August 17, 1984

Docket Nos. 50-260/296

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

Dear Mr. Parris:

The Commission has issued the enclosed Amendment Nos. 104 and 77 to Facility Operating License Nos. DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 2 and 3. These amendments are in response to your application dated September 21, 1981 (TVA BFNP TS 167) as supplemented by your letter dated June 3, 1982.

The amendments revise the Technical Specifications to change the neutron flux trip setting adjustment factor from a limiting safety system setting to a limiting condition for operation with a six hour action statement.

A copy of the Safety Evaluation is also enclosed.

Sincerely,

Richard J. Clark

Richard J. Clark, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 104 to License No. DPR-52
2. Amendment No. 77 to License No. DPR-68
3. Safety Evaluation

cc w/enclosures:
See next page

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Mr. Hugh G. Parris
Tennessee Valley Authority
Browns Ferry Nuclear Plant, Units 1, 2 and 3

cc:

H. S. Sanger, Jr., Esquire
General Counsel
Tennessee Valley Authority
400 Commerce Avenue
E 11B 330
Knoxville, Tennessee 37902

Mr. Ron Rogers
Tennessee Valley Authority
400 Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Mr. Charles R. Christopher
Chairman, Limestone County Commission
Post Office Box 188
Athens, Alabama 35611

Ira L. Meyers, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36130

Mr. H. N. Culver
249A HBD
400 Commerce Avenue
Tennessee Valley Authority
Knoxville, Tennessee 37902

James P. O'Reilly
Regional Administrator
Region II Office
U. S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

U. S. Environmental Protection
Agency
Region IV Office
Regional Radiation Representative
345 Courtland Street, N. W.
Atlanta, Georgia 30308

Resident Inspector
U. S. Nuclear Regulatory Commission
Route 2, Box 311
Athens, Alabama 35611

Mr. Donald L. Williams, Jr.
Tennessee Valley Authority
400 West Summit Hill Drive, W10B85
Knoxville, Tennessee 37902

George Jones
Tennessee Valley Authority
Post Office Box 2000
Decatur, Alabama 35602

Mr. Oliver Havens
U. S. Nuclear Regulatory Commission
Reactor Training Center
Osborne Office Center, Suite 200
Chattanooga, Tennessee 37411

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 104
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 21, 1981 as supplemented June 3, 1982, 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 Operating 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. 104, 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 issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 17, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 104

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

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

   ii, 9, 10, 16, 21, 23, 31, 47, 48, 74, 160a, 169, 169a (new page)

2. The marginal lines on these pages denote the area being changed.

1.1 FUEL CLADDING INTEGRITY

2.1 FUEL CLADDING INTEGRITY

b.

c. 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 ≤13.4 kw/ft for 8x8, 8x8R, and P8x8R, and MCPR within limits of Specification 3.5.k. 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.

d. The APRM Rod block trip setting shall be:

$$S_{RB} \leq (0.65W + 42\%)$$

where:

S_{RB} = Rod block setting in 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)

1.1 FUEL CLADDING INTEGRITY

2. Reactor Pressure ≤800 PSIA or Core Flow ≤10% of rated.

When the reactor pressure is ≤800 PSIA or core flow is ≤10% of rated, the core thermal power shall not exceed 823 MWt (~25% of rated thermal power).

2.1 FUEL CLADDING INTEGRITY

e. Fixed High Neutron Flux Scram Trip Setting--When the mode switch is in the RUN position, the APRM fixed high flux scram trip setting shall be: S≤120% power.

2. APRM and IRM Trip Settings (Startup and Hot Standby Modes).

a. 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.

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

Amendment No. 32, 82, ~~85~~, 104

.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 = 1.07 would not produce boiling transition. Thus, although it is not required to establish the safety limit additional margin exists between the safety limit and the actual 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.

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

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.07 when the transient is initiated from MCPR limits specified in specification 3.5.k.

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. 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 leve, 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 pressurer 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 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.

