April 30, 1993

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

See Proposed Change to Tech Specs

Dr. Mark O. Medford, Vice President
Nuclear Assurance, Licensing & Fuels
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Dr. Medford:

SUBJECT: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS REGARDING ROD SEQUENCE CONTROL SYSTEM DELETION AND ROD WORTH MINIMIZER SETPOINT CHANGE (TS 310) TAC NOS. M84184, M84185, AND M84186)

The Commission has issued the enclosed Amendment Nos. 196 , 212 ,and 169 to Facility Operating Licenses Nos. DPR-33, DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3, respectively. These amendments are in response to your application dated July 20, 1992, as supplemented March 18, 1993, requesting changes to the BFN Technical Specifications to delete requirements associated with the Rod Sequence Control System, and to decrease the power level setpoint above which the Rod Worth Minimizer is no longer required.

A copy of the NRC's Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
Original signed by
Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 196 to License No. DPR-33
2. Amendment No. 212 to License No. DPR-52
3. Amendment No. 169 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

OFC	PDII-4LA	PDII-4PM	PDII-4PM	SRXB	OGC	PDII-4D
NAME	MSanders	TRoss	JWilliams	RJones		FHebdon
DATE	4/15/93	4/19/93	4/19/93	4/16/93	4/20/93	4/30/93

DOCUMENT NAME: ts310.amd

DF01 1/1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 30, 1993

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

Dr. Mark O. Medford, Vice President
Nuclear Assurance, Licensing & Fuels
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Dr. Medford:

SUBJECT: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS REGARDING ROD SEQUENCE CONTROL SYSTEM DELETION AND ROD WORTH MINIMIZER SETPOINT CHANGE (TS 310) (TAC NOS. M84184, M84185, AND M84186)

The Commission has issued the enclosed Amendment Nos. 196, 212 ,and 169 to Facility Operating Licenses Nos. DPR-33, DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3, respectively. These amendments are in response to your application dated July 20, 1992, as supplemented March 18, 1993, requesting changes to the BFN Technical Specifications to delete requirements associated with the Rod Sequence Control System, and to decrease the power level setpoint above which the Rod Worth Minimizer is no longer required.

A copy of the NRC's Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.196 to
   License No. DPR-33
2. Amendment No.212 to
   License No. DPR-52
3. Amendment No.169 to
   License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

Tennessee Valley Authority
ATTN: Dr. Mark O. Medford

Browns Ferry Nuclear Plant

cc:
Mr. John B. Waters, Chairman
Tennessee Valley Authority
ET 12A
400 West Summit Hill Drive
Knoxville, Tennessee 37902

Mr. J. R. Bynum, Vice President
Nuclear Operations
3B Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Site Licensing Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, Alabama 35602

Mr. O. J. Zeringue, Vice President
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, Alabama 35602

Mr. M. J. Burzynski, Manager
Nuclear Licensing and Regulatory Affairs
5B Lookout Place
Chattanooga, Tennessee 37402-2801

TVA Representative
Tennessee Valley Authority
11921 Rockville Pike
Suite 402
Rockville, Maryland 20852

General Counsel
Tennessee Valley Authority
ET 11H
400 West Summit Hill Drive
Knoxville, Tennessee 37902

Chairman, Limestone County Commission
P.O. Box 188
Athens, Alabama 35611

State Health Officer
Alabama Dept. of Public Health
434 Monroe Street
Montgomery, Alabama 36130-1701

Regional Administrator
U.S.N.R.C. Region II
101 Marietta Street, N.W.
Suite 2900
Atlanta, Georgia 30323

Mr. Charles Patterson
Senior Resident Inspector
Browns Ferry Nuclear Plant
U.S.N.R.C.
Route 12, Box 637
Athens, Alabama 35611

Site Quality Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P. O. Box 2000
Decatur, Alabama 35602

AMENDMENT NO. 196 FOR BROWNS FERRY UNIT 1 - DOCKET NO. 50-259
AMENDMENT NO. 212 FOR BROWNS FERRY UNIT 2 - DOCKET NO. 50-260
AMENDMENT NO. 169 FOR BROWNS FERRY UNIT 3 - DOCKET NO. 50-296
DATED: April 30, 1993

<u>Distribution</u>

Docket File
NRC & Local PDRs
BFN Reading
S. Varga
F. Hebdon
M. Sanders
J. Williams
T. Ross
OGC
D. Hagan
G. Hill (4)
Wanda Jones
C. Grimes
ACRS (10)
OPA
OC/LFDCB
E. Merschoff
P. Frederickson
C. Patterson

cc: Plant Service list

050025

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

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.196 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 212 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

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

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

1.0 INTRODUCTION

By letter dated July 20, 1992 as supplemented March 18, 1993, the Tennessee Valley Authority (the licensee) submitted a request for changes to the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3 Technical Specifications (TS) and associated Bases. The staff's proposed finding of no significant hazards considerations is unaffected by the March 18, 1993 supplement. The requested changes would remove the Rod Sequence Control System (RSCS) and decrease the power level setpoint above which the Rod Worth Minimizer (RWM) is no longer required.

2.0 DISCUSSION

The RSCS restricts control rod movement to minimize the individual worth of control rods to lessen the consequences of a Rod Drop Accident (RDA). Control rod movement is restricted through the use of rod select, insert, and withdrawal blocks. The RSCS is a hard-wired (as opposed to a computer controlled), redundant backup system to the RWM. It is independent of the RWM in terms of inputs and outputs, but the two systems are compatible. The RSCS is designed to monitor and block, when necessary, operator actions to select, withdraw, or insert control rods, and thus assist in preventing significant control rod pattern errors which could lead to a control rod with a high potential reactivity worth. A significant pattern error is one of several abnormal conditions that must occur for an RDA to exceed the fuel energy density limit criteria. The RSCS was designed only for possible mitigation of the RDA and is active only during low power (currently less than 20 percent rated power) when an RDA might be significant. It provides rod blocks on detection of a significant pattern error. It does not, by itself, prevent an RDA. A similar pattern control function is provided by the RWM, a computer-controlled system.

By letter dated August 15, 1986, the BWR Owner's Group (BWROG), in cooperation with the General Electric Company (GE), proposed Amendment 17 to GE Licensing Topical Report NEDE-24011-P-A which would eliminate the requirement for the RSCS and retain the RWM but lower the setpoint for turnoff (during startup) or

9305060268 930430
PDR ADOCK 05000259
P PDR

turnon (during shutdown) from 20 to 10 percent. The NRC staff review of this report, documented by a letter by A. Thadani (USNRC) dated December 27, 1987 to J. Charnley (General Electric), concluded that the proposed changes were acceptable, and approved Amendment 17, but imposed several additional requirements which would be necessary to implement the changes. The additional requirements were:

1. The TS should require provisions for minimizing operations without the RWM system operable.

2. The occasional necessary use of a second operator as a replacement for the RWM should be strengthened by a utility review of relevant procedures, related forms and quality control to assure that the second operator provides an effective and truly independent monitoring process. A discussion of this review should accompany the request for RSCS removal.

3. Rod patterns used should be at least equivalent to Banked Position Withdrawal Sequence (BPWS) patterns.

3.0 EVALUATION

The licensee has proposed changes to several TS and associated Bases. These changes are:

1. Sections 3.3.A.2.d, 4.3.A.2.b, 3.3/4.3.B.3.a, and Tables 3/4.2.C were edited since they contained requirements pertaining only to the RSCS.

2. Sections 4.3.A.2.a and 4.3.B.1.a were edited to remove references to the RSCS (4.3.B.1.a also had an administrative change).

3. Sections 4.3.B.3.b.1.a, 4.3.B.3.b.2.a, 3.3.B.3.c and 3.3.B.3.b.1, 2, and 3 were edited as a result of the lowering of the RWM setpoint (3.3.B.3.c also had an administrative change).

4. Section 4.3.C.1 was edited since it contained references to RSCS requirements and the RWM setpoint, and to make an administrative change.

5. Section 4.3.B.3.b.3 was edited to make an administrative change.

6. The Table of Contents was changed to reflect a change in page numbers.

7. The BASES, Sections 3.3/4.3.A.2, 3.3/4.3.B.1 and 3, and 3.3/4.3.C were edited to appropriately reflect the changes made in TS.

8. Reference to the RSCS was deleted from Definition 1.M Note 4.

9. Reference to the RSCS was deleted from Bases 2.1.2.

Changes 1 through 7 were submitted by the July 20, 1992 letter. Changes 8 and 9 were part of the March 18, 1993 supplement.

These changes implement three items:

A. Elimination of the RSCS requirements.

B. Reduction of the RWM setpoint to 10 percent of full power.

C. Increased administrative control of RWM operability (intended to result in decreased use of the second operator as a substitute for the RWM). The licensee has also discussed the procedures for second operator actions, when required, to ensure independent monitoring of the control rod patterns. BPWS control rod patterns are already required by the TS. However, this requirement has been reemphasized in several of the TS changes.

NRC staff review of the generic basis for removal of the RSCS and reducing the RWM setpoint is provided in the December 27, 1987 letter referenced above. The staff has reviewed the licensee's proposed TS changes relevant to items A and B above, and concludes that these proposed changes fall within the scope of the generic staff evaluation. The staff finds that the proposed TS changes are appropriate, clearly stated, and satisfy relevant technical criteria.

The licensee has also proposed provisions which minimize operations without the RWM operable (item C, above). The proposed revision to the TS requires the RWM to be operable at the beginning of each startup, with only one exception per year. This follows the pattern of previously approved RWM TS and previous reviews for RSCS removal (e.g., Hatch, see safety evaluation dated May 20, 1992). These changes have been previously found to provide the desired improvement in reliability for the system. Also, as required, the TS and procedures for the use of a second operator (when the RWM is inoperable) have been reviewed by the licensee. Staff review of the licensee's submittal confirms that appropriate administrative measures are provided to ensure a suitable independent check on the rod patterns used. Finally, as required, the TS revision prescribes the use of rod patterns equivalent to the BPWS patterns approved by previous staff reviews to maintain low control rod reactivity. The proposed changes to the TS and bases appropriately implement these changes.

