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Docket No. 50-259

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 41 to Facility License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit No. 1. This Amendment changes the Technical Specifications to permit operation with one recirculation loop isolated. This change is in response to your letter dated September 15, 1978.

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

Sincerely,

/s/

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

Enclosures:
1. Amendment No. 41 to DPR-33
2. Safety Evaluation
3. Notice

cc w/enclosures:
see next page

Ok with modification particularly on p. 2 of SER and corresponding change in T.S. 9/19/78

OFFICE➤	ORB#3	ORB#3	OELD	ORB#3	RS	concurred
SURNAME➤	SSHeppard	RClark:acr		TIppolito	PCheck	
DATE➤	9/ /78	9/17 /78	9/ /78	9/19 /78	9/20 /78	CCP

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 41
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 15, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 19, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 41

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

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

   9/10
   73/74
   181/182

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

.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,

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 two recirculation loop operation.

$S \leq (0.66W + 50.7\%) \frac{FRP}{CMFLPD}$ for one recirculation loop operation.

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

(Note: These settings assume operation within the basic thermal hydraulic design criteria. These criteria are LHGR $\leq$ 18.5 kw/ft for 7X7 fuel and $\leq$ 13.4 kw/ft for 8X8 fuel and MCPR within limits of Figure 3.5.3. 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:

$$S_{RB} \leq (0.66W + 42\%) \frac{FRP}{CMFLPD}$$ for two recirculation loop operation.

$$S_{RB} \leq (0.66W + 38.7\%) \frac{FRP}{CMFLPD}$$ for one recirculation loop operation.

C. Scram and isolation--reactor low water ≥ 538 in. above vessel zero level

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

E. Scram--turbine control valve

1. Fast closure — Upon trip of the fast acting solenoid valves
2. Loss of control oil pressure ≥ 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

Amendment No. 35, 41

TABLE 3.2.C
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) for two recirculation loop operation. $\leq$ 0.66W + 37.7%(2) for one recirculation loop operation.
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)	Rod Block Logic	N/A
2(1)	RSCS Restraint (PS-85-61A & PS-85-61B)	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 $\geq$ 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 $\leq$ 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.

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.

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.

The reactor may be started and operated with one recirculation loop out of service for the duration of cycle 2 provided the MAPLHGR limits in Tables 3.5.I-1, -2, -3, & -4 are reduced by 30% for 7x7 fuel and 15% for 8x8 fuel, power level is limited to a maximum of 50% of licensed power and the suction valve in the idle loop is closed and electrically disconnected.

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.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NO. DPR-33

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-259

Introduction

By letter dated September 15, 1978, the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit No. 1 (BFNP-1). Unit No. 1 was shutdown on September 14, 1978 to investigate an apparent restriction in flow in the "A" recirculation loop. The investigation disclosed that the stem in the discharge valve was separated from the disc. To repair the valve would require a shutdown of at least four weeks. and require off loading the full core into the spent fuel pool (SFP). There are presently 324 fuel assemblies in the Unit No. 1 SFP which does not leave sufficient space to off load the full core. Unit No. 3 is presently shutdown for refueling. Unit No. 1 is scheduled to shutdown for refueling about November 1, 1978, The request for the changes to the Technical Specifications is to permit Unit No. 1 to operate with one recirculation loop isolated until the plant is shutdown for its second refueling, at which time the valve will be repaired.

Evaluation

A. Accidents (other than LOCA) and Transients Affected by One Recirculation Loop Out of Service

1. One Pump Seizure Accident

The Browns Ferry Units have two reactor water recirculation loops that feed the jet pumps. The licensee has qualitatively compared the consequences of a pump seizure accident during single loop operation with the consequences of a LOCA during full power operation with all loops in service. Previous analyses have demonstrated that the pump seizure accident is not as severe as a LOCA for two pump operation. The same conclusion can be made for the one pump case by analyzing the two events. In both events the recirculation driving loop flow is lost instantaneously, in the seizure because of pump stoppage, in the LOCA because of a line severance. In the seizure event natural circulation flow continues, water level is maintained, and the core remains submerged; thus a continuous core cooling mechanism is provided. However, for a LOCA complete flow stoppage occurs and the water level decreases resulting in core uncovery and subsequent fuel rod cladding overheating. In addition, the reactor pressure does not decrease for a pump seizure event, whereas complete depressurization occures for the LOCA. Since the potential effects of a pump seizure accident are bounded by the effects of a LOCA, the licensee has taken the position that specific pump seizure analyses for one loop operation are not necessary. Although this gives some assurance of acceptability of the pump seizure event, the Staff notes that the acceptance criteria for pump seizure are more stringent than the criteria for a LOCA. Standard Review Plan 15.3.3 (Reactor Coolant Pump Rotor Seizure, and Reactor Coolant Pump Shaft Break) requires that for the pump seizure accident, the release of radioactivity should be a fraction of 10 CFR 100 guidelines. Only limited amounts of fuel failures are acceptable for pump seizures, whereas significantly more failures are acceptable for LOCA. Therefore showing that pump seizures is less severe than LOCA does not by itself necessarily demonstrate acceptability of the pump seizure event.