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 1.07 in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 31 inches below the normal operating range and is thus adequate to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

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

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50% greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first state pressure.

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.

C. When it is determined that a channel is failed in the unsafe condition, the other RPS channe that monitor the same variable shall be functionally tested immediately before the trip sys tem containing the failure is tripped. The trip system containing the unsafe failure may untripped for short periods of time to allow functional testin 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.

4.1 BASES

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 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.

4.1 BASES

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. The APRM system, which uses the LPRM readings to detect a change in thermal power, will be calibrated every seven days using a heat balance to compensate for this change in sensitivity. The RBM system uses the LPRM reading to detect a localized change in thermal power. It applies a correction factor based on the APRM output signal to determine the percent thermal power and therefore any change in LPRM sensitivity is compensated for by the APRM calibration. The technical specification limits of CMFLPD, CPR, and MAPLHGR are determined by the use of the process computer or other backup methods. These methods use LPRM readings and TIP data to determine the power distribution.

Compensation in the process computer for changes in LPRM sensitivity will be made by performing a full core TIP traverse to update the computer calculated LPRM correction factors every 1000 effective full power hours.

As a minimum the individual LPRM meter readings will be adjusted at the beginning of each operating cycle before reaching 100 percent power.

1. The minimum number of operable channels for each trip function is detailed for the startup and run positions of the reactor mode selector switch. The SRM, IRM, and APRM (startup mode), blocks need not be operable in "run" mode, and the APRM (flow biased) rod blocks need not be operable in "startup" mode.

   With the number of OPERABLE channels less than required by the minimum OPERABLE channels ~~per~~ trip function requirement, place at least one inoperable channel in the tripped condition within one hour.

2. W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (3293 MWt).

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

4. SRM's A and C downscale functions are bypassed when IRM's A, C, E, and G are above range 2. SRM's B and D downscale function is bypassed when IRM's B, D, F, and H are above range 2.

   SRM detector not in startup position is bypassed when the count rate is ≥100 CPS or the above condition is satisfied.

5. During repair or calibration of equipment, not more than one SRM or RBM channel nor more than two APRM or IRM channels may be bypassed. Bypassed channels are not counted as operable channels to meet the minimum operable channel requirements. Refer to section 3.10.B for SRM requirements during core alterations.

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

   IRM channels B, F, D, H all in range 8 or above bypasses SRM channels B and D functions.

7. The following operational restraints apply to the RBM only.

   a. Both RBM channels are bypassed when reactor power is ≤ 30% and when a peripheral control rod is selected.
   b. The RBM need not be operable in the "startup" position of the reactor mode selector switch.

   c. Two RBM channels are provided and only one of these may be bypassed from the console. If the inoperable channel cannot be restored within 24 hours, the inoperable channel shall be placed in the tripped condition within one hour.

   d. If minimum conditions for Table 3.2.C are not met, administrative controls shall be immediately imposed to prevent control rod withdrawal.

3.5 Core and Containment Cooling Systems

L. APRM Setpoints

1. Whenever the core thermal power is $>$ 25% of rated, the ration of FRP/CMFLPD shall be $\geq$ 1.0, or the APRM scram and rod block setpoint equations listed in sections 2.1.A and 2.1.B shall be multiplied by FRP/CMFLPD as follows:

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

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

2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.

3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to $\leq$ 25% of rated thermal power within 4 hours.

M. Reporting Requirements

If any of the limiting values identified in Specifications 3.5.I, J, K, or L.3 are exceeded and the specified remedial action is taken, the event shall be logged and reported in a 30-day written report.

4.5 Core and Containment Cooling Systems

L. APRM Setpoints

FRP/CMFLPD shall be determined daily when the reactor is $\geq$ 25% of rated thermal power.

3.5.J. Linear Heat Generatior ate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated.