In conclusion, the NRC staff has reviewed the reports submitted by the licensee for proposing TS changes relating to the removal of the RSCS and decrease of the power level setpoint above which the RWM is no longer required. Based on this review, we have concluded that appropriate documentation was submitted and the proposed TS changes satisfy staff positions and requirements in these areas. Therefore, the proposed TS changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change the Surveillance Requirements and Bases. The NRC 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 (57 FR 48827). 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.

6.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: C. Beardslee and J. Williams

Date: April 30, 1993

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 196
License No. DPR-33

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

A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 20, 1992, as supplemented March 18, 1993, 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-33 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 196, 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 its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: April 30, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 96

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

UNIT 1
EFFECTIVE PAGE LIST

REMOVE	INSERT
i	i
ii	ii*
1.0-3	1.0-3*
1.0-4	1.0-4
1.1/2.1-12	1.1/2.1-12*
1.1/2.1-13	1.1/2.1-13
3.2/4.2-25	3.2/4.2-25
---	3.2/4.2-25a
3.2/4.2-50	3.2/4.2-50
---	3.2/4.2-50a
3.3/4.3-1	3.3/4.3-1*
3.3/4.3-2	3.3/4.3-2
3.3/4.3-3	3.3/4.3-3
3.3/4.3-4	3.3/4.3-4
3.3/4.3-5	3.3/4.3-5
3.3/4.3-6	3.3/4.3-6
3.3/4.3-7	3.3/4.3-7
3.3/4.3-8	3.3/4.3-8
3.3/4.3-9	3.3/4.3-9
3.3/4.3-10	3.3/4.3-10
3.3/4.3-13	3.3/4.3-13*
3.3/4.3-14	3.3/4.3-14
3.3/4.3-15	3.3/4.3-15
3.3/4.3-16	3.3/4.3-16
3.3/4.3-19	3.3/4.3-19
3.3/4.3-20	3.3/4.3-20*

*DENOTES OVERLEAF PAGES

<u>TABLE OF CONTENTS</u>

Section		Page No.
1.0	Definitions	1.0-1
	SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS	
1.1/2.1	Fuel Cladding Integrity	1.1/2.1-1
1.2/2.2	Reactor Coolant System Integrity.	1.2/2.2-1
	LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS	
3.1/4.1	Reactor Protection System..	3.1/4.1-1
3.2/4.2	Protective Instrumentation.	3.2/4.2-1
	A. Primary Containment and Reactor Building Isolation Functions.	3.2/4.2-1
	B. Core and Containment Cooling Systems - Initiation and Control	3.2/4.2-1
	C. Control Rod Block Actuation.	3.2/4.2-2
	D. Radioactive Liquid Effluent Monitoring Instrumentation.	3.2/4.2-3
	E. Drywell Leak Detection	3.2/4.2-4
	F. Surveillance Instrumentation	3.2/4.2-4
	G. Control Room Isolation	3.2/4.2-4
	H. Flood Protection	3.2/4.2-4
	I. Meteorological Monitoring Instrumentation. . .	3.2/4.2-4
	J. Seismic Monitoring Instrumentation	3.2/4.2-5
	K. Radioactive Gaseous Effluent Monitoring Instrumentation	3.2/4.2-6
	L. ATWS Recirculation Pump Trip	3.2/4.2-6a
3.3/4.3	Reactivity Control.	3.3/4.3-1
	A. Reactivity Limitations	3.3/4.3-1
	B. Control Rods	3.3/4.3-5
	C. Scram Insertion Times.	3.3/4.3-9

Section		Page No.
	D. Reactivity Anomalies	3.3/4.3-11
	E. Reactivity Control	3.3/4.3-12
	F. Scram Discharge Volume	3.3/4.3-12
3.4/4.4	Standby Liquid Control System	3.4/4.4-1
	A. Normal System Availability	3.4/4.4-1
	B. Operation with Inoperable Components	3.4/4.4-3
	C. Sodium Pentaborate Solution.	3.4/4.4-3
3.5/4.5	Core and Containment Cooling Systems.	3.5/4.5-1
	A. Core Spray System (CSS).	3.5/4.5-1
	B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)	3.5/4.5-4
	C. RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS). . .	3.5/4.5-9
	D. Equipment Area Coolers	3.5/4.5-13
	E. High Pressure Coolant Injection System (HPCIS).	3.5/4.5-13
	F. Reactor Core Isolation Cooling System (RCICS).	3.5/4.5-14
	G. Automatic Depressurization System (ADS). . . .	3.5/4.5-16
	H. Maintenance of Filled Discharge Pipe	3.5/4.5-17
	I. Average Planar Linear Heat Generation Rate . .	3.5/4.5-18
	J. Linear Heat Generation Rate (LHGR)	3.5/4.5-18
	K. Minimum Critical Power Ratio (MCPR).	3.5/4.5-19
	L. APRM Setpoints	3.5/4.5-20
3.6/4.6	Primary System Boundary	3.6/4.6-1
	A. Thermal and Pressurization Limitations	3.6/4.6-1
	B. Coolant Chemistry.	3.6/4.6-5
	C. Coolant Leakage.	3.6/4.6-9
	D. Relief Valves.	3.6/4.6-10

H. Reactor Power Operation - Reactor power operation is any operation in the STARTUP/HOT STANDBY or RUN MODE with the reactor critical and above 1 percent rated power.

I. STARTUP CONDITION - The reactor is in the STARTUP CONDITION when the withdrawal of control rods for the purpose of making the reactor critical has begun, reactor power is less than or equal to 1 percent of rated, and the reactor is in the STARTUP/HOT STANDBY MODE.

J. HOT STANDBY CONDITION - The reactor is in the HOT STANDBY CONDITION when reactor power is less than or equal to 1 percent of rated. The reactor is in the STARTUP/HOT STANDBY MODE, and the reactor is not in the STARTUP CONDITION. The reactor coolant temperature may be greater than 212° F.

Note that a HOT STANDBY CONDITION cannot exist simultaneously with a STARTUP CONDITION due to the difference in intent. A HOT STANDBY CONDITION exists when the reactor mode switch is placed in the STARTUP/HOT STANDBY position (for example, to comply with an LCO) and power level has been reduced to 1 percent or lower. Anytime control rods are being withdrawn for the purpose of increasing reactor power level, the reactor mode switch has been placed in the STARTUP/HOT STANDBY position, and reactor power level is at or below one percent, a STARTUP CONDITION exists.

K. SHUTDOWN CONDITION - The reactor is in the SHUTDOWN CONDITION when the reactor is in the Shutdown or Refuel Mode.

1. HOT SHUTDOWN CONDITION - The reactor is in the HOT SHUTDOWN CONDITION when reactor coolant temperature is greater than 212° F and the reactor is in the SHUTDOWN CONDITION.

2. COLD SHUTDOWN CONDITION - The reactor is in the COLD SHUTDOWN CONDITION when reactor coolant temperature is equal to or less than 212° F and the reactor is in the SHUTDOWN CONDITION.

L. COLD CONDITION - The reactor is in the COLD CONDITION when reactor coolant temperature is equal to or less than 212° F in any Mode of Operation (except as defined in K.2 above).

M. Mode of Operation - The reactor mode switch position determines the Mode of Operation of the reactor when there is fuel in the reactor vessel, except that the Mode of Operation may remain unchanged when the reactor mode switch is temporarily moved to another position as permitted by the notes. When there is no fuel in the reactor vessel, the reactor is considered not to be in any Mode of Operation or operational condition. The reactor mode switch may then be in any position or may be inoperable.

1. Startup/Hot Standby Mode - The reactor is in the STARTUP/HOT STANDBY MODE when the reactor mode switch is in the "STARTUP/HOT STANDBY" position. This is often referred to as just the STARTUP MODE.

2. Run Mode - The reactor is in the Run Mode when the reactor mode switch is in the "Run" position.

3. Shutdown Mode - The reactor is in the Shutdown Mode when the reactor mode switch is in the "Shutdown" position.(1)(2)(3)(4)

4. Refuel Mode - The reactor is in the Refuel Mode when the reactor mode switch is in the "Refuel" position.(1)

---

(1) The reactor mode switch may be placed in any position to perform required tests or maintenance authorized by the shift operations supervisor, provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

(2) The reactor mode switch may be placed in the "Refuel" position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.10.A.5 provided that reactor coolant temperature is equal to or less than 212° F.

(3) The reactor mode switch may be placed in the "Refuel" position while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock is OPERABLE.

(4) The reactor mode switch may be placed in the "Startup/Hot Standby" position and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

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 3,440 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 setpoints are discussed below:

A. Neutron Flux Scram

1. APRM Flow-Biased 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 core average neutron flux.

During transients, the instantaneous fuel surface heat flux is less than the instantaneous neutron flux by an amount depending upon the duration of the transient and the fuel time constant. For this reason, the flow-biased scram APRM flux signal is passed through a filtering network with a time constant which is representative of the fuel time constant. As a result of this filtering, APRM flow-biased scram will occur only if the neutron flux signal is in excess of the setpoint and of sufficient time duration to overcome the fuel time constant and result in an average fuel surface heat flux which is equivalent to the neutron flux trip setpoint. This setpoint is variable up to 120 percent of rated power based on recirculation drive flow according to the equations given in Section 2.1.A.1 and the graph in Figure 2.1-2. For the purpose of licensing transient analysis, neutron flux scram is assumed to occur at 120 percent of rated power. Therefore, the flow biased provides additional margin to the thermal limits for slow transients such as loss of feedwater heating. No safety credit is taken for flow-biased scrams.