Therefore, until a full analysis of the pump seizure event can be performed demonstrating that significant fuel failures do not occur for this event, the licensee will limit power during one-loop operation to the power defined by the intersection of the natural recirculation and the 100% load line on the BF-1 flow control operating map (50% of full power). That is, power will be restricted to a value (defined above) where no fuel damage will occur even if a pump seizure should occur (flow cannot decrease below the natural recirculation value no matter what happens to the recirculation pump).

BF-1 specific analyses will be performed for the one-loop operation pump seizure event and will determine a power level at which MCPR will not decrease below 1.0 during this event (this criterion is acceptable to indicate no significant fuel damage for this accident condition). These analyses should be performed utilizing methods identical to or comparably conservative with respect to the methods utilized to evaluate the transients and accident analyzed for the last BF-1 fuel loading, including conservative inputs valid for BF-1, and submitted to NRC for approval.

The staff finds the interim power limit of 50% to be acceptable on the basis that the power limit will assure no significant fuel damage will result should the pump seizure event occur during one loop operation at BF-1. Authorization for operation at a higher power level will be dependent on the results of the additional analyses to be performed.

2. Abnormal Transients

2.1. Idle Loop Startup

The previous core wide transient analyses for two-pump operation are bounding for one-loop operation, except for the idle loop startup transient analysis. In the Browns Ferry 1 FSAR the idle loop startup transient was analyzed with an initial power of 68%. Before exceeding 68% power during one-loop operation Browns Ferry Unit 1 must submit a revised idle loop startup analysis which is acceptable to the staff. We note that BF-1 has extensive Technical Specifications (Section 3.6.F.5) and Administrative Controls in place to prevent any unplanned, rapid idle loop startup. All valve positions on the isolated loop are indicated in the control room. To preclude opening the idle loop, the Technical Specifications are being amended to require that during one-loop operation that the suction valve on the idle loop be closed and electrically tagged out of service.

2.2. Flow Increase

The Minimim Critical Power Ratios (MCPRs) in the present Technical Specifications for operation at full power have previously been reviewed and found to be acceptable.
A large inadvertent flow increase could cause the MCPR to decrease below the Safety Limit MCPR for a low initial MCPR at reduced flow conditions. Therefore, the required MCPR must be increased at reduced core flow by a flow factor, K_f. The K_f factors are derived assuming both recirculation loops increase speed to the maximum permitted by the scoop tube position set screws. This condition maximizes the power increase and hence the ΔMCPR for transients initiated from less than rated conditions. When operating on one loop the flow and power increase will be less than with two pumps increasing speed, therefore the K_f factors derived from the two-pump assumption are conservative for one loop operation.

2.3 Rod Withdrawal Error

The rod withdrawal error at rated power analysis indicated that the RBM will stop rod withdrawal at a CPR which is higher than the safety limit. The MCPR requirement for one loop operation will be equal to that for two loop operation because the nuclear characteristics are independent of whether core flow is attained by one or two pump operation, if flow asymmetries are not incurred with one-loop operation. Tests at Quad Cities have shown that flow is uniform across the core for one pump operation with the equalizer valve closed. Flow resistance in the Quad Cities and Browns Ferry reactors are similar and the results of the tests at Quad Cities are considered applicable and acceptable for Browns Ferry Unit 1. However, one-pump operation results in backflow through ten of the twenty jet pumps while flow is being supplied to the lower plenum from the two active jet pumps. Because of this backflow through the inactive jet pumps the present rod-block equation and APRM settings must be modified. The licensee has modified the two-pump rod block equation and APRM settings that exist in the Technical Specification, for one-pump operation and the staff has found them acceptable.

The staff finds that one loop transients and accidents other than LOCA which is discussed below (except the idle loop startup) are bounded by the two loop operation analysis and are therefore acceptable. The present idle loop startup analysis is acceptable for single loop operation up to 68% power.