The LHGR

shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the R factor would have to be less than 0.241 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns, which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MPCR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

3.5.L APRM Setpoints

Operation is constrained to a maximum LHGR of 18.5 kW/ft for 7x7 fuel and 13.4 kW/ft for 8x8, 8x8R, and P8x8R. This limit is reached when core maximum fraction of limiting power density (CMFLPD) equals 1.0. For the case where CMFLPD exceeds the fraction of rated thermal power, operation is permitted only at less than 100-percent rated power and only with APRM scram settings as required by specification 3.5.L.1. The scram trip setting and rod block trip setting are adjusted to ensure that no combination of CMFLPD and FRP will increase the LHGR transient peak beyond that allowed by the 1-percent plastic strain limit. A 6-hour time period to achieve this condition is justified since the additional margin gained by the setdown adjustment is above and beyond that ensured by the safety analysis.

3.5.M Reporting Requirements

The LCO's associated with monitoring the fuel rod operating conditions are required to be met at all times, i.e., there is no allowable time in which the plant can knowingly exceed the limiting values for MAPLHGR, LHGR, and MCPR. It is a requirement, as stated in Specification 3.5.I, J, and K, that if at any time during steady state power operation it is determined that the limiting values for MAPLHGR, LHGR, or MCPR are exceeded, action is then initiated to restore operation to within the prescribed limits. This action is initiated as soon as normal surveillance indicates that an operating limit has been reached. Each event involving steady state operation beyond a specified limit shall be reported within 30 days. It must be recognized that there is always an action which would return any of the parameters (MAPLHGR, LHGR, or MCPR) to within prescribed limits, namely power reduction. Under most circumstances, this will not be the only alternative.

3.5.N References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEIM-10735, August 1973.

2. Suplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 14, 1974 (USA Regulatory Staff).

3. Communication: V. A. Moore to I.S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

4. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

5. Letter from R. H. Buchholz (GE) to P. S. Check (NRC), "Response to NRC request for information on ODYN computer model," September 5, 1980.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 77
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 21, 1981 as supplemented June 3, 1982, 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 Operating License No. DPR-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 77, 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 issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 17, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 77

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

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

   ii, 10, 11, 16, 20, 22, 31, 46, 47, 77, 167a, 177, 178

2. The marginal lines on these pages denote the area being changed.

Amendment No. 77

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
1.1 FUEL CLADDING INTEGRITY	2.1 FUEL CLADDING INTEGRITY
	W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)
	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 ≤ 13.4kW/ft and MCPR within limits of specification 3.5.K.

1.1 FUEL CLADDING INTEGRITY

2.1 FUEL CLADDING INTEGRITY

If it is determined that either of these design criteria is being violated during operation, action shall be initiated within 15 minutes to restore operation within the prescribed limits.

Surveillance requirements for APRM scram setpoints are given in Specification 4.1.B).

d. The APRM Rod block trip setting shall be:

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

where:

S_{RB} = Rod block setting in 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) uncertainties employed in deriving the safety limit are provided at the beginning of each fuel cycle.

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

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°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 applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

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

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

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

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

3. IRM-Flux Scram Trip Setting

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

*** See Section 3.5.K.

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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 varied with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during the steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system.

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

The set point for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR subsection N14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than 1.05 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 trip anticipates the pressure- neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2).

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control

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.

C. When it is determined that a channel is failed in the unsafe condition, the other RPS channels 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.

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 Table 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.

4.1 BASES

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. The APRM system, which uses the LPRM readings to detect a change in thermal power, will be calibrated every seven days using a heat balance to compensate for this change in sensitivity. The RBM system uses the LPRM reading to detect a localized change in thermal power. It applies a correction factor based on the APRM output signal to determine the percent thermal power and therefore any change in LPRM sensitivity is compensated for by the APRM calibration. The technical specification limits of CMFLPD, CPR, and MAPLHGR are determined by the use of the process computer or other backup methods. These methods use LPRM readings and TIP data to determine the power distribution.

Compensation in the process computer for changes in LPRM sensitivity will be made by performing a full core TIP traverse to update the computer calculated LPRM correction factors every 1000 effective full power hours.