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. Thus, of all 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 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.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5)	Function	Trip Level Setting
4(1)	APRM Upscale (Flow Bias)	$\leq$0.66W + 42% (2)
4(1)	APRM Upscale (Startup Mode) (8)	$\leq$12%
4(1)	APRM Downscale (9)	$\geq$3%
4(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (Flow Bias)	$\leq$0.66W + 40% (2)(13)
2(7)	RBM Downscale (9)	$\geq$3%
2(7)	RBM Inoperative	(10c)
6(1)	IRM Upscale (8)	$\leq$108/125 of full scale
6(1)	IRM Downscale (3)(8)	$\geq$5/125 of full scale
6(1)	IRM Detector not in Startup Position (8)	(11)
6(1)	IRM Inoperative (8)	(10a)
3(1) (6)	SRM Upscale (8)	$\leq$ 1X10^5 counts/sec.
3(1) (6)	SRM Downscale (4)(8)	$\geq$3 counts/sec.
3(1) (6)	SRM Detector not in Startup Position (4)(8)	(11)
3(1) (6)	SRM Inoperative (8)	(10a)
2(1)	Flow Bias Comparator	$\leq$10% difference in recirculation flows
2(1)	Flow Bias Upscale	$\leq$115% recirculation flow
1	Rod Block Logic	N/A
1(12)	High Water Level in West Scram Discharge Tank (LS-85-45L)	$\leq$25 gal.
1(12)	High Water Level in East Scram Discharge Tank (LS-85-45M)	$\leq$25 gal.

THIS PAGE INTENTIONALLY LEFT BLANK

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

Function	Functional Test	Calibration (17)	Instrument Check
APRM Upscale (Flow Bias)	(1) (13)	once/3 months	once/day (8)
APRM Upscale (Startup Mode)	(1) (13)	once/3 months	once/day (8)
APRM Downscale	(1) (13)	once/3 months	once/day (8)
APRM Inoperative	(1) (13)	N/A	once/day (8)
RBM Upscale (Flow Bias)	(1) (13)	once/6 months	once/day (8)
RBM Downscale	(1) (13)	once/6 months	once/day (8)
RBM Inoperative	(1) (13)	N/A	once/day (8)
IRM Upscale	(1)(2) (13)	once/3 months	once/day (8)
IRM Downscale	(1)(2) (13)	once/3 months	once/day (8)
IRM Detector Not in Startup Position	(2) (once operating cycle)	once/operating cycle (12)	N/A
IRM Inoperative	(1)(2) (13)	N/A	N/A
SRM Upscale	(1)(2) (13)	once/3 months	once/day (8)
SRM Downscale	(1)(2) (13)	once/3 months	once/day (8)
SRM Detector Not in Startup Position	(2) (once/operating cycle)	once/operating cycle (12)	N/A
SRM Inoperative	(1)(2) (13)	N/A	N/A
Flow Bias Comparator	(1)(15)	once/operating cycle (20)	N/A
Flow Bias Upscale	(1)(13)	once/3 months	N/A
Rod Block Logic	(16)	N/A	N/A
West Scram Discharge Tank Water Level High (LS-85-45L)	once/quarter	once/operating cycle	N/A
East Scram Discharge Tank Water Level High (LS-85-45M)	once/quarter	once/operating cycle	N/A

THIS PAGE INTENTIONALLY LEFT BLANK

LIMITING CONDITIONS FOR OPERATION

3.3 REACTIVITY CONTROL

Applicability

Applies to the operational status of the control rod system.

Objective

To assure the ability of the control rod system to control reactivity.

Specification

A. Reactivity Limitations

1. Reactivity margin - core loading

A sufficient number of control rods shall be operable so that the core could be made subcritical in the most reactive condition during the operating cycle with the strongest control rod fully withdrawn and all other operable control rods fully inserted.

SURVEILLANCE REQUIREMENTS

4.3 REACTIVITY CONTROL

Applicability

Applies to the surveillance requirements of the control rod system.

Objective

To verify the ability of the control rod system to control reactivity.

Specification

A. Reactivity Limitations

1. Reactivity margin - core loading

Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.38% Δ k/k the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest operable control rod fully withdrawn and all other operable rods fully inserted.

LIMITING CONDITIONS FOR OPERATION

3.3.A.2 Reactivity margin - inoperable control rods

a. Control rod drives which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to the COLD SHUTDOWN CONDITION within 24 hours and shall not be started unless (1) investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.2.c.

SURVEILLANCE REQUIREMENTS

4.3.A.2 Reactivity margin - inoperable control rods

a. Each partially or fully withdrawn OPERABLE control rod shall be exercised one notch at least once each week when operating above the power level cutoff of the RWM. In the event power operation is continuing with three or more inoperable control rods, this test shall be performed at least once each day, when operating above the power level cutoff of the RWM.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.3.A.2 Reactivity margin - inoperable control rods (Cont'd)

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically.

c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

4.3.A.2 Reactivity margin - inoperable control rods (Cont'd)

b. DELETED

c. When it is initially determined that a control rod is incapable of normal insertion a test shall be conducted to demonstrate that the cause of the malfunction is not a failure in the control rod drive mechanism. If this can be demonstrated an attempt to fully insert the control rod shall be made. If the control rod cannot be inserted and an investigation has demonstrated that the cause of failure is not a failed control rod drive mechanism collet housing, a shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined highest worth control rod capable of withdrawal fully withdrawn, and all other control rods capable of insertion fully inserted.

LIMITING CONDITIONS FOR OPERATION

3.3.A.2 Reactivity margin - inoperable control rods (Cont'd)

d. DELETED

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5x5 array may be inoperable (at least 4 OPERABLE control rods must separate any 2 inoperable ones). If this specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a SHUTDOWN CONDITION within 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.A.2 Reactivity margin - inoperable control rods (Cont'd)

d. The control rod accumulators shall be determined OPERABLE at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.B. Control Rods	4.3.B. Control Rods
1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the SHUTDOWN CONDITION when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.	1. The coupling integrity shall be verified for each withdrawn control rod as follows: a. Verify that the control rod is following the drive by observing any response in the nuclear instrumentation each time a rod is moved when the reactor is operating above the preset power level cutoff of the RWM. b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.
2. The control rod drive housing support system shall be in place during REACTOR POWER OPERATION or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.	2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.3.B. Control Rods

3.a DELETED

3.b Whenever the reactor is in the startup or run modes below 10% rated power, the Rod Worth Minimizer (RWM) shall be OPERABLE.

1. Should the RWM become inoperable after the first twelve rods have been withdrawn, the start-up may continue provided that a second licensed operator or other technically qualified member of the plant staff is present at the console verifying compliance with the prescribed control rod program.

2. Should the RWM be inoperable before the first twelve rods are withdrawn, start-up may continue provided a second licensed operator or other technically qualified member of the plant staff is present at the console verifying compliance with the prescribed control rod program. Use of this provision is limited to one plant startup per calendar year.

4.3.B. Control Rods

3.a DELETED

3.b.1 The Rod Worth Minimizer (RWM) shall be demonstrated to be OPERABLE for a reactor startup by the following checks:

a. By demonstrating that the control rod patterns and Banked Position Withdrawal Sequence (or equivalent) input to the RWM computer are correctly loaded following any loading of the program into the computer.

b. Within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical verify proper annunciation of the selection error of at least one out-of-sequence control rod.

c. Within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, the rod block function of the RWM shall be verified by moving an out-of-sequence control rod.

LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

3.b (Cont'd)

3. Should the RWM become inoperable on a shutdown, shutdown may continue provided that a second licensed operator or other technically qualified member of the plant staff is present at the console verifying compliance with the prescribed control rod program.

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

3.b.2 The Rod Worth Minimizer (RWM) shall be demonstrated to be OPERABLE for a reactor shutdown by the following checks:

a. By demonstrating that the control rod patterns and Banked Position Withdrawal Sequence (or equivalent) input to the RWM computer are correctly loaded following any loading of the program into the computer.

b. Within 8 hours prior to RWM automatic initiation when reducing thermal power, verify proper annunciation of the selection error of at least one out-of-sequence control rod.

c. Within one hour after RWM automatic initation when reducing thermal power, the rod block function of the RWM shall be verified by moving an out-of-sequence control rod.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.B. Control Rods	4.3.B. Control Rods
3.c. If Specifications 3.3.B.3.b.1 through 3.3.B.3.b.3 cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 10% rated power, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the shutdown position.	3.b.3 When the RWM is not OPERABLE a second licensed operator or other technically qualified member of the plant staff shall verify that the correct rod program is followed.
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.	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. 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.	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.

LIMITING CONDITIONS FOR OPERATION

3.3.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	3.500

SURVEILLANCE REQUIREMENTS

4.3.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 800 psig. This testing shall be completed prior to exceeding 40% power. Below 10% power, only rods in those sequences which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram-time tested.

THIS PAGE INTENTIONALLY LEFT BLANK

A. Reactivity Limitation

1. The requirements for the control rod drive system have been identified by evaluating the need for reactivity control via control rod movement over the full spectrum of plant conditions and events. As discussed in subsection 3.4 of the Final Safety Analysis Report, the control rod system design is intended to provide sufficient control of core reactivity that the core could be made subcritical with the strongest rod fully withdrawn. This reactivity characteristic has been a basic assumption in the analysis of plant performance. Compliance with this requirement can be demonstrated conveniently only at the time of initial fuel loading or refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration shall be performed with the reactor core in the cold, xenon-free condition and will show that the reactor is subcritical by at least R + 0.38 percent Δk with the analytically determined strongest control rod fully withdrawn.

The value of "R", in units of percent Δk, is the amount by which the core reactivity, in the most reactive condition at any time in the subsequent operating cycle, is calculated to be greater than at the time of the demonstration. "R", therefore, is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of "R" must be positive or zero and must be determined for each fuel cycle.

