UNITED STATES NUCLEAR REGULATORY COMMISSION

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

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 45 to Facility Operating License No. DPR-33, Amendment No. 41 to Facility Operating License No. DPR-52 and Amendment No. 18 to Facility Operating License No. DPR-68 issued to Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3, located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

Amendment No. 18 changes the Technical Specifications to incorporate the limiting conditions for operation associated with the initial 2000 megawatt davs per tonne (MWD/t) fuel exposure during the second fuel cycle for Unit No. 3. The amendments also incorporate minor changes in the test setups to be used to test certain primary containment isolation and check valves.

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

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

For further details with respect to this action, see (1) the application for amendments dated August 3, 1978, as supplemented by letter dated October 20, 1978, (2) Amendment No. 45 to License No. DPR-33, Amendment No. 41 to License No. DPR-52, and Amendment No. 18 to License No. DPR-68, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 18th day of November 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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

11/16/78

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

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

Dear Mr. Hughes:

The Commission has issued the enclosed Amendments Nos. 44, 40 and 17 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. These amendments consist of changes to the Technical Specifications in response to your requests of August 2, 1978 (BFNP TS 112) and August 11, 1978 (BFNP TS 114).

The changes: (1) permit the average power range monitor system to be inoperable in the refuel mode, provided the source range monitors are connected to give a non-coincidence, high flux scram; (2) permit less than three intermediate range monitors per trip channel to be operable in the shutdown or refuel modes, provided at least four IRMs (one in each core quadrant) are connected to give a non-coincidence, high flux scram; (3) clarify ambiguous portions of the Technical Specifications related to the rod block monitor system; (4) remove reference to an obsolete 1968 version of an ASTM procedure; (5) modify the list of snubbers that are required to be operable; (6) remove a specification for additional tests of secondary containment that only applied during the first fuel cycle for each Browns Ferry Unit, and (7) alter one of the four locations where milk samples are collected. With the concurrence of your staff, we have made several minor changes in the proposed Technical Specifications which you submitted.

CP-1

MLo 20040267

OFFICE➤						
SURNAME➤						
DATE➤						

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

Sincerely,

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

Enclosures:
1. Amendment No. to DPR-33
2. Amendment No. to DPR-52
3. Amendment No. to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures:
see next page

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Tennessee Valley Authority

cc: H. S. Sanger, Jr., Esquire
General Counsel
Tennessee Valley Authority
400 Commerce Avenue
E 11B 33 C
Knoxville, Tennessee 37902

Mr. D. McCloud
Tennessee Valley Authority
303 Power Building
Chattanooga, Tennessee 37401

Mr. William E. Garner
Route 4, Box 354
Scottsboro, Alabama 35768

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 36104

Mr. C. S. Walker
Tennessee Valley Authority
400 Commerce Avenue
W 9D199 C
Knoxville, Tennessee 37902

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Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U.S. Environmental Protection Agency
Room 645, East Tower
401 M Street, SW
Washington, D.C. 20460

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS Coordinator
345 Courtland Street
Atlanta, Georgia 30308

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 44
License No. DPR-33

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

   A. The applications for amendments by Tennessee Valley Authority (the licensee) dated August 2, 1978 and August 11, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 44, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 16, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 44

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

Remove the following pages and replace with identically numbered pages:

33/34
35/36
51/52
73/74
75/76
113/114
131/132
193/194
197/198
240/241
292/293
304/305

Revise Appendix B as follows:

Remove the following page and replace with identically numbered page:

41/42

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

TABLE 3.1.A

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Min. No. of Operable Inst. Channels Per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable: Shut-down	Refuel (7)	Startup/Hot Standby	Run	Act
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤ 120/125 Indicated on scale	X(22)	X (22)	X	(5)	1.A
3	Inoperative			X	X	(5)	1.A
	APRM (16)						
2	High Flux	See Spec. 2.1.A.1				X	1.A o
2	High Flux	≤ 15% rated power (13)		X(21)	X(17)	(15)	1.A o
2	Inoperative			X(21)	X(17)	X	1.A o
2	Downscale	≥ 3 Indicated on Scale		(11)	(11)	X(12)	1.A
2	High Reactor Pressure	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14)	≤ 2 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14)	≥ 538" above vessel zero		X	X	X	1.A
2	High Water Level in Scram Discharge Tank	≤ 50 Gallons	X	X(2)	X	X	1.A

TABLE 3.1.A (Continued)