As a minimum the individual LPRM meter readings will be adjusted at the beginning of each operating cycle before reaching 100 percent power.

1. The minimum number of operable channels for each trip function is detailed for the startup and run positions of the reactor mode selector switc The SRM, IRM, and APRM (star mode), blocks need not be operable in "run" mode, and the APRM (flow biased) rod blocks need not be operable in "startup" mode.

   With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip function requirement, place at least one inoperable channel in the tripped condition within one hour.

2. W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (3293 MWt).

   See Specification 2.1 for APRM control rod block setpoint.

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

4. SRM's A and C downscale functions are bypassed when IRM's A, C, E, and G are above range 2. SRM's B and D downscale function is bypassed when IRM's B, D, F, and H are above range 2.

   SRM detector not in startup position is bypassed when the count rate is ≥100 CPS or the above condition is satisfied.

5. During repair or calibration of equipment, not more than one SRM or RBM channel nor more than two APRM or IRM channels may be bypassed. Bypassed channels are not counted as operable channels to meet the minimum operable channel requirements. Refer to section 3.10.B for SRM requirements during core alterations.

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

   IRM channels B, F, D, H all in range 8 or above bypasses SRM channels B and D functions.

7. The following operational restraints apply to the RBM only.

   a. Both RBM channels are bypassed when reactor power is ≤ 30% and when a peripheral control rod is selected.
   b. The RBM need not be operable in the "startup" position of the reactor mode selector switch.
   c. Two RBM channels are provided and only one of these may be bypassed from the console. If the inoperable channel cannot be restored within 24 hours, the inoperable channel shall be placed in the tripped condition within one hour.
   d. If minimum conditions for Table 3.2.C are not met, administrative controls, shall be immediately imposed to prevent control rod withdrawal.

3.5 Core and Containment Cooling Systems

L. APRM Setpoints

1. Whenever the core thermal power is $>$ 25% of rated, the ration of FRP/CMFLPD shall be $\geq$ 1.0, or the APRM scram and rod block setpoint equations listed in sections 2.1.A and 2.1.B shall be multiplied by FRP/CMFLPD as follows:

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

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

2. When it is determined that

   3.5.L.1 is not being met, 6 hours is allowed to correct the condition.

3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to $\leq$ 25% of rated thermal power within 4 hours.

M. Reporting Requirements

If any of the limiting values identified in Specifications 3.5.I, J, K, or L.3 are exceeded and the specified remedial action is taken, the event shall be logged and reported in a 30-day written report.

4.5 Core and Containment Cooling Systems

L. APRM Setpoints

FRP/CMFLPD shall be determined daily when the reactor is $\geq$ 25% of rated thermal power.

generation if fuel pellet densification is postulated.

The LHGR shall be checked daily during reactor operation at ≥ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at thermal limit.

3.5.L APRM Setpoints

Operation is constrained to a maximum LHGR of 18.5 kW/ft for 7x7 fuel and 13.4 kW/ft for 8x8, 8x8R, and P8x8R. This limit is reached when core maximum fraction of limiting power density (CMFLPD) equals 1.0. For the case where CMFLPD exceeds the fraction of rated thermal power, operation is permitted only at less than 100-percent rated power and only with APRM scram settings as required by specification 3.5.L.1. The scram trip setting and rod block trip setting are adjusted to ensure that no combination of CMFLPD and FRP will increase the LHGR transient peak beyond that allowed by the 1-percent plastic strain limit. A 6-hour time period to achieve this condition is justified since the additional margin gained by the setdown adjustment is above and beyond that ensured by the safety analysis.