The demonstration is performed with a control rod which is calculated to be the strongest rod. In determining this "analytically strongest" rod, it is assumed that every fuel assembly of the same type has identical material properties. In the actual core, however, the control cell material properties vary within allowed manufacturing tolerances, and the strongest rod is determined by a combination of the control cell geometry and local k_{∞}. Therefore, an additional margin is included in the shutdown margin test to account for the fact that the rod used for the demonstration (the "analytically strongest") is not necessarily the strongest rod in the core. Studies have been made which compare experimental criticals with calculated criticals. These studies have shown that actual criticals can be predicted within a given tolerance band. For gadolinia cores the additional margin required due to control cell material manufacturing tolerances and calculational uncertainties has experimentally been determined to be 0.38 percent Δk. When this additional margin is demonstrated, it assures that the reactivity control requirement is met.

2. Reactivity margin - inoperable control rods - Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If the rod is fully inserted and disarmed electrically*, it is in a safe position of maximum contribution to shutdown reactivity. If it is disarmed electrically in a nonfully inserted position, that position shall be consistent with the shutdown reactivity limitations stated in Specification 3.3.A.1. This assures that the core can be shut down at all times with the remaining control rods assuming the strongest OPERABLE control rod does not insert. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress-assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed rod after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings. The Rod Worth Minimizer is not automatically bypassed until reactor power is above the preset power level cutoff. Therefore, control rod movement is restricted and the single notch exercise surveillance test is only performed above this power level. The Rod Worth Minimizer prevents movement of out-of-sequence rods unless power is above the preset power level cutoff.

B. Control Rods

1. Control rod dropout accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement could indicate an uncoupled condition. Rod position indication is required for proper function of the Rod Worth Minimizer.

* To disarm the drive electrically, four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred because, in this condition, drive water cools and minimizes crud accumulation in the drive. Electrical disarming does not eliminate position indication.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in subsection 3.5.2 of the FSAR and the safety evaluation is given in subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) restricts withdrawals and insertions of control rods to prespecified 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. Reference Sections 3.6.6, 7.16.5.3, and 14.6.2 of the FSAR, and NEDE-24011-P-A, Amendment 17.

   In performing the function described above, the RWM is not required to impose any restrictions at core power levels in excess of 10 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 10 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 10 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 constrains the control rod sequences and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer provides 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. Reference Section 7.16.5.3 of the FSAR. The RWM functions as a backup to procedural control of control rod sequences, which limit the maximum reactivity worth of control rods. When the Rod Worth Minimizer is out of service, special criteria allow a second licensed operator or other technically qualified member of the plant staff to manually fulfill the control rod pattern conformance functions of this system. The requirement that the RWM be OPERABLE for the withdrawal of the first twelve rods on a startup is to ensure that a high degree of RWM availability is maintained.

The functions of the RWM 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 10 percent, the RWM forces adherence to acceptable (Banked Position Withdrawal Sequence or equivalent) rod patterns. Above 10 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 10 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.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One OPERABLE SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two OPERABLE SRMs are provided as an added conservatism.

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10 percent 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 BWRs 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 which are assumed to scram on high neutron flux, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of control rod motion.

This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from sensor and circuit delays after which the pilot scram solenoid deenergizes to 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds, rather than 120 milliseconds, are conservatively assumed for this time interval in the transient analyses and are also included in the allowable scram insertion times of Specification 3.3.C.

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 percent ΔK. Deviations in core reactivity greater than 1 percent 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.

E. No BASES provided for this specification

F. Scram Discharge Volume

The nominal stroke time for the scram discharge volume vent and drain valves is $\leq$ 30 seconds following a scram. The purpose of these valves is to limit the quantity of reactor water discharged after a scram and no direct safety function is performed. The surveillance for the valves assures that system drainage is not impeded by a valve which fails to open and that the valves are operable and capable of closing upon a scram.

References

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 212
License No. DPR-52

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

A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 20, 1992, as supplemented March 18, 1993, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 212, 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 its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: April 30, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 212

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

UNIT 2
EFFECTIVE PAGE LIST

REMOVE	INSERT
i	i
ii	ii*
1.0-3	1.0-3*
1.0-4	1.0-4
1.1/2.1-12	1.1/2.1-12*
1.1/2.1-13	1.1/2.1-13
3.2/4.2-25	3.2/4.2-25
3.2/4.2-25a	3.2/4.2-25a
3.2/4.2-50	3.2/4.2-50
---	3.2/4.2-50a
3.3/4.3-1	3.3/4.3-1*
3.3/4.3-2	3.3/4.3-2
3.3/4.3-3	3.3/4.3-3
3.3/4.3-4	3.3/4.3-4
3.3/4.3-5	3.3/4.3-5
3.3/4.3-6	3.3/4.3-6
3.3/4.3-7	3.3/4.3-7
3.3/4.3-8	3.3/4.3-8
3.3/4.3-9	3.3/4.3-9
3.3/4.3-10	3.3/4.3-10
3.3/4.3-13	3.3/4.3-13*
3.3/4.3-14	3.3/4.3-14
3.3/4.3-15	3.3/4.3-15
3.3/4.3-16	3.3/4.3-16
3.3/4.3-19	3.3/4.3-19
3.3/4.3-20	3.3/4.3-20*

*DENOTES OVERLEAF PAGES

TABLE OF CONTENTS

Section		Page No.
1.0	Definitions	1.0-1
	SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS	
1.1/2.1	Fuel Cladding Integrity	1.1/2.1-1
1.2/2.2	Reactor Coolant System Integrity.	1.2/2.2-1
	LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS	
3.1/4.1	Reactor Protection System..	3.1/4.1-1
3.2/4.2	Protective Instrumentation.	3.2/4.2-1
	A. Primary Containment and Reactor Building Isolation Functions.	3.2/4.2-1
	B. Core and Containment Cooling Systems - Initiation and Control	3.2/4.2-1
	C. Control Rod Block Actuation.	3.2/4.2-2
	D. Radioactive Liquid Effluent Monitoring Instrumentation.	3.2/4.2-3
	E. Drywell Leak Detection	3.2/4.2-4
	F. Surveillance Instrumentation	3.2/4.2-4
	G. Control Room Isolation	3.2/4.2-4
	H. Flood Protection	3.2/4.2-4
	I. Meteorological Monitoring Instrumentation. . .	3.2/4.2-4
	J. Seismic Monitoring Instrumentation	3.2/4.2-5
	K. Radioactive Gaseous Effluent Monitoring Instrumentation	3.2/4.2-6
	L. ATWS Recirculation Pump Trip	3.2/4.2-6a
3.3/4.3	Reactivity Control.	3.3/4.3-1
	A. Reactivity Limitations	3.3/4.3-1
	B. Control Rods	3.3/4.3-5
	C. Scram Insertion Times.	3.3/4.3-9

Section		Page No.
	D. Reactivity Anomalies	3.3/4.3-11
	E. Reactivity Control	3.3/4.3-12
	F. Scram Discharge Volume	3.3/4.3-12
3.4/4.4	Standby Liquid Control System	3.4/4.4-1
	A. Normal System Availability	3.4/4.4-1
	B. Operation with Inoperable Components	3.4/4.4-3
	C. Sodium Pentaborate Solution.	3.4/4.4-3
3.5/4.5	Core and Containment Cooling Systems.	3.5/4.5-1
	A. Core Spray System (CSS).	3.5/4.5-1
	B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)	3.5/4.5-4
	C. RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS). . .	3.5/4.5-9
	D. Equipment Area Coolers	3.5/4.5-13
	E. High Pressure Coolant Injection System (HPCIS).	3.5/4.5-13
	F. Reactor Core Isolation Cooling System (RCICS).	3.5/4.5-14
	G. Automatic Depressurization System (ADS). . . .	3.5/4.5-16
	H. Maintenance of Filled Discharge Pipe	3.5/4.5-17
	I. Average Planar Linear Heat Generation Rate . .	3.5/4.5-18
	J. Linear Heat Generation Rate (LHGR)	3.5/4.5-18
	K. Minimum Critical Power Ratio (MCPR).	3.5/4.5-19
	L. APRM Setpoints	3.5/4.5-20
	M. Core Thermal-Hydraulic Stability	3.5/4.5-20
3.6/4.6	Primary System Boundary	3.6/4.6-1
	A. Thermal and Pressurization Limitations	3.6/4.6-1
	B. Coolant Chemistry.	3.6/4.6-5
	C. Coolant Leakage.	3.6/4.6-9
	D. Relief Valves.	3.6/4.6-10

H. Reactor Power Operation - Reactor power operation is any operation in the STARTUP/HOT STANDBY or RUN MODE with the reactor critical and above 1 percent rated power.

I. STARTUP CONDITION - The reactor is in the STARTUP CONDITION when the withdrawal of control rods for the purpose of making the reactor critical has begun, reactor power is less than or equal to 1 percent of rated, and the reactor is in the STARTUP/HOT STANDBY MODE.

J. HOT STANDBY CONDITION - The reactor is in the HOT STANDBY CONDITION when reactor power is less than or equal to 1 percent of rated. The reactor is in the STARTUP/HOT STANDBY MODE, and the reactor is not in the STARTUP CONDITION. The reactor coolant temperature may be greater than 212° F.

Note that a HOT STANDBY CONDITION cannot exist simultaneously with a STARTUP CONDITION due to the difference in intent. A HOT STANDBY CONDITION exists when the reactor mode switch is placed in the STARTUP/HOT STANDBY position (for example, to comply with an LCO) and power level has been reduced to 1 percent or lower. Anytime control rods are being withdrawn for the purpose of increasing reactor power level, the reactor mode switch has been placed in the STARTUP/HOT STANDBY position, and reactor power level is at or below one percent, a STARTUP CONDITION exists.

K. SHUTDOWN CONDITION - The reactor is in the SHUTDOWN CONDITION when the reactor is in the Shutdown or Refuel Mode.

1. HOT SHUTDOWN CONDITION - The reactor is in the HOT SHUTDOWN CONDITION when reactor coolant temperature is greater than 212° F and the reactor is in the SHUTDOWN CONDITION.

2. COLD SHUTDOWN CONDITION - The reactor is in the COLD SHUTDOWN CONDITION when reactor coolant temperature is equal to or less than 212° F and the reactor is in the SHUTDOWN CONDITION.

L. COLD CONDITION - The reactor is in the COLD CONDITION when reactor coolant temperature is equal to or less than 212° F in any Mode of Operation (except as defined in K.2 above).

M. Mode of Operation - The reactor mode switch position determines the Mode of Operation of the reactor when there is fuel in the reactor vessel, except that the Mode of Operation may remain unchanged when the reactor mode switch is temporarily moved to another position as permitted by the notes. When there is no fuel in the reactor vessel, the reactor is considered not to be in any Mode of Operation or operational condition. The reactor mode switch may then be in any position or may be inoperable.

1. Startup/Hot Standby Mode - The reactor is in the STARTUP/HOT STANDBY MODE when the reactor mode switch is in the "STARTUP/HOT STANDBY" position. This is often referred to as just the STARTUP MODE.

2. Run Mode - The reactor is in the Run Mode when the reactor mode switch is in the "Run" position.

3. Shutdown Mode - The reactor is in the Shutdown Mode when the reactor mode switch is in the "Shutdown" position.(1)(2)(3)(4)

4. Refuel Mode - The reactor is in the Refuel Mode when the reactor mode switch is in the "Refuel" position.(1)

---

(1) The reactor mode switch may be placed in any position to perform required tests or maintenance authorized by the shift operations supervisor, provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

(2) The reactor mode switch may be placed in the "Refuel" position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.10.A.5 provided that reactor coolant temperature is equal to or less than 212° F.

(3) The reactor mode switch may be placed in the "Refuel" position while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock is OPERABLE.

(4) The reactor mode switch may be placed in the "Startup/Hot Standby" position and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

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 3,440 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 setpoints are discussed below:

A. Neutron Flux Scram

1. APRM Flow-Biased 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 core average neutron flux.

During power increase transients, the instantaneous fuel surface heat flux is less than the instantaneous neutron flux by an amount depending upon the duration of the transient and the fuel time constant. For this reason, the flow-biased scram APRM flux signal is passed through a filtering network with a time constant which is representative of the fuel time constant. As a result of this filtering, APRM flow-biased scram will occur only if the neutron flux signal is in excess of the setpoint and of sufficient time duration to overcome the fuel time constant and result in an average fuel surface heat flux which is equivalent to the neutron flux trip setpoint. This setpoint is variable up to 120 percent of rated power based on recirculation drive flow according to the equations given in Section 2.1.A.1 and the graph in Figure 2.1-2. For the purpose of licensing transient analysis, neutron flux scram is assumed to occur at 120 percent of rated power. Therefore, the flow biased scram provides additional margin to the thermal limits for slow transients such as loss of feedwater heating. No safety credit is taken for flow-biased scrams.

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. Thus, of all 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 five 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 eight chambers, four in each of the reactor protection system logic channels. The IRM is a five-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The five decades are covered by the IRM by means of a range switch and the five decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be at 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions on that range.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5)	Function	Trip Level Setting
4(1)	APRM Upscale (Flow Bias)	≤0.58W + 50% (2)
4(1)	APRM Upscale (Startup Mode) (8)	≤12%
4(1)	APRM Downscale (9)	≥3%
4(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (Flow Bias)	≤0.66W + 40% (2)(13)
2(7)	RBM Downscale (9)	≥3%
2(7)	RBM Inoperative	(10c)
6(1)	IRM Upscale (8)	≤108/125 of full scale
6(1)	IRM Downscale (3)(8)	≥5/125 of full scale
6(1)	IRM Detector not in Startup Position (8)	(11)
6(1)	IRM Inoperative (8)	(10a)
3(1) (6)	SRM Upscale (8)	≤ $1X10^5$ counts/sec.
3(1) (6)	SRM Downscale (4)(8)	≥3 counts/sec.
3(1) (6)	SRM Detector not in Startup Position (4)(8)	(11)
3(1) (6)	SRM Inoperative (8)	(10a)
2(1)	Flow Bias Comparator	≤10% difference in recirculation flows
2(1)	Flow Bias Upscale	≤115% recirculation flow
1	Rod Block Logic	N/A
1(12)	High Water Level in West Scram Discharge Tank (LS-85-45L)	≤25 gal.
1(12)	High Water Level in East Scram Discharge Tank (LS-85-45M)	≤25 gal.

THIS PAGE INTENTIONALLY LEFT BLANK

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

Function	Functional Test	Calibration (17)	Instrument Check
APRM Upscale (Flow Bias)	(1) (13)	once/3 months	once/day (8)
APRM Upscale (Startup Mode)	(1) (13)	once/3 months	once/day (8)
APRM Downscale	(1) (13)	once/3 months	once/day (8)
APRM Inoperative	(1) (13)	N/A	once/day (8)
RBM Upscale (Flow Bias)	(1) (13)	once/6 months	once/day (8)
RBM Downscale	(1) (13)	once/6 months	once/day (8)
RBM Inoperative	(1) (13)	N/A	once/day (8)
IRM Upscale	(1)(2) (13)	once/3 months	once/day (8)
IRM Downscale	(1)(2) (13)	once/3 months	once/day (8)
IRM Detector Not in Startup Position	(2) (once operating cycle)	once/operating cycle (12)	N/A
IRM Inoperative	(1)(2) (13)	N/A	N/A
SRM Upscale	(1)(2) (13)	once/3 months	once/day (8)
SRM Downscale	(1)(2) (13)	once/3 months	once/day (8)
SRM Detector Not in Startup Position	(2) (once/operating cycle)	once/operating cycle (12)	N/A
SRM Inoperative	(1)(2) (13)	N/A	N/A
Flow Bias Comparator	(1)(15)	once/operating cycle (20)	N/A
Flow Bias Upscale	(1)(15)	once/3 months	N/A
Rod Block Logic	(16)	N/A	N/A
West Scram Discharge Tank Water Level High (LS-85-45L)	once/quarter	once/18 months	N/A
East Scram Discharge Tank Water Level High (LS-85-45M)	once/quarter	once/18 months	N/A

THIS PAGE INTENTIONALLY LEFT BLANK

LIMITING CONDITIONS FOR OPERATION

3.3 REACTIVITY CONTROL

Applicability

Applies to the operational status of the control rod system.

Objective

To assure the ability of the control rod system to control reactivity.

Specification

A. Reactivity Limitations

1. Reactivity Margin - Core Loading

A sufficient number of control rods shall be OPERABLE so that the core could be made subcritical in the most reactive condition during the operating cycle with the strongest control rod fully withdrawn and all other OPERABLE control rods fully inserted.

SURVEILLANCE REQUIREMENTS

4.3 REACTIVITY CONTROL

Applicability

Applies to the surveillance requirements of the control rod system.

Objective

To verify the ability of the control rod system to control reactivity.

Specification

A. Reactivity Limitations

1. Reactivity Margin - Core Loading

Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.38% Δ k/k the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest OPERABLE control rod fully withdrawn and all other OPERABLE rods fully inserted.

LIMITING CONDITIONS FOR OPERATION

3.3.A.2 Reactivity Margin - Inoperable Control Rods

a. Control rod drives which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to the COLD SHUTDOWN CONDITION within 24 hours and shall not be started unless (1) investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.2.c.

SURVEILLANCE REQUIREMENTS

4.3.A.2 Reactivity Margin - Inoperable Control Rods

a. Each partially or fully withdrawn OPERABLE control rod shall be exercised one notch at least once each week when operating above the power level cutoff of the RWM. In the event power operation is continuing with three or more inoperable control rods, this test shall be performed at least once each day, when operating above the power level cutoff of the RWM.

LIMITING CONDITIONS FOR OPERATION

3.3.A.2 Reactivity Margin - Inoperable Control Rods (Cont'd)

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically.

c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

SURVEILLANCE REQUIREMENTS

4.3.A.2 Reactivity Margin - Inoperable Control Rods (Cont'd)

b. DELETED

c. When it is initially determined that a control rod is incapable of normal insertion a test shall be conducted to demonstrate that the cause of the malfunction is not a failure in the control rod drive mechanism. If this can be demonstrated an attempt to fully insert the control rod shall be made. If the control rod cannot be inserted and an investigation has demonstrated that the cause of failure is not a failed control rod drive mechanism collet housing, a shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined highest worth control rod capable of withdrawal fully withdrawn, and all other control rods capable of insertion fully inserted.

LIMITING CONDITIONS FOR OPERATION

3.3.A.2 Reactivity Margin - Inoperable Control Rods (Cont'd)

d. DELETED

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5x5 array may be inoperable (at least 4 OPERABLE control rods must separate any 2 inoperable ones). If this specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a SHUTDOWN CONDITION within 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.A.2 Reactivity Margin - Inoperable Control Rods (Cont'd)

d. The control rod accumulators shall be determined OPERABLE at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the SHUTDOWN CONDITION when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

2. The control rod drive housing support system shall be in place during REACTOR POWER OPERATION or when the reactor coolant system is pressurized above atmospheric with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

   a. Verify that the control rod is following the drive by observing any response in the nuclear instrumentation each time a rod is moved when the reactor is operating above the preset power level cutoff of the RWM.

   b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

3.a DELETED

3.b Whenever the reactor is in the startup or run modes below 10% rated power, the Rod Worth Minimizer (RWM) shall be OPERABLE.

1. Should the RWM become inoperable after the first twelve rods have been withdrawn, the start-up may continue provided that a second licensed operator or other technically qualified member of the plant staff is present at the console verifying compliance with the prescribed control rod program.

2. Should the RWM be inoperable before the first twelve rods are withdrawn, start-up may continue provided a second licensed operator or other technically qualified member of the plant staff is present at the console verifying compliance with the prescribed control rod program. Use of this provision is limited to one plant startup per calendar year.

