation has taken place to ensure disintegration of a rubber shoecover that had fallen into the Unit 1 vessel during the refueling outage.

Amendment Nos. 35 to DPR-33 32 to DPR-52, and 9 to DPR-68 change the Technical Specifications for each of the Browns Ferry Nuclear Plant Units to clarify the operability requirements of the Rod Worth Minimizer and the Rod Sequence Control System during scram time testing, delete the Annual Operating Report requirements, add standards for qualifications of the Health Physics Supervisor, change the frequency of cycling fire protection system valves from quarterly to annually, and substitute revised, but equivalent, terms in the equations for the limiting settings on the Average Power Range Monitors' scram and rod block setpoints.

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on September 15, 1977 (42 FR 46430) and on November 1, 1977 (42 FR 57186). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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) and 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 applications for amendments dated January 12, May 11, July 8, September 23, 26, 27, October 28, November 16, December 13, 1977, and January 3, 1978, (2) Amendment No. 35 to License No. DPR-33, Amendment No. 32 to License No. DPR-52, and Amendment No. 9 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, NW., Washington, D.C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 10th day of January 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors