July 2, 1984

Docket Nos. 50-259/260/296

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

Dear Mr. Parris:

DISTRIBUTION

Docket File	NRC PDR
Local PDR	ORB#2 Rdg.
DEisenhut	SNorris
WLong	RClark
OELD	LJHarmon
ELJordan	JNGrace
TBarnhart (12)	WJones
DBrinkman	ACRS (10)
OPA, CMiles	RDiggs
Gray File	Extra - 5

The Commission has issued the enclosed Amendment Nos. 102 , 96 and 69 to Facility Operating License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3. These amendments are in response to your application dated September 14, 1982 (TVA BFNP TS 176).

These amendments change the Technical Specifications to correct errors in the table of primary containment isolation valves, specify action required in the event of a rod block monitor channel failure, update the list of welded joints requiring inservice inspection - on Units 2 and 3, correct typographical errors, modify access control requirements for high radiation areas and incorporate 10 CFR 50 Appendix J reporting requirements.

A copy of the Safety Evaluation is also enclosed.

The change you requested for Table 3.7.A regarding Suppression Chamber Drain Valves is not included in this amendment. It will be addressed in response to your letter of June 24, 1983 (TS 176-S8).

Sincerely,

Original signed by
Richard J. Clark

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

Enclosures:
1. Amendment No. 102 to
   License No. DPR-33
2. Amendment No. 96 to
   License No. DPR-52
3. Amendment No. 69 to
   License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

*Please see previous concurrence page.

DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:ORB#2	OELD	DL:AD-OR
SNorris:ajs*	WLong*	RClark*	DVassallo*	JGray*	GLainas
06/26/84	06/26/84	06/26/84	06/26/84	06/27/84	06/27/84

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

cc:

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

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

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

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

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

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

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

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

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

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

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

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 102
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 14, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 102, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 2, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 102

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

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

   74
   110
   113
   252
   253
   253a - (New Page)
   339
   356

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

NOTES FOR TABLE 3.2.C

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

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

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

   A ratio of FRP/CMFLPD $<$1.0 is permitted at reduced power. See specification 2.1 for APRM control rod block setpoint.

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

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

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

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

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

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

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

   a. Both RBM channels are bypassed when reactor power is $\leq$30% and when a peripheral control rod is selected.

   b. The RBM need not be operable in the "startup" position of the reactor mode selector switch.

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

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

NOTES FOR TABLES 4.2.A THROUGH 4.2.H (Continued)

14. Upscale trip is functionally tested during functional test time as required by section 4.7.B.1.a and 4.7.C.1.c.

15. The flow bias comparator will be tested by putting one flow unit in "Test" (producing 1/2 scram) and adjusting the test input to obtain comparator rod block. The flow bias upscale will be verified by observing a local upscale trip light during operation and verified that it will produce a rod block during the operating cycle.

16. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

17. This calibration consists of removing the function from service and performing an electronic calibration of the channel.

18. Functional test is limited to the condition where secondary containment integrity is not required as specified in sections 3.7.C.2 and 3.7.C.3.

19. Functional test is limited to the time where the SGTS is required to meet the requirements of section 4.7.C.1.a.

20. Calibration of the comparator requires the inputs from both recirculation loops to be interrupted, thereby removing the flow bias signal to the APRM and RBM and scramming the reactor. This calibration can only be performed during an outage.

21. Logic test is limited to the time where actual operation of the equipment is permissible.

22. One channel of either the reactor zone or refueling zone Reactor Building Ventilation Radiation Monitoring System may be administratively bypassed for a period not to exceed 24 hours for functional testing and calibration.

23. The Reactor Cleanup System Space Temperature monitors are RTD's that feed a temperature switch in the control room. The temperature switch may be tested monthly by using a simulated signal. The RTD itself is a highly reliable instrument and less frequent testing is necessary.

24. This instrument check consists of comparing the thermocouple readings for all valves for consistence and for nominal expected values (not required during refueling outages).