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

3.a DELETED

3.b.1 The Rod Worth Minimizer (RWM) shall be demonstrated to be OPERABLE for a reactor startup by the following checks:

a. By demonstrating that the control rod patterns and Banked Position Withdrawal Sequence (or equivalent) input to the RWM computer are correctly loaded following any loading of the program into the computer.

b. Within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical verify proper annunciation of the selection error of at least one out-of-sequence control rod.

c. Within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, the rod block function of the RWM shall be verified by moving an out-of-sequence control rod.

LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

3.b (Cont'd)

3. Should the RWM become inoperable on a shutdown, shutdown may continue provided that a second licensed operator or other technically qualified member of the plant staff is present at the console verifying compliance with the prescribed control rod program.

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

3.b.2 The Rod Worth Minimizer (RWM) shall be demonstrated to be OPERABLE for a reactor shutdown by the following checks:

a. By demonstrating that the control rod patterns and Banked Position Withdrawal Sequence (or equivalent) input to the RWM computer are correctly loaded following any loading of the program into the computer.

b. Within 8 hours prior to RWM automatic initiation when reducing thermal power, verify proper annunciation of the selection error of at least one out-of-sequence control rod.

c. Within one hour after RWM automatic initation when reducing thermal power, the rod block function of the RWM shall be verified by moving an out-of-sequence control rod.

LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

3.c. If Specifications 3.3.B.3.b.1 through 3.3.B.3.b.3 cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 10% rated power, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the shutdown position.

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.

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

3.b.3 When the RWM is not OPERABLE a second licensed operator or other technically qualified member of the plant staff shall verify that the correct rod program is followed.

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.

LIMITING CONDITIONS FOR OPERATION

3.3.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	3.500

SURVEILLANCE REQUIREMENTS

4.3.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 800 psig. This testing shall be completed prior to exceeding 40% power. Below 10% power, only rods in those sequences which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram-time tested.

THIS PAGE INTENTIONALLY LEFT BLANK

A. Reactivity Limitation

1. The requirements for the control rod drive system have been identified by evaluating the need for reactivity control via control rod movement over the full spectrum of plant conditions and events. As discussed in subsection 3.4 of the Final Safety Analysis Report, the control rod system design is intended to provide sufficient control of core reactivity that the core could be made subcritical with the strongest rod fully withdrawn. This reactivity characteristic has been a basic assumption in the analysis of plant performance. Compliance with this requirement can be demonstrated conveniently only at the time of initial fuel loading or refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration shall be performed with the reactor core in the cold, xenon-free condition and will show that the reactor is subcritical by at least R + 0.38 percent Δk with the analytically determined strongest control rod fully withdrawn.

   The value of "R", in units of percent Δk, is the amount by which the core reactivity, in the most reactive condition at any time in the subsequent operating cycle, is calculated to be greater than at the time of the demonstration. "R", therefore, is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of "R" must be positive or zero and must be determined for each fuel cycle.

   The demonstration is performed with a control rod which is calculated to be the strongest rod. In determining this "analytically strongest" rod, it is assumed that every fuel assembly of the same type has identical material properties. In the actual core, however, the control cell material properties vary within allowed manufacturing tolerances, and the strongest rod is determined by a combination of the control cell geometry and local k_∞. Therefore, an additional margin is included in the shutdown margin test to account for the fact that the rod used for the demonstration (the "analytically strongest") is not necessarily the strongest rod in the core. Studies have been made which compare experimental criticals with calculated criticals. These studies have shown that actual criticals can be predicted within a given tolerance band. For gadolinia cores the additional margin required due to control cell material manufacturing tolerances and calculational uncertainties has experimentally been determined to be 0.38 percent Δk. When this additional margin is demonstrated, it assures that the reactivity control requirement is met.

2. Reactivity Margin - Inoperable Control Rods - Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If the rod is fully inserted and disarmed electrically*, it is in a safe position of maximum contribution to shutdown reactivity. If it is disarmed electrically in a nonfully inserted position, that position shall be consistent with the shutdown reactivity limitations stated in Specification 3.3.A.1. This assures that the core can be shut down at all times with the remaining control rods assuming the strongest OPERABLE control rod does not insert. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress-assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed rod after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings. The Rod Worth Minimizer is not automatically bypassed until reactor power is above the preset power level cutoff. Therefore, control rod movement is restricted and the single notch exercise surveillance test is only performed above this power level. The Rod Worth Minimizer prevents movement of out-of-sequence rods unless power is above the preset power level cutoff.

B. Control Rods

1. Control rod dropout accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement could indicate an uncoupled condition. Rod position indication is required for proper function of the Rod Worth Minimizer.

* To disarm the drive electrically, four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred because, in this condition, drive water cools and minimizes crud accumulation in the drive. Electrical disarming does not eliminate position indication.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in subsection 3.5.2 of the FSAR and the safety evaluation is given in subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) restricts withdrawals and insertions of control rods to prespecified 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. Reference Sections 3.6.6, 7.16.5.3, and 14.6.2 of the FSAR, and NEDE-24011-P-A, Amendment 17.

   In performing the function described above, the RWM is not required to impose any restrictions at core power levels in excess of 10 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 10 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 10 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 constrains the control rod sequences and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer provides 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. Reference Section 7.16.5.3 of the FSAR. The RWM functions as a backup to procedural control of control rod sequences, which limit the maximum reactivity worth of control rods. When the Rod Worth Minimizer is out of service, special criteria allow a second licensed operator or other technically qualified member of the plant staff to manually fulfill the control rod pattern conformance functions of this system. The requirement that the RWM be OPERABLE for the withdrawal of the first twelve rods on a startup is to ensure that a high degree of RWM availability is maintained.

The functions of the RWM 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 10 percent, the RWM forces adherence to acceptable (Banked Position Withdrawal Sequence or equivalent) rod patterns. Above 10 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 10 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.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One OPERABLE SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two OPERABLE SRMs are provided as an added conservatism.

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10 percent 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 BWRs 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 which are assumed to scram on high neutron flux, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of control rod motion.

This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from sensor and circuit delays after which the pilot scram solenoid deenergizes to 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds, rather than 120 milliseconds, are conservatively assumed for this time interval in the transient analyses and are also included in the allowable scram insertion times of Specification 3.3.C.

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 percent ΔK. Deviations in core reactivity greater than 1 percent 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.

E. No BASES provided for this specification

F. Scram Discharge Volume

The nominal stroke time for the scram discharge volume vent and drain valves is $\leq$ 30 seconds following a scram. The purpose of these valves is to limit the quantity of reactor water discharged after a scram and no direct safety function is performed. The surveillance for the valves assures that system drainage is not impeded by a valve which fails to open and that the valves are OPERABLE and capable of closing upon a scram.

References

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

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. 169
License No. DPR-68

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

A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 20, 1992, as supplemented March 18, 1993, 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. 169 , 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 its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: April 30, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. ____

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

UNIT 3
EFFECTIVE PAGE LIST

REMOVE	INSERT
i	i
ii	ii*
1.0-3	1.0-3*
1.0-4	1.0-4
1.1/2.1-12	1.1/2.1-12*
1.1/2.1-13	1.1/2.1-13
3.2/4.2-24	3.2/4.2-24
---	3.2/4.2-24a
3.2/4.2-49	3.2/4.2-49
---	3.2/4.2-49a
3.3/4.3-1	3.3/4.3-1*
3.3/4.3-2	3.3/4.3-2
3.3/4.3-3	3.3/4.3-3*
3.3/4.3-4	3.3/4.3-4
3.3/4.3-5	3.3/4.3-5
3.3/4.3-6	3.3/4.3-6
3.3/4.3-7	3.3/4.3-7
3.3/4.3-8	3.3/4.3-8
3.3/4.3-9	3.3/4.3-9
3.3/4.3-10	3.3/4.3-10
3.3/4.3-13	3.3/4.3-13*
3.3/4.3-14	3.3/4.3-14
3.3/4.3-15	3.3/4.3-15
3.3/4.3-16	3.3/4.3-16
3.3/4.3-19	3.3/4.3-19
3.3/4.3-20	3.3/4.3-20*

*DENOTES OVERLEAF PAGES

TABLE OF CONTENTS

Section		Page No.
1.0	Definitions	1.0-1
	SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS	
1.1/2.1	Fuel Cladding Integrity	1.1/2.1-1
1.2/2.2	Reactor Coolant System Integrity	1.2/2.2-1
	LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS	
3.1/4.1	Reactor Protection System	3.1/4.1-1
3.2/4.2	Protective Instrumentation	3.2/4.2-1
	A. Primary Containment and Reactor Building Isolation Functions	3.2/4.2-1
	B. Core and Containment Cooling Systems - Initiation and Control	3.2/4.2-1
	C. Control Rod Block Actuation	3.2/4.2-2
	D. Radioactive Liquid Effluent Monitoring Instrumentation	3.2/4.2-3
	E. Drywell Leak Detection	3.2/4.2-4
	F. Surveillance Instrumentation	3.2/4.2-4
	G. Control Room Isolation	3.2/4.2-4
	H. Flood Protection	3.2/4.2-4
	I. Meteorological Monitoring Instrumentation	3.2/4.2-4
	J. Seismic Monitoring Instrumentation	3.2/4.2-5
	K. Radioactive Gaseous Effluent Monitoring Instrumentation	3.2/4.2-6
	L. ATWS Recirculation Pump Trip	3.2/4.2-6a
3.3/4.3	Reactivity Control	3.3/4.3-1
	A. Reactivity Limitations	3.3/4.3-1
	B. Control Rods	3.3/4.3-5
	C. Scram Insertion Times	3.3/4.3-9