Min. No. of Operable Inst. Channels Per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable: Refuel(7)	Startup/Hot Standby	Run	Action(1)
4	Main Steam Line Isolation Valve Closure	≤ 10% Valve Closure	X(3)(6)	X(3)(6)	X(6)	1.A or 1.C
2	Turbine Cont. Valve Fast Closure	Upon trip of the fast acting solenoid valves	X(4)	X(4)	X(4)	1.A or 1.D
4	Turbine Stop Valve Closure	≤ 10% Valve Closure	X(4)	X(4)	X(4)	1.A or 1.D
2	Turbine Control Valve - Loss of Control Oil Pressure	≥ 550 psig	X(4)	X(4)	X(4)	1.A or 1.D
2	Turbine First Stage Pressure Permissive	≤ 154 psig	X(18)	X(18)	X(18)	(19)
2	Turbine Condenser Low Vacuum	≥ 23 In. Hg. Vacuum	X(3)	X(3)	X	1.A or 1.C
2	Main Steam Line High Radiation (14)	≤ 3X Normal Full Power Background (20)	X(9)	X(9)	X(9)	1.A or 1.C

NOTES FOR TABLE 3.1.A

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable instrument channels per trip system cannot be met for both trip systems, the appropriate actions listed below shall be taken.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

   B. Reduce power level to IRM range and place mode switch in the Startup/Hot Standby position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30% of rated.

2. Scram discharge volume high bypass may be used in shutdown or refuel to bypass scram dischatge volume scram with control rod block for reactor protection system reset.

3. Bypassed if reactor pressure < 1055 psig and mode switch not in run.

4. Bypassed when turbine first stage pressure is less than 154 psig.

5. IRM's are bypassed when APRM's are onscale and the reactor mode switch is in the run position.

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   A. Mode switch in shutdown

   B. Manual scram

   C. High flux IRM

   D. Scram discharge volume high level

   E. APRM 15% scram

8. Not required to be operable when primary containment integrity is not required.

9. Not required if all main steamlines are isolated.

10. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

11. The APRM downscale trip function is only active when the reactor mode switch is in run.

12. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.

13. Less than 14 operable LPRM's will cause a trip system trip.

14. Channel shared by Reactor Protection System and Primary Containment and Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

15. The APRM 15% scram is bypassed in the Run Mode.

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system.

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

18. Operability is required when normal first-stage pressure is below 30% (≤ 154 psig).

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required.

20. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in primary coolant.

21. The APRM High Flux and Inoperative Trips do not have to be operable in the Refuel Mode if the Source Range Monitors are connected to give a non-coincidence, High Flux scram, at $\leq 5 \times 10^5$ cps. The SRM's shall be operable per Specification 3.10.B.1. The removal of eight (8) shorting links is required to provide non-coincidence high-flux scram protection from the Source Range Monitors.

22. The three required IRM's per trip channel is not required in the Shutdown or Refuel Modes if at least four IRM's (one in each core quadrant) are connected to give a non-coincidence, High Flux scram. The removal of four (4) shorting links is required to provide non-coincidence high-flux scram protection from the IRMs.

3.2.B Core and Containment Cooling Systems - Initiation & Control

C. Control Rod Block Actuation

The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.C.

DELETE
Now covered by note 7.c.

D. Off-Gas Post Treatment Isolation Function

1. Off Gas Post Treatment Monitors

(a) Except as specified in (b) below, both off-gas post treatment radiation monitors shall be operable during reactor operation. The isolation function trip settings for the monitors shall be set at a value not to exceed the equivalent of the stack release limit specified in specification 3.8.B.1.

4.2.B Core and Containment Cooling Systems - Initiation & Control

are required to be operable shall be considered operable if they are within the required surveillance testing frequency and there is no reason to suspect that they are inoperable.

C. Control Rod Block Actuation

Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.C.

System logic shall be functionally tested as indicated in Table 4.2.C.

D. Off-Gas Post Treatment Isolation Functions

1. Off-Gas Post Treatment Monitoring System

Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.D.

System logic shall be functionally tested as indicated in Table 4.2.D.

3.2.D Off-Gas Post Treatment Isolation Functions

(b) From and after the date that one of the two off-gas post treatment radiation monitors is made or found to be inoperable, continued reactor power operation is permissible during the next seven days, provided that the inoperable monitor is tripped in the downscale position. One radiation monitor may be out of service for four hours for functional test and/or calibration without the monitor being in a downscale tripped condition.