25. During each refueling outage, all acoustic monitoring channels shall be calibrated. This calibration includes verification of accelerometer response due to mechanical excitation in the vicinity of the sensor.

26. This instrument check consists of comparing the background signal levels for all valves for consistency and for nominal expected values (not required during refueling outages).

3.2 BASES

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic, and all sensors are required to be operable.

High temperature in the vicinity of the HPCI equipment is sensed by 4 sets of 4 bimetallic temperature switches. The 16 temperature switches are arranged in 2 trip systems with 8 temperature switches in each trip system.

The HPCI trip settings of 90 psi for high flow and 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of 450" H_2O for high flow and 200°F for temperature are based on the same criteria as the HPCI.

High temperature at the Reactor Cleanup System floor drain could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This does not significantly increase the risk of an inadvertent control rod withdrawal, as the other channel is available, and the RBM is a backup system to the written sequence for withdrawal of control rods.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
6	Suppression Chamber purge inlet (FCV-64-19)		1	2.5	C	SC
6	Drywell/Suppression Chamber nitrogen purge inlet (FCV-76-17)		1	5	C	SC
6	Drywell Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-31)		1	5	C	SC
6	Suppression Chamber Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-34)		1	5	C	SC
6	Drywell/Suppression Chamber Nitrogen Purge Inlet (FCV-76-24)		1	5	C	SC
6	System Suction Isolation Valves to Air Compressors "A" and "B" (FCV-32-62, 63)		2	15	O	GC
7	RCIC Steamline Drain (FCV-71-6A, 6B)		2	5	O	GC
7	RCIC Condensate Pump Drain (FCV-71-7A, 7B)		2	5	C	SC
7	HPCI Hotwell pump discharge isolation valves (FCV-73-17A, 17B)		2	5	C	SC
7	HPCI steamline drain (FCV 73-6A, -6B)		2	5	O	GC
8	TIP Guide Tubes (5)		1 per guide tube	NA	C	GC

Amendment No: 78, 85, 102

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
	Standby liquid control system check valves CV 63-526 & 525	1	1	NA	C	Process
	Feedwater check valves CV-3-558, 572, 554, & 568	2	2	NA	0	Process
	Control rod hydraulic return check valves CV-85-576 & 573	1	1	NA	0	Process
	RHRS - LPCI to reactor check valves CV-74-54 & 68	2		NA	C	Process
6	CAD System Torus/Drywell Exhaust to Standby Gas Treatment (FCV-84-19,20)		2	10	C	SC
	Core Spray Discharge to Reactor Check Valves FCV-75-26,54	2		NA	C	Process

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
6	Drywell ΔP air compressor suction valve (FCV-64-139)		1	10	C	SC
6	Drywell ΔP air compressor discharge valve (FCV-64-140)		1	10	C	SC
6	Drywell CAM suction valves (FCV-90-254A and 254B)		2	10	0	GC
6	Drywell CAM discharge valves (FCV-90-257A and 257B)		2	10	0	GC
6	Drywell CAM suction valve (FCV-90-255)		1	10	0	GC

C. Drills on actions to be taken under emergency conditions involving release of radioactivity are specified in the radiological emergency plan and shall be conducted annually. Annual drills shall also be conducted on the actions to be taken following failures of safety related systems or components.

D. Radiation Control Procedures

Radiation Control Procedures shall be maintained and made available to all station personnel. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20 except in lieu of the "control device" or "alarm signal" required by paragraph 20.203 (c) of 10 CFR 20:

1. Each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Special Work Permit[a]. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

   a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

   b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

   c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Special Work Permit.

2. Each high radiation area in which the intensity of radiation is greater than 1,000 mrem/hr shall be subject to the provisions of (1) above; and, in addition, access to the source and/or area shall be secured by lock(s). The key(s) shall be under the administrative control of the shift engineer. In the case of a high radiation area established for a period of 30 days or less, direct surveillance to prevent unauthorized entry may be substituted for permanent access control.

[a] Health Physics personnel, or personnel escorted by Health Physics personnel, in accordance with approved emergency procedures, shall be exempt from the SWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

B. Source Tests

Results of required leak tests performed on sources if the tests reveal the presence of 0.005 microcurie or more of removable contamination.

C. Special Reports (in writing to the Director of Regional Office of Inspection and Enforcement).

1. Reports on the following areas shall be submitted as noted:

a.	Secondary Containment Leak Rate Testing (5)	4.7.C	Within 90 days of completion of each test.
b.	Fatigue Usage Evaluation	6.6	Annual Operating Report
c.	Relief Valve Tailpipe Instrumentation	3.2.F	Within 30 days after inoperability of thermocouple and acoustic monitor on one valve.
d.	Seismic Instrumentation Inoperability	3.2.J.3	Within 10 days after 30 days of inoperability
e.	Meteorological Monitoring Instrumentation Inoperability	3.2.I.2	Within 10 days after 7 days of inoperability
f.	Primary Containment Integrated Leak Rate Testing	4.7.A.2	Within 90 days of completion of each test.

D. Special Report (in writing to the Director of Regional Office of Inspection and Enforcement)

Data shall be retrieved from all seismic instruments actuated during a seismic event and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be submitted within 10 days after the event describing the magnitude, frequency spectrum, and resultant effect upon plant features important to safety.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 96
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 14, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility Operating License No. DPR-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 96, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 2, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 96

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

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

   74
   110
   113
   183
   339
   356

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

NOTES FOR TABLE 3.2.C

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

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

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

   A ratio of FRP/CMFLPD $<$1.0 is permitted at reduced power. See specification 2.1 for APRM control rod block setpoint.

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

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

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

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

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

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

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

   a. Both RBM channels are bypassed when reactor power is $\leq$30% and when a peripheral control rod is selected.

   b. The RBM need not be operable in the "startup" position of the reactor mode selector switch.

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

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

NOTES FOR TABLES 4.2.A THROUGH 4.2.H (Continued)

14. Upscale trip is functionally tested during functional test time as required by section 4.7.B.1.a and 4.7.C.1.c.

15. The flow bias comparator will be tested by putting one flow unit in "Test" (producing 1/2 scram) and adjusting the test input to obtain comparator rod block. The flow bias upscale will be verified by observing a local upscale trip light during operation and verified that it will produce a rod block during the operating cycle.

16. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

17. This calibration consists of removing the function from service and performing an electronic calibration of the channel.

18. Functional test is limited to the condition where secondary containment integrity is not required as specified in sections 3.7.C.2 and 3.7.C.3.

19. Functional test is limited to the time where the SGTS is required to meet the requirements of section 4.7.C.1.a.

20. Calibration of the comparator requires the inputs from both recirculation loops to be interrupted, thereby removing the flow bias signal to the APRM and RBM and scramming the reactor. This calibration can only be performed during an outage.

21. Logic test is limited to the time where actual operation of the equipment is permissible.

22. One channel of either the reactor zone or refueling zone Reactor Building Ventilation Radiation Monitoring System may be administratively bypassed for a period not to exceed 24 hours for functional testing and calibration.

23. The Reactor Cleanup System Space Temperature monitors are RTD's that feed a temperature switch in the control room. The temperature switch may be tested monthly by using a simulated signal. The RTD itself is a highly reliable instrument and less frequent testing is necessary.

24. This instrument check consists of comparing the thermocouple readings for all valves for consistence and for nominal expected values (not required during refueling outages).

25. During each refueling outage, all acoustic monitoring channels shall be calibrated. This calibration includes verification of accelerometer response due to mechanical excitation in the vicinity of the sensor.

26. This instrument check consists of comparing the background signal levels for all valves for consistency and for nominal expected values (not required during refueling outages).

Amendment No. ~~68~~, 96

3.2 BASES

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic, and all sensors are required to be operable.

High temperature in the vicinity of the HPCI equipment is sensed by 4 sets of 4 bimetallic temperature switches. The 16 temperature switches are arranged in 2 trip systems with 8 temperature switches in each trip system.