Section		Page No.
	D. Reactivity Anomalies	3.3/4.3-11
	E. Reactivity Control	3.3/4.3-12
	F. Scram Discharge Volume	3.3/4.3-12
3.4/4.4	Standby Liquid Control System	3.4/4.4-1
	A. Normal System Availability	3.4/4.4-1
	B. Operation with Inoperable Components	3.4/4.4-3
	C. Sodium Pentaborate Solution.	3.4/4.4-3
3.5/4.5	Core and Containment Cooling Systems.	3.5/4.5-1
	A. Core Spray System (CSS).	3.5/4.5-1
	B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)	3.5/4.5-4
	C. RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS). . .	3.5/4.5-9
	D. Equipment Area Coolers	3.5/4.5-13
	E. High Pressure Coolant Injection System (HPCIS).	3.5/4.5-13
	F. Reactor Core Isolation Cooling System (RCICS).	3.5/4.5-14
	G. Automatic Depressurization System (ADS). . . .	3.5/4.5-16
	H. Maintenance of Filled Discharge Pipe	3.5/4.5-17
	I. Average Planar Linear Heat Generation Rate . .	3.5/4.5-18
	J. Linear Heat Generation Rate (LHGR)	3.5/4.5-18
	K. Minimum Critical Power Ratio (MCPR).	3.5/4.5-19
	L. APRM Setpoints	3.5/4.5-20
3.6/4.6	Primary System Boundary	3.6/4.6-1
	A. Thermal and Pressurization Limitations	3.6/4.6-1
	B. Coolant Chemistry.	3.6/4.6-5
	C. Coolant Leakage.	3.6/4.6-9
	D. Relief Valves.	3.6/4.6-10
	E. Jet Pumps.	3.6/4.6-11

1.0 DEFINITIONS (Cont'd)

H. Reactor Power Operation - Reactor power operation is any operation in the STARTUP/HOT STANDBY or RUN MODE with the reactor critical and above 1 percent rated power.

I. STARTUP CONDITION - The reactor is in the STARTUP CONDITION when the withdrawal of control rods for the purpose of making the reactor critical has begun, reactor power is less than or equal to 1 percent of rated, and the reactor is in the STARTUP/HOT STANDBY MODE.

J. HOT STANDBY CONDITION - The reactor is in the HOT STANDBY CONDITION when reactor power is less than or equal to 1 percent of rated. The reactor is in the STARTUP/HOT STANDBY MODE, and the reactor is not in the STARTUP CONDITION. The reactor coolant temperature may be greater than 212° F.

Note that a HOT STANDBY CONDITION cannot exist simultaneously with a STARTUP CONDITION due to the difference in intent. A HOT STANDBY CONDITION exists when the reactor mode switch is placed in the STARTUP/HOT STANDBY position (for example, to comply with an LCO) and power level has been reduced to 1 percent or lower. Anytime control rods are being withdrawn for the purpose of increasing reactor power level, the reactor mode switch has been placed in the STARTUP/HOT STANDBY position, and reactor power level is at or below one percent, a STARTUP CONDITION exists.

K. SHUTDOWN CONDITION - The reactor is in the SHUTDOWN CONDITION when the reactor is in the Shutdown or Refuel Mode.

1. HOT SHUTDOWN CONDITION - The reactor is in the HOT SHUTDOWN CONDITION when reactor coolant temperature is greater than 212° F and the reactor is in the SHUTDOWN CONDITION.

2. COLD SHUTDOWN CONDITION - The reactor is in the COLD SHUTDOWN CONDITION when reactor coolant temperature is equal to or less than 212° F and the reactor is in the SHUTDOWN CONDITION.

L. COLD CONDITION - The reactor is in the COLD CONDITION when reactor coolant temperature is equal to or less than 212° F in any Mode of Operation (except as defined in K.2 above).

1.0 DEFINITIONS (Cont'd)

M. Mode of Operation - The reactor mode switch position determines the Mode of Operation of the reactor when there is fuel in the reactor vessel, except that the Mode of Operation may remain unchanged when the reactor mode switch is temporarily moved to another position as permitted by the notes. When there is no fuel in the reactor vessel, the reactor is considered not to be in any Mode of Operation or operational condition. The reactor mode switch may then be in any position or may be inoperable.

1. Startup/Hot Standby Mode - The reactor is in the STARTUP/HOT STANDBY MODE when the reactor mode switch is in the "STARTUP/HOT STANDBY" position. This is often referred to as just the STARTUP MODE.

2. Run Mode - The reactor is in the Run Mode when the reactor mode switch is in the "Run" position.

3. Shutdown Mode - The reactor is in the Shutdown Mode when the reactor mode switch is in the "Shutdown" position.[1][2][3][4]

4. Refuel Mode - The reactor is in the Refuel Mode when the reactor mode switch is in the "Refuel" position.[1]

---

(1) The reactor mode switch may be placed in any position to perform required tests or maintenance authorized by the shift operations supervisor, provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

(2) The reactor mode switch may be placed in the "Refuel" position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.10.A.5 provided that reactor coolant temperature is equal to or less than 212° F.

(3) The reactor mode switch may be placed in the "Refuel" position while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock is OPERABLE.

(4) The reactor mode switch may be placed in the "Startup/Hot Standby" position and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

2.1 BASES (Cont'd)

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 3,440 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 setpoints are discussed below:

A. Neutron Flux Scram

1. APRM Flow-Biased 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 core average neutron flux.

During transients, the instantaneous fuel surface heat flux is less than the instantaneous neutron flux by an amount depending upon the duration of the transient and the fuel time constant. For this reason, the flow-biased scram APRM flux signal is passed through a filtering network with a time constant which is representative of the fuel time constant. As a result of this filtering, APRM flow-biased scram will occur only if the neutron flux signal is in excess of the setpoint and of sufficient time duration to overcome the fuel time constant and result in an average fuel surface heat flux which is equivalent to the neutron flux trip setpoint. This setpoint is variable up to 120 percent of rated power based on recirculation drive flow according to the equations given in Section 2.1.A.1 and the graph in Figure 2.1-2. For the purpose of licensing transient analysis, neutron flux scram is assumed to occur at 120 percent of rated power. Therefore, the flow biased provides additional margin to the thermal limits for slow transients such as loss of feedwater heating. No safety credit is taken for flow-biased scrams.

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

2. APRM Flux Scram Trip Setting (REFUEL or START & HOT STANDBY MODE)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all 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 example, if the instrument was on range 1, the scram setting would be 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions on that range.

***See Section 3.5.K

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5)	Function	Trip Level Setting
4(1)	APRM Upscale (Flow Bias)	≤0.66W + 42% (2)
4(1)	APRM Upscale (Startup Mode) (8)	≤12%
4(1)	APRM Downscale (9)	≥3%
4(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (Flow Bias)	≤0.66W + 40% (2)(13)
2(7)	RBM Downscale (9)	≥3%
2(7)	RBM Inoperative	(10c)
6(1)	IRM Upscale (8)	≤108/125 of full scale
6(1)	IRM Downscale (3)(8)	≥5/125 of full scale
6(1)	IRM Detector not in Startup Position (8)	(11)
6(1)	IRM Inoperative (8)	(10a)
3(1) (6)	SRM Upscale (8)	≤ $1X10^5$ counts/sec.
3(1) (6)	SRM Downscale (4)(8)	≥3 counts/sec.
3(1) (6)	SRM Detector not in Startup Position (4)(8)	(11)
3(1) (6)	SRM Inoperative (8)	(10a)
2(1)	Flow Bias Comparator	≤10% difference in recirculation flows
2(1)	Flow Bias Upscale	≤115% recirculation flow
1	Rod Block Logic	N/A
1(12)	High Water Level in West Scram Discharge Tank (LS-85-45L)	≤25 gal.
1(12)	High Water Level in East Scram Discharge Tank (LS-85-45M)	≤25 gal.

THIS PAGE INTENTIONALLY LEFT BLANK

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

Function	Functional Test	Calibration (17)	Instrument Check
APRM Upscale (Flow Bias)	(1) (13)	once/3 months	once/day (8)
APRM Upscale (Startup Mode)	(1) (13)	once/3 months	once/day (8)
APRM Downscale	(1) (13)	once/3 months	once/day (8)
APRM Inoperative	(1) (13)	N/A	once/day (8)
RBM Upscale (Flow Bias)	(1) (13)	once/6 months	once/day (8)
RBM Downscale	(1) (13)	once/6 months	once/day (8)
RBM Inoperative	(1) (13)	N/A	once/day (8)
IRM Upscale	(1)(2) (13)	once/3 months	once/day (8)
IRM Downscale	(1)(2) (13)	once/3 months	once/day (8)
IRM Detector Not in Startup Position	(2) (once operating cycle)	once/operating cycle (12)	N/A
IRM Inoperative	(1)(2) (13)	N/A	N/A
SRM Upscale	(1)(2) (15)	once/3 months	once/day (8)
SRM Downscale	(1)(2) (13)	once/3 months	once/day (8)
SRM Detector Not in Startup Position	(2) (once/operating cycle)	once/operating cycle (12)	N/A
SRM Inoperative	(1)(2) (13)	N/A	N/A
Flow Bias Comparator	(1)(15)	once/operating cycle (20)	N/A
Flow Bias Upscale	(1)(15)	once/3 months	N/A
Rod Block Logic	(16)	N/A	N/A
West Scram Discharge Tank Water Level High (LS-85-45L)	once/quarter	once/operating cycle	N/A
East Scram Discharge Tank Water Level High (LS-85-45M)	once/quarter	once/operating cycle	N/A

THIS PAGE INTENTIONALLY LEFT BLANK

LIMITING CONDITIONS FOR OPERATION

3.3 REACTIVITY CONTROL

Applicability

Applies to the operational status of the control rod system.

Objective

To assure the ability of the control rod system to control reactivity.

Specification

A. Reactivity Limitations

1. Reactivity margin - core loading

A sufficient number of control rods shall be OPERABLE so that the core could be made subcritical in the most reactive condition during the operating cycle with the strongest control rod fully withdrawn and all other OPERABLE control rods fully inserted.