3.5.M Reporting Requirements

The LCO's associated with monitoring the fuel rod operating conditions are required to be met at all times, i.e., there is no allowable time in which the plant can knowingly exceed the limiting values for MAPLHGR, LHGR, and MCPR. It is a requirement, as stated in Specification 3.5.I, J, and K, that if at any time during steady state power operation it is determined that the limiting values for MAPLHGR, LHGR, or MCPR are exceeded, action is then initiated to restore operation to within the prescribed limits. This action is initiated as soon as normal surveillance indicates that an operating limit has been reached. Each event involving steady state operation beyond a specified limit shall be reported within 30 days. It must be recognized that there is always an action which would return any of the parameters (MAPLHGR, LHGR, or MCPR) to within prescribed limits, namely power reduction. Under most circumtances, this will not be the only alternative.

N. References

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3, NEDO-24194A and Addenda.

2. "BWR Transient Analysis Model Utilizing the RETRAN Program," TVA-TR81-01-A.

3. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 77 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3

DOCKET NOS. 50-260 AND 50-296

1.0 Introduction

By letters dated September 21, 1981 and June 3, 1982, the Tennessee Valley Authority (the licensee or TVA) requested amendments to Facility Operating License Nos. DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 2 and 3. The amendments would modify the Technical Specifications by transferring one of the equations relating to Average Power Range Monitor (APRM) trip setting adjustments from the Limiting Safety System Settings (LSSS) section of the Technical Specifications to the Limiting Conditions for Operation (LCO) section. The equation involved is one that relates the fraction of rated thermal power (FRP) and core maximum fraction of limiting power density (CMFLPD) to flow in the recirculation system loops. The amendment would also establish a time period of six hours for completing corrective actions if the ratio of FRP/CMPLPD is outside acceptable limits.

2.0 Evaluation

Under the old departure from nucleate boiling (DNB) heat transfer correlation based on the Hench-Levy method, the figure of merit was the critical heat flux ratio (CHFR), i.e., the ratio of the critical heat flux for boiling transition to the existing local heat flux. Since the existing local heat flux is directly proportional to the product of reactor power and the total peaking factor at the point of CHFR, any increase in the peaking factor resulted in a corresponding decrease in CHFR. Thus, under the old minimum critical heat flux ratio (MCHFR) correlations, the peaking factor (MFLPD/FRP) adjustment to the flow biased scram and rod block equations had relevance to maintaining core limits in certain flow excursion transients.

As a result of extensive experimental tests conducted by the General Electric Company (GE), it was demonstrated that the transition boiling point can be predicted with definable accuracy by plotting critical quality (Xc) as a function of distance from initiation of bulk boiling (Boiling Length $-L_B$) in a fuel bundle. This is referred to as the GEXL correlation. With the introduction of the GEXL correlation, the functional form of the safety limit and operational limit for preventing DNB became the minimum critical power ratio (MCPR). Determination of an operating limit MCPR is based on analysis initiated from rated core conditions assuming a fixed 120 percent flux scram. The operating limit is determined from results of transient analyses, both core-wide and localized events. The resultant change in CPR due to the transients is added to safety limit MCPR to determine the necessary operating limit MCPR to ensure adequate thermal margin. Additionally, the required operating limit is increased at reduced core flow to ensure the safety limit is not violated in the event of a flow increase transient. Since power shape is essentially accounted for in the calculation which determines actual CPR margin, it is not necessary to reduce the scram setpoints as a function of peaking factor.

The MCPR safety analyses take no credit for an APRM flow-biased scram, and consequently, this flow-biased scram does not ensure additional margin to the safety limit MCPR. Since the flow-biased scram does not ensure additional margin to the safety limit MCPR, the six hours allowed for corrective action does not result in a decrease in the margin of safety. Similar revisions to the Browns Ferry Unit 1 Facility Operating License No. DPR-33 were granted by Amendment No. 76, September 15, 1981. On this basis, there is sufficient justification for relaxing the corrective action and time allowances with regard to the standard core limits (MCPR, LHGR, etc.).

3.0 Environmental Considerations

The amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

4.0 Conclusion

We have concluded, based on the considerations discussed above, 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.

Principal Contributor: R. Clark

Dated: August 17, 1984