(c) Upon the loss of both off-gas post treatment radiation monitors, initiate an orderly shutdown and shut the mainsteam isolation valves or the off-gas isolation valve within 10 hours.

E. Drywell Leak Detection

The limiting conditions of operation for the instrumentation that monitors drywell leak detection are given in Table 3.2.E.

F. Surveillance Instrumentation

The limiting conditions for the instrumentation that provides surveillance information readouts are given in Table 3.2.F.

G. Control Room Isolation

The limiting conditions for instrumentation that isolates the control room and initiates the control room emergency pressurization systems are given in Table 3.2.G.

4.2.D Off-Gas Post Treatment Isolation Function

E. Drywell Leak Detection

Instrumentation shall be calibrated and checked as indicated in Table 4.2.E.

F. Surveillance Instrumentation

Instrumentation shall be calibrated and checked as indicated in Table 4.2.F.

G. Control Room Isolation

Instrumentation shall be calibrated and checked as indicated in Table 4.2.G.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum No. Operable Per Trip Sys (5)	Function	Trip Level Setting
2(1)	APRM Upscale (Flow Bias)	$\leq$ 0.66W + 42% (2)
2(1)	APRM Upscale (Startup Mode) (8)	$\leq$ 12%
2(1)	APRM Downscale (9)	$\geq$ 3%
2(1)	APRM Inoperative	(10_b)
1(7)	RBM Upscale (Flow Bias)	$\leq$ 0.66W + 41% (2) for two recirculation loop operation $\leq$ 0.66W + 37.7%(2) for one recirculation loop operati
1(7)	RBM Downscale (9)	$\geq$ 3%
1(7)	RBM Inoperative	(10_c)
3(1)	IRM Upscale (8)	$\leq$ 108/125 of full scale
3(1)	IRM Downscale (3)(8)	$\geq$ 5/125 of full scale
3(1)	IRM Detector not in Startup Position (8)	(11)
3(1)	IRM Inoperative (8)	(10^a)
2(1)(6)	SRM Upscale (8)	$\leq$ 1×10^5 counts/sec.
2(1)(6)	SRM Downscale (4)(8)	$\geq$ 3 counts/sec.
2(1)(6)	SRM Detector not in Startup Position (4)(8)	(11)
2(1)(6)	SRM Inoperative (8)	(10a)
2(1)	Flow Bias Comparator	$\leq$ 10% difference in recirculation flows
2(1)	Flow Bias Upscale	$\leq$ 110% recirculation flow
1(1)	Rod Block Logic	N/A
2(1)	RSCS Restraint (PS-85-61A & PS-85-61B)	147 psig turbine first stage pressure (approximately 30% power)

NOTES FOR TABLE 3.2.C

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

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

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

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

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

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

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

7. The following operational restraints apply to the RBM only:
   a. Both RBM channels are bypassed when reactor power is ≤ 30%.
   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. An RBM channel may be out of service for testing and/or maintenance provided this condition does not last longer than 24 hours in any thirty day period.
   d. If minimum conditions for Table 3.2.C are not met, administrative controls shall be immediately imposed to prevent control rod withdrawal.

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

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

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

   a. SRM and IRM

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

      (2) Power supply voltage low.

      (3) Circuit boards not in circuit.

   b. APRM

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

      (2) Less than 14 LPRM inputs.

      (3) Circuit boards not in circuit.

   c. RBM

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

      (2) Circuit boards not in circuit.

      (3) RBM fails to null.

      (4) Less than required number of LPRM inputs for rod selected.

11. Detector traverse is adjusted to 114 $\pm$ 2 inches, placing the detector lower position 24 inches below the lower core plate.

TABLE 3.2.D
OFF-GAS POST TREATMENT ISOLATION INSTRUMENTATION

Min. No. Operable (1)	Function	Trip Level Setting	Action (2)	Remarks
2	Off-Gas Post Treatment Monitor	Note 3	A or B	1. 2 upscales, or 1 downscale and 1 upscale, or 2 downscales will isolate off-gas line.
1	Off-Gas Post Treatment Isolation	Note 3	B	1. One trip system with auto transfer to another source

NOTES:

1. Whenever the minimum number operable cannot be met, the indicated action shall be taken.
2. Action
   A. Refer to Section 3.2.D.1.b
   B. Refer to Section 3.2.D.1.c
3. Trip setting to correspond to Specification 3.2.D.1.a

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 1.06. 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. Two RBM channels are provided and one of these may be bypassed from the console, for maintenance and/or testing, provided that this out of service condition does not last longer than 24 hours in any thirty day period. This time period is only 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

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 1.06.