SURVEILLANCE REQUIREMENTS

4.3 REACTIVITY CONTROL

Applicability

Applies to the surveillance requirements of the control rod system.

Objective

To verify the ability of the control rod system to control reactivity.

Specification

A. Reactivity Limitations

1. Reactivity margin - core loading

Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.38% Δ k/k the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest OPERABLE control rod fully withdrawn and all other OPERABLE rods fully inserted.

LIMITING CONDITIONS FOR OPERATION

3.3.A.2 Reactivity margin - inoperable control rods

a. Control rod drives which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to the COLD SHUTDOWN CONDITION within 24 hours and shall not be started unless (1) investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.2.c.

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically.

SURVEILLANCE REQUIREMENTS

4.3.A.2 Reactivity margin - inoperable control rods

a. Each partially or fully withdrawn OPERABLE control rod shall be exercised one notch at least once each week when operating above the power level cutoff of the RWM. In the event power operation is continuing with three or more inoperable control rods, this test shall be performed at least once each day, when operating above the power level cutoff of the RWM.

b. DELETED

LIMITING CONDITIONS FOR OPERATION

3.3.A.2 Reactivity margin - INOPERABLE control rods (Cont'd)

c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are INOPERABLE, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

SURVEILLANCE REQUIREMENTS

4.3.A.2 Reactivity margin - INOPERABLE control rods (Cont'd)

c. When it is initially determined that a control rod is incapable of normal insertion a test shall be conducted to demonstrate that the cause of the malfunction is not a failure in the control rod drive mechanism. If this can be demonstrated an attempt to fully insert the control rod shall be made. If the control rod cannot be inserted and an investigation has demonstrated that the cause of failure is not a failed control rod drive mechanism collet housing, a shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined highest worth control rod capable of withdrawal fully withdrawn, and all other control rods capable of insertion fully inserted.

LIMITING CONDITIONS FOR OPERATION

3.3.A.2 Reactivity margin - inoperable control rods (Cont'd)

d. DELETED

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5x5 array may be inoperable (at least 4 OPERABLE control rods must separate any 2 inoperable ones). If this specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a SHUTDOWN CONDITION within 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.A.2 Reactivity margin - inoperable control rods (Cont'd)

d. The control rod accumulators shall be determined OPERABLE at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the SHUTDOWN CONDITION when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

2. The control rod drive housing support system shall be in place during REACTOR POWER OPERATION or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

   a. Verify that the control rod is following the drive by observing any response in the nuclear instrumentation each time a rod is moved when the reactor is operating above the preset power level cutoff of the RWM.

   b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

3.a DELETED

3.b Whenever the reactor is in the startup or run modes below 10% rated power, the Rod Worth Minimizer (RWM) shall be OPERABLE.

1. Should the RWM become inoperable after the first twelve rods have been withdrawn, the start-up may continue provided that a second licensed operator or other technically qualified member of the plant staff is present at the console verifying compliance with the prescribed control rod program.

2. Should the RWM be inoperable before the first twelve rods are withdrawn, start-up may continue provided a second licensed operator or other technically qualified member of the plant staff is present at the console verifying compliance with the prescribed control rod program. Use of this provision is limited to one plant startup per calendar year.

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

3.a DELETED

3.b.1 The Rod Worth Minimizer (RWM) shall be demonstrated to be OPERABLE for a reactor startup by the following checks:

a. By demonstrating that the control rod patterns and Banked Position Withdrawal Sequence (or equivalent) input to the RWM computer are correctly loaded following any loading of the program into the computer.

b. Within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical verify proper annunciation of the selection error of at least one out-of-sequence control rod.

c. Within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, the rod block function of the RWM shall be verified by moving an out-of-sequence control rod.

LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

3.b (Cont'd)

3. Should the RWM become inoperable on a shutdown, shutdown may continue provided that a second licensed operator or other technically qualified member of the plant staff is present at the console verifying compliance with the prescribed control rod program.

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

3.b.2 The Rod Worth Minimizer (RWM) shall be demonstrated to be OPERABLE for a reactor shutdown by the following checks:

a. By demonstrating that the control rod patterns and Banked Position Withdrawal Sequence (or equivalent) input to the RWM computer are correctly loaded following any loading of the program into the computer.

b. Within 8 hours prior to RWM automatic initiation when reducing thermal power, verify proper annunciation of the selection error of at least one out-of-sequence control rod.

c. Within one hour after RWM automatic initation when reducing thermal power, the rod block function of the RWM shall be verified by moving an out-of-sequence control rod.

LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

3.c. If Specifications 3.3.B.3.b.1 through 3.3.B.3.b.3 cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 10% rated power, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the shutdown position.

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.

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

3.b.3 When the RWM is not OPERABLE a second licensed operator or other technically qualified member of the plant staff shall verify that the correct rod program is followed.

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.

LIMITING CONDITIONS FOR OPERATION

3.3.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	3.5

SURVEILLANCE REQUIREMENTS

4.3.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 800 psig. This testing shall be completed prior to exceeding 40% power. Below 10% power, only rods in those sequences which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram-time tested.

THIS PAGE INTENTIONALLY LEFT BLANK

A. Reactivity Limitation

1. The requirements for the control rod drive system have been identified by evaluating the need for reactivity control via control rod movement over the full spectrum of plant conditions and events. As discussed in subsection 3.4 of the Final Safety Analysis Report, the control rod system design is intended to provide sufficient control of core reactivity that the core could be made subcritical with the strongest rod fully withdrawn. This reactivity characteristic has been a basic assumption in the analysis of plant performance. Compliance with this requirement can be demonstrated conveniently only at the time of initial fuel loading or refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration shall be performed with the reactor core in the cold, xenon-free condition and will show that the reactor is subcritical by at least R + 0.38 percent Δk with the analytically determined strongest control rod fully withdrawn.

   The value of "R", in units of percent Δk, is the amount by which the core reactivity, in the most reactive condition at any time in the subsequent operating cycle, is calculated to be greater than at the time of the demonstration. "R", therefore, is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of "R" must be positive or zero and must be determined for each fuel cycle.

   The demonstration is performed with a control rod which is calculated to be the strongest rod. In determining this "analytically strongest" rod, it is assumed that every fuel assembly of the same type has identical material properties. In the actual core, however, the control cell material properties vary within allowed manufacturing tolerances, and the strongest rod is determined by a combination of the control cell geometry and local k_∞. Therefore, an additional margin is included in the shutdown margin test to account for the fact that the rod used for the demonstration (the "analytically strongest") is not necessarily the strongest rod in the core. Studies have been made which compare experimental criticals with calculated criticals. These studies have shown that actual criticals can be predicted within a given tolerance band. For gadolinia cores the additional margin required due to control cell material manufacturing tolerances and calculational uncertainties has experimentally been determined to be 0.38 percent Δk. When this additional margin is demonstrated, it assures that the reactivity control requirement is met.

2. Reactivity margin - inoperable control rods - Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If the rod is fully inserted and disarmed electrically*, it is in a safe position of maximum contribution to shutdown reactivity. If it is disarmed electrically in a nonfully inserted position, that position shall be consistent with the shutdown reactivity limitations stated in Specification 3.3.A.1. This assures that the core can be shut down at all times with the remaining control rods assuming the strongest OPERABLE control rod does not insert. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress-assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed rod after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings. The Rod Worth Minimizer is not automatically bypassed until reactor power is above the preset power level cutoff. Therefore, control rod movement is restricted and the single notch exercise surveillance test is only performed above this power level. The Rod Worth Minimizer prevents movement of out-of-sequence rods unless power is above the preset power level cutoff.

B. Control Rods

1. Control rod dropout accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement could indicate an uncoupled condition. Rod position indication is required for proper function of the Rod Worth Minimizer.

* To disarm the drive electrically, four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred because, in this condition, drive water cools and minimizes crud accumulation in the drive. Electrical disarming does not eliminate position indication.

2. The control rod housing support restricts the outward movement of a control rod to less than three inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in subsection 3.5.2 of the FSAR and the safety evaluation is given in subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) restricts withdrawals and insertions of control rods to prespecified 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. Reference Sections 3.6.6, 7.16.5.3, and 14.6.2 of the FSAR, and NEDE-24011-P-A, Amendment 17.

   In performing the function described above, the RWM is not required to impose any restrictions at core power levels in excess of 10 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 10 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 10 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 constrains the control rod sequences and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer provides 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. Reference Section 7.16.5.3 of the FSAR. The RWM functions as a backup to procedural control of control rod sequences, which limit the maximum reactivity worth of control rods. When the Rod Worth Minimizer is out of service, special criteria allow a second licensed operator or other technically qualified member of the plant staff to manually fulfill the control rod pattern conformance functions of this system. The requirement that the RWM be OPERABLE for the withdrawal of the first twelve rods on a startup is to ensure that a high degree of RWM availability is maintained.

The functions of the RWM 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 10 percent, the RWM forces adherence to acceptable (Banked Position Withdrawal Sequence or equivalent) rod patterns. Above 10 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 10 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.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least three counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One OPERABLE SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two OPERABLE SRMs are provided as an added conservatism.

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10 percent 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 BWRs 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 which are assumed to scram on high neutron flux, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of control rod motion.

This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from sensor and circuit delays after which the pilot scram solenoid deenergizes to 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds, rather than 120 milliseconds, are conservatively assumed for this time interval in the transient analyses and are also included in the allowable scram insertion times of Specification 3.3.C.

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 percent ΔK. Deviations in core reactivity greater than 1 percent ΔK are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

E. No BASES provided for this specification

F. Scram Discharge Volume

The nominal stroke time for the scram discharge volume vent and drain valves is $\leq$ 30 seconds following a scram. The purpose of these valves is to limit the quantity of reactor water discharged after a scram and no direct safety function is performed. The surveillance for the valves assures that system drainage is not impeded by a valve which fails to open and that the valves are OPERABLE and capable of closing upon a scram.

References

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