The HPCI trip settings of 90 psi for high flow and 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of 450" H_2O for high flow and 200°F for temperature are based on the same criteria as the HPCI.

High temperature at the Reactor Cleanup System floor drain could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This does not significantly increase the risk of an inadvertent control rod withdrawal, as the other channel is available, and the RBM is a backup system to the written sequence for withdrawal of control rods.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.G Structural Integrity	4.6.G Structural Integrity

maintained at the level required by the original acceptance standards throughout the life of the plant. The reactor shall be maintained in a cold shutdown condition until each indication of a defect has been investigated and evaluated.

inservice inspection surveillance requirements of the reactor coolant system as follows:

a. areas to be inspected

b. percent of areas to be inspected during the inspection interval

c. inspection frequency

d. methods used for inspection

2. Evaluation of inservice inspections will be made to the acceptance standards specified for the original equipment.

3. The inspection interval shall be 10 years.

4. Additional inspections shall be performed on certain circumferential pipe welds as listed to provide additional protection against pipe whip, which could damage auxiliary and control systems.

Feedwater	-	GFW-9, KFW-13 GFW-12, GFW-26, KFW-31, GFW-29, KFW-39, GFW-15, KFW-38, and GFW-32
Main steam	-	GMS-6, KMS-24, GMS-32, KMS-104 GMS-15, and GMS-24
RHR	-	DSRHR-4, DSRHR-7, DSRHR-6
Core Spray	-	TCS-407 TCS-423 TSCS-408 TSCS-424

C. Drills on actions to be taken under emergency conditions involving release of radioactivity are specified in the radiological emergency plan and shall be conducted annually. Annual drills shall also be conducted on the actions to be taken following failures of safety related systems or components.

D. Radiation Control Procedures

Radiation Control Procedures shall be maintained and made available to all station personnel. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20 except in lieu of the "control device" or "alarm signal" required by paragraph 20.203 (c) of 10 CFR 20:

1. Each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Special Work Permit[a]. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

   a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

   b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

   c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Special Work Permit.

2. Each high radiation area in which the intensity of radiation is greater than 1,000 mrem/hr shall be subject to the provisions of (1) above; and, in addition, access to the source and/or area shall be secured by lock(s). The key(s) shall be under the administrative control of the shift engineer. In the case of a high radiation area established for a period of 30 days or less, direct surveillance to prevent unauthorized entry may be substituted for permanent access control.

[a] Health Physics personnel, or personnel escorted by Health Physics personnel, in accordance with approved emergency procedures, shall be exempt from the SWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

6.0 ADMINISTRATIVE CONTROLS

B. Source Tests

Results of required leak tests performed on sources if the tests reveal the presence of 0.005 microcurie or more of removable contamination.

C. Special Reports (in writing to the Director of Regional Office of Inspection and Enforcement).

1. Reports on the following areas shall be submitted as noted:

a.	Secondary Containment Leak Rate Testing(5)	4.7.C	Within 90 days of completion of each test.
b.	Fatigue Usage Evaluation	6.6	Annual Operating Report
c.	Relief Valve Tailpipe Instrumentation	3.2.F	Within 30 days after inoperability of thermocouple and acoustic monitor on one valve.
d.	Seismic Instrumentation Inoperability	3.2.J.3	Within 10 days after 30 days of inoperability
e.	Meteorological Monitoring Instrumentation Inoperability	3.2.I.2	Within 10 days after 7 days of inoperability
f.	Primary Containment Integrated Leak Rate Testing	4.7.A.2	Within 90 days of completion of each test.

D. Special Report (in writing to the Director of Regional Office of Inspection and Enforcement)

Data shall be retrieved from all seismic instruments actuated during a seismic event and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be submitted within 10 days after the event describing the magnitude, frequency spectrum, and resultant effect upon plant features important to safety.