B. Loss-of-Coolant Accident

1. LOCA One-Recirculation-Loop-Out-of-Service-Model Evaluation

The General Electric evaluation model for two loop BWRs contains a detailed evaluation of system parameters to determine the blowdown heat transfer, and subsequent calculations to determine the time for rated core spray and core reflood. The heat transfer during blowdown can be characterized by a period of high heat transfer until boiling transition occurs (5-9 seconds), a period of relatively low heat transfer by pool boiling, and a subsequent period of flow film boiling heat transfer during lower plenum flashing. These heat transfer coefficients are input to the CHASTE computer program which calculates the fuel assembly hot plane temperature transients. The code assumes the appropriate values for spray heat transfer after the time of rated core spray, and terminates the temperature transient when the time for hot spot recovery occurs.

For one loop operation, General Electric has proposed a revised model to evaluate the heat transfer during blowdown. The limiting condition would occur if one loop is assumed out of service and the postulated LOCA occurred in the second operating loop. This condition would result in early flow reversal and subsequent dryout. No flow coastdown is assumed for this case as the only operating loop is assumed to be broken, whereas if the break is assumed in the idle loop, credit would be possible for flow coastdown in the operating unbroken loop. To consider this condition, the CHASTE heatup calculation assumes that dryout occurs at 0.1 second and is followed by a period of heat transfer by pool boiling until the hot spot is uncovered. Following the time of uncovery, a convective heat transfer coefficient of zero is assumed until the time of rated core spray. No credit is taken for heat transfer during lower plenum flashing. The heatup calculation is continued in the normal manner after the time of rated core spray until reflood is calculated.

To determine the significant event times to be used in the CHASTE calculations, a series of SAFE/REFLOOD calculations are performed to cover the break spectrum. These calculations provide the times for hot spot uncovery, rated core spray, and hot node reflood.

The staff is reviewing the generic LOCA model for one loop operation provided in NEDO-20566-2, Rev. 1. Our review to date has included only those portions of the report necessary for LOCA evaluation of one-loop operation for Browns Ferry Unit 1 (BF-1). That is, a complete, plant specific set of blowdown-reflood calculations were performed for BF-1, as noted below. It was therefore necessary to utilize only a small portion of the generic material (i.e., the generic heatup analyses) from the referenced generic report. We find that portion of the generic material acceptable for application to BF-1 for the remainder of the present cycle on the basis that: (1) the portion of the generic material utilized for the BF-1 "one-loop" analysis consists of a family of heatup analyses performed utilizing a computer code (CHASTE) that meets all requirements of Appendix K to 10 CFR 50.46, (2) those input parameters implicitly utilized in calculating this family of heatup analyses have been compared with the BF-1 plant specific values for those parameters and have been found to be conservative and acceptable, and (3) the parameters necessary to apply this family of analyses to BF-1 have been specifically calculated for BF-1 utilizing a computer code (SAFE-REFLOOD) that meets all requirements of Appendix K to 10 CFR 50.46. Much of the remainder of the generic report describes a systemmatic methodology for evaluating one-loop MAPLHGR limits for all plants, in many cases without plant specific blowdown-reflood analyses such as those that were performed for BF-1. That material has not been reviewed. Therefore, this approval is for this application only and should not be interpreted as providing generic approval of NEDO-20566-2, Rev. 1.

2. LOCA One-Recirculation-Loop-Out-of-Service-Model Application to Browns Ferry Unit 1

General Electric has performed ECCS-LOCA Safe-Reflood calculations for Browns Ferry 1 at 102% of rated power for a complete spectrum of large recirculation suction and discharge breaks (i.e., area ≥ 1 ft^2) with one loop-out-of-service with the equalizer valve closed, and has compared the results to calculations for Browns Ferry 1 assuming all-loops-in-service. The calculations and comparisons performed have demonstrated the effects of operating with one-loop-out-of-service on stored-energy-related phenomena (i.e., time to DNB) and on decay heat related phenomena (i.e., core uncovery times and time-to-rated-spray). Since there is no flow coastdown in the case of operation with a recirculation loop out of service (unbroken loop), nucleate boiling is not maintained and earlier DNB represents the most significant effect. Therefore, the calculations and comparisons performed demonstrate all effects of one-loop-out-of-service that might significantly affect PCT.