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. 69
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 14, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility Operating License No. DPR-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 69, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 2, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 69

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

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

   77
   107
   110
   197
   265
   369
   386

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

NOTES FOR TABLE 3.2.C

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

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

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

   -A ratio of FRP/CMFLPD $<$ 1.0 is permitted at reduced power. See specification 2.1 for APRM control rod block setpoint.

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

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

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

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

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

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

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

   a. Both RBM channels are bypassed when reactor power is $\leq$30% and when a peripheral control rod is selected.

   b. The RBM need not be operable in the "startup" position of the reactor mode selector switch.

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

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

JUN 18 1981

NOTES FOR TABLES 4.2.A THROUGH 4.2.H (Continued)

14. Upscale trip is functionally tested during functional test time as required by section 4.7.B.1.a and 4.7.C.1.c.

15. The flow bias comparator will be tested by putting one flow unit in "Test" (producing 1/2 scram) and adjusting the test input to obtain comparator rod block. The flow bias upscale will be verified by observing a local upscale trip light during operation and verified that it will produce a rod block during the operating cycle.

16. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

17. This calibration consists of removing the function from service and performing an electronic calibration of the channel.

18. Functional test is limited to the condition where secondary containment integrity is not required as specified in sections 3.7.C.2 and 3.7.C.3.

19. Functional test is limited to the time where the SGTS is required to meet the requirements of section 4.7.C.1.a.

20. Calibration of the comparator requires the inputs from both recirculation loops to be interrupted, thereby removing the flow bias signal to the APRM and RBM and scramming the reactor. This calibration can only be performed during an outage.

21. Logic test is limited to the time where actual operation of the equipment is permissible.

22. One channel of either the reactor zone or refueling zone Reactor Building Ventilation Radiation Monitoring System may be administratively bypassed for a period not to exceed 24 hours for functional testing and calibration.

23. The Reactor Cleanup System Space Temperature monitors are RTD's that feed a temperature switch in the control room. The temperature switch may be tested monthly by using a simulated signal. The RTD itself is a highly reliable instrument and less frequent testing is necessary.

24. This instrument check consists of comparing the thermocouple readings for all valves for consistence and for nominal expected values (not required during refueling outages).

25. During each refueling outage, all acoustic monitoring channels shall be calibrated. This calibration includes verification of accelerometer response due to mechanical excitation in the vicinity of the sensor.

26. This instrument check consists of comparing the background signal levels for all valves for consistency and for nominal expected values (not required during refueling outages).

Pressure instrumentation is provided to close the main steam isolation valves in Run Mode when the main steam line pressure drops below 825 psig.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic, and all sensors are required to be operable.

High temperature in the vicinity of the HPCI equipment is sensed by 4 sets of 4 bimetallic temperature switches. The 16 temperature switches are arranged in 2 trip systems with 8 temperature switches in each trip system.

The HPCI trip settings of 90 psi for high flow and 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of 450" water for high flow and 200°F for temperature are based on the same criteria as the HPCI.

High temperature at the Reactor Cleanup System floor drain could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07 The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This does not significantly increase the risk of an inadvertent control rod withdrawal, as the other channel is available, and the RBM is a backup system to the written sequence for withdrawal of control rods.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control

Amendment No. ~~4~~, ~~18~~, 69

3.6 PRIMARY SYSTEM BOUNDARY

4.6 PRIMARY SYSTEM BOUNDARY

Main steam	-GMS-6, KMS-24, GMS-32, KMS-104, GMS-15, and GMS-24
RHR	-DSRHR-6, DSRHR-7, and DSRHR-4
Core Spray	- TCS-407 TCS-423 TSCS-408 TSCS-424
Reactor Cleanup	-DSRWC-4, DSRWC-3, DSRWC-6, and DSRWC-5
HPCI	-THPCI-70 THPCI-70A THPCI-71, and THPCI-72

5. System hydrostatic tests in accordance with Article IS-500 of Section XI of the ASME Code at or near the end of each inspection interval and prior to startup following each refueling outage. The pressure-temperature limits for these tests will be in accordance with specification 3.6.A.3.

REFERENCE

1. Plant Safety Analysis (BFNP FSAR subsection 4.12)

Amendment No. ~~57~~, 69

TABLE 3.7.A
PRIMARY CONTAINMENT ISOLATION VALVES

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
	Standby liquid control system check valves (CV 63-526 & 525)	1	1	NA	C	Process
	Feedwater check valves (CV-3-558, 572, 554 & 568)	2	2	NA	0	Process
	Control rod hydraulic return check valves (CV-85-576 & 573)	1	1	NA	0	Process
	RHRS -'LPCI to reactor check valves (CV-74-54 & 68)	2		NA	C	Process
	Core Spray discharge to reactor check valves (FCV-75-26 and 54)	2		NA	C	Process
6	Drywell ΔP air compressor suction valve (FCV 64-139)		1	10	C	SC
6	Drywell ΔP air compressor discharge valve (FCV 64-140)		1	10	C	SC
6	Drywell CAM discharge valves (FCV 90-257A and 257B)		2	10	0	GC
6	Drywell CAM suction valves (FCV 90-254A and 254B)		2	10	0	GC
6	Drywell CAM suction valve (FCV 90-255)		1	10	0	GC
6	CAD System Torus/Drywell Exhaust to Standby Gas Treatment (FCV 84-19, 20)		2	10	C	SC

C. Drills on actions to be taken under emergency conditions involving release of radioactivity are specified in the radiological emergency plan and shall be conducted annually. Annual drills shall also be conducted on the actions to be taken following failures of safety related systems or components.

D. Radiation Control Procedures

Radiation Control Procedures shall be maintained and made available to all station personnel. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20 except in lieu of the "control device" or "alarm signal" required by paragraph 20.203 (c) of 10 CFR 20:

1. Each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Special Work Permit[a]. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

   a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

   b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

   c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Special Work Permit.

2. Each high radiation area in which the intensity of radiation is greater than 1,000 mrem/hr shall be subject to the provisions of (1) above; and, in addition, access to the source and/or area shall be secured by lock(s). The key(s) shall be under the administrative control of the shift engineer. In the case of a high radiation area established for a period of 30 days or less, direct surveillance to prevent unauthorized entry may be substituted for permanent access control.

[a] Health Physics personnel, or personnel escorted by Health Physics personnel, in accordance with approved emergency procedures, shall be exempt from the SWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

6.0 ADMINISTRATIVE CONTROLS

B. Source Tests

Results of required leak tests performed on sources if the tests reveal the presence of 0.005 microcurie or more of removable contamination.

C. Special Reports (in writing to the Director of Regional Office of Inspection and Enforcement).

1. Reports on the following areas shall be submitted as noted:

a.	Secondary Containment Leak Rate Testing (5)	4.7.C	Within 90 days of completion of each test.
b.	Fatigue Usage Evaluation	6.6	Annual Operating Report
c.	Relief Valve Tailpipe Instrumentation	3.2.F	Within 30 days after inoperability of thermocouple and acoustic monitor on one valve.
d.	Seismic Instrumentation Inoperability	3.2.J.3	Within 10 days after 30 days of inoperability
e.	Meteorological Monitoring Instrumentation Inoperability	3.2.I.2	Within 10 days after 7 days of inoperability
f.	Primary Containment Integrated Leak Rate Testing	4.7.A.2	Within 90 days of completion of each test.

D. Special Report (in writing to the Director of Regional Office of Inspection and Enforcement)

Data shall be retrieved from all seismic instruments actuated during a seismic event and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be submitted within 10 days after the event describing the magnitude, frequency spectrum, and resultant effect upon plant features important to safety.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 96 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 69 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

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

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

1.0 Introduction

By letter dated September 14, 1982 (TVA BFNP TS 176) the Tennessee Valley Authority (the licensee or TVA) requested amendments to Facility Operating License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3. The requested changes would modify Appendix A, Technical Specifications Table 3.2.C, "Instrumentation That Initiates Rod Blocks," Table 3.7.A, "Primary Containment Isolation Valves," Specification 6.3.D, "Radiation Control Procedures," Specification 6.7.3.C, "Special Reports," and 4.6.G, "Structural Integrity."