Generic heatup calculations have been performed with the one-loop-out-of-service LOCA model and compared to previously performed two-loop calculations. The results are presented in Figure II.A.7.4-1 of NEDO-20566-2, Rev. 1, GE LOCA Analysis - One Recirculation Loop Out of Service, July 1978. The results are in the form of a correction (reduction) factor to be applied to two-loop derived MAPLHGR limits for one-loop operation. The correction factor is presented as a function of two variables: two-loop operation reflooding time, and two-loop operation boiling transition time. The generic heatup calculations were performed for fuel at various exposures, and the most severe reduction factor at any exposure is presented. Also, the generic heatup calculations were performed assuming 7x7 fuel which is more sensitive to the amount of stored energy remaining after boiling transition, therefore yielding a greater MAPLHGR reduction than would be the case for 8x8 fuel (i.e., it is conservative to apply the reduction factors to 8x8 fuel also).

The plant-specific (BF-1) SAFE-REFLOOD large recirculation break spectrum calculations provided the parameters necessary to utilize these conservative generic heatup calculations to determine a conservative MAPLHGR reduction factor for BF-1 one-recirculation-loop-out-of-service operation. This was done for a wide range of break sizes in both the recirculation suction line and the recirculation discharge line, and the most severe reduction factor for any size and location combination was selected. This procedure insures that the one-loop-out-of-service MAPLHGR limits are derived from the most limiting break size and location for BF-1. The resulting MAPLHGR reduction factors are 0.70 for 7x7 fuel and 0.85 for 8x8 fuel, derived from the most limiting break size and location (for one loop operation) which is a discharge line break which is 66% of the maximum size double-ended guillotine discharge line break.

The single failure which results in the highest PCT (lowest MAPLHGR) is unchanged in going from two-pump to one-pump operation. The limiting failure for either case is that which results in the longest reflooding time, which is the same for both two-pump and one-pump operation.

For breaks smaller than 1.0 ft^2, the assumption of DNB at 0.1 sec. is highly conservative; nevertheless, even when this assumption is applied to the highest temperature small break for BF-1, (a break having a 2 loop PCT of < 2000 °F), it is found that the relatively long time to hot plane uncovery, during which time Ellion pool boiling is assumed, tends to compensate for the early DNB (i.e., stored heat is removed) and the PCT is not limiting (increases only about 30°F due to the conservative assumption of 0.1 sec DNB). In addition, further conservatism is introduced since the < 2000°F worst small break PCT was calculated assuming 100% of the MAPLHGR limit. In actual fact, this MAPLHGR will be reduced 15% to 30% (depending upon fuel type) for one loop operation. Therefore small breaks do not become limiting for one loop-out-of-service operation.

GE provided results of calculations showing that if the out-of-service loop is isolated from the reactor vessel, the resulting smaller amount of water inventory available for blowdown will increase PCT by less than 5°F compared to the same LOCA with an unisolated out-of-service loop. We conclude that this slight non-conservatism, and any other slight non-conservatisms associated with the methods and assumptions in the GE proposed one-loop-out-of service methods, are more than compensated for by the conservative heatup calculations (primarily the conservative assumption of DNB in 0.1 second following the break).

Summary of Accidents and Transients

We therefore conclude that the MCPR and MAPLHGR limits proposed in the BF-1 Technical Specifications for one-loop-out-of-service operation during the remainder of the present cycle are based on acceptably conservative analyses, and are therefore acceptable so long as core power does not exceed 68% of full power (assumed in the FSAR idle loop startup transient analysis).

Corrosion

The recirculation line will be closed to flow from the reactor vessel by the recirculation pump suction shutoff valve and discharge flow control valve. Under this condition, the recirculation piping will contain stagnant water at a reduced temperature and will not be subjected to the higher stresses existing during normal plant operation. The reduced stresses, temperature, and water conditions in the piping system are not expected to initiate general corrosion or stress corrosion cracking during the relatively short time period (two months) that BF-1 will be operating with one recirculation loop out of service.

Loose Parts

From the description of operating events and measures taken to detect the failed valve part, it is suspected that the failure was by shear of a part which produced relatively small or insignificant loose parts. Because of the loop design and the proposed mode of operation, there would be insufficient pressure differential to force any parts of significant size into the reactor vessel - if a loose part exists. It is therefore concluded that loose parts will not cause safety concerns during the proposed mode of operation.

Environmental Considerations

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

Conclusion

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

Dated: September 19, 1978

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-259

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 41 to Facility Operating License No. DPR-33 issued to Tennessee Valley Authority (the licensee), which revised the Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit No. 1 (the facility) located in Limestone County, Alabama. The amendment is effective as of the date of issuance.

This amendment changes the Technical Specifications to permit operation with one recirculation loop isolated.

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

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

For further details with respect to this action, see (1) the application for amendment dated September 15, 1978, (2) Amendment No. 41 to License No. DPR-33, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 19 day of September 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

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