2.0 Evaluation

Rod Block Monitor (RBM)

The licensee proposes to revise the notes to Table 3.2.C relating to the RBM. The RBM is provided for protection against a rod withdrawal error at high power. It consists of two redundant channels. Either channel initiates a rod block when tripped unless power is below 30 percent in which case both channels are automatically bypassed. Under provisions of the current Technical Specifications (TSs), if an RBM channel is discovered to be inoperable administrative controls are immediately established to prevent rod withdrawal. If the inoperable RBM channel cannot be restored in a time frame so as not to exceed 24 hours inoperability in a 30-day period, the licensee has not met the limiting conditions for operation of TS 3.2.C and must be in HOT STANDBY/STARTUP mode within 6 hours. Under provisions of the revised notes, the licensee would be required, upon discovering an RBM channel to be inoperable, to establish administrative controls to prevent rod withdrawal, then if the inoperable channel cannot be restored within the following 24 hours (regardless of the cumulative inoperable time) the channel would be tripped within the next hour. This would establish a rod withdrawal inhibit in the rod manual control system electrically preventing rod withdrawal. The reactor could be maintained at power indefinitely until the inoperable RBM channel was restored (so long as no rod withdrawal was necessary). This action is consistent with the Standard Technical Specifications (STSs) (NUREG-0123) and thus meets the acceptance criteria of Standard Review Plan Section 16. With one channel inoperable and tripped, the probability of an accident is less than during normal operation with both channels operable. This change is therefore acceptable.

Add Additional Valves to Table 3.7.A

Table 3.7.A lists primary containment isolation valves. Valves identified in this table are subject to operability and surveillance requirements. Since valves not listed in this table are not required to be periodically tested, the addition of valves to this table is inherently conservative, and does not reduce the margin of safety. These changes are therefore acceptable.

Valves added include:

Unit 1	Unit 3
FCV-84-19, 20	FCV-84-19, 20
FCV-75-26, 54	
FCV-64-139, 140	
FCV-90-254A, B	
-255	
-257A, B	

Correction of Errors in Tables 3.7.A (Unit 1)

TVA has requested correction of certain errors in Table 3.7.A. These errors include (1) incorrect valve nomenclature (FCV-73-6A, B); (2) incorrect indication of normal position (FCV-71-7A,B); and (3) incorrect indication of valve movement upon initiating signal (FCV-71-7A, B). The staff has confirmed, that (with one exception) the requested error corrections are consistent with the requirements of Standard Review Plan Section 6.2.4 "Containment Isolation System" and are acceptable. TVA requested deletion of Reactor Water Cleanup System return isolation valve FCV 69-12 from the table on the basis that it is not a primary containment isolation valve. This information is inconsistent with the Final Safety Analysis Report (FSAR). The staff will address this issue via separate correspondence.

Typographical Errors

Note 19 to Tables 4.2.A through 4.2.H cross-references section 4.7.C.1.c. The correct cross-reference is to 4.7.C.1.a.

Weld Numbers (Units 2 and 3 only)

TS 4.6.G identifies welds which require additional inspections for structural integrity. A proposed change would update 4.6.G to reflect new weld numbers resulting from pipe modifications in which part of the core spray piping was changed from stainless steel to carbon steel. This change is therefore acceptable.

High Radiation Area Access Control

The licensee has requested a change to the administrative procedures for access to high radiation areas to permit use of direct surveillance as a substitute for permanent access control to prevent unauthorized entry to a high radiation area if the high radiation area will be established for only 30 days or less.

The proposed change is consistent with 10 CFR 20.203(c)(4) and is therefore acceptable.

3.0 Environmental Considerations

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

4.0 Conclusion

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: W. O. Long

Dated: July 2, 1984