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.

If the IRM channels are in the worst condition of allowed bypass; the sealing arrangement is such that for unbypassed IRM channels, a rod block signal is generated before the detected neutrons flux has increased by more than a factor of 10.

A downscale indication is an indication the instrument has failed or the instrument is not sensitive enough; In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented.

The refueling interlocks also operate one logic channel; and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two post treatment off-gas radiation monitors are provided and, when their trip point is reached, cause an isolation of the off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip or both have a downscale trip.

Both instruments are required for trip but the instruments are set so that any instruments are set so that the instantaneous stack release rate limit given in Specification 3.8 is not exceeded.

Four radiation monitors are provided for each unit which initiate Primary Containment Isolation (Group 6 isolation valves) Reactor Building Isolation and operation of the Standby Gas Treatment System. These instrument channels monitor the radiation in the Reactor zone ventilation exhaust ducts and in the Refueling Zone.

Trip setting of 100 mr/hr for the monitors in the Refueling Zone are based upon initiating normal ventilation isolation and SGTS operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the SGTS.

Flow integrators and sump fill rate and pump out rate timers are used to determine leakage in the drywell. A system whereby the time interval to fill a known volume will be utilized to provide a backup. An air sampling system is also provided to detect leakage inside the primary containment (See Table 3.2.E).

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 SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two RBM channels are provided and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from one of the channels will block erroenous rod withdrawal soon enough to prevent fuel damage. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

   A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit, (ie, MCPR given by figure 3.5.3 or LHGR of 18.5 for 7x7 or 13.4 for 8x8) During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is normally the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform these functions may be designated by the plant superintendent to perform these functions.

Scram Insertion Times

The control rod system is designated to bring the reactor subcritical at the rate fast enough to prevent fuel damage: ie, to prevent the MCPR from becoming less than 1.06. The limiting power transient is given in Reference 1. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification provide the required protection, and MCPR remains greater than 1.06.

On an early BWR, some degradation of control rod scram performance occured during plant startup and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model drive with a modified (larger screen size) internal filter which is less prone to plugging. Data has been documented by surveillance reports in various operating plants. These include Oyster Creek, Monticello, Dresden 2 and Dresden 3. Approximately 5000 drive tests have been recorded to date.

Following identification of the "plugged filter" problem, very frequent scram tests were necessary to ensure proper performance. However, the more frequent scram tests are now considered totally unnecessary and unwise for the following reasons:

1. Erratic scram performance has been identified as due to an obstructed drive filter in type "A" drives. The drives in BFNP are of the new "B" type design whose scram performance is unaffected by filter condition.

2. The dirt load is primarily released during startup of the reactor when the reactor and its systems are first subjected to flows and pressure and thermal stresses. Special attention and measures are now being taken to assure cleaner systems. Reactors with drives identical or similar (shorter stroke, smaller piston areas) have operated through many refueling cycles with no sudden or erratic changes in scram performance. This preoperational and startup testing is sufficient to detect anomalous drive performance.

3. he 72-hour outage limit which initiated the start of the frequent scram testing is arbitrary, having no logical basis other than quantifying a "major outage" which might reasonably be caused by an event so severe as to possibly affect drive performance. This requirement is unwise because it provides an incentive for shortcut actions to hasten returning "on line" to avoid the additional testing due a 72-hour outage.

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown *	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible Du Normal Opera
R16 upper	RHR	598				X
R16 lower	RHR	598				X
R19	RHR	555				X
R20 upper	RHR	549				X
R21 - east	RHR	572				X
R21 - west	RHR	572				X
R22	RHR	573				X
R24	RHR	580			X	
R25	RHR	579			X	
R26	RHR	575			X	
R41 inside	RHR	555				X
R41 outside	RHR	555				X
R29	RHR head spray	636			X	
R29	RHR head spray	636			X	

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TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown *	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible Duri Normal Operati
R1	Core spray	606			X	
R2	Core spray	606			X	
R6 - north	Core spray	544				X
R6 - south	Core spray	544				X
R8	Core spray	609			X	
R9	Core spray	609			X	
R13 - north	Core spray	544				X
R13 - south	Core spray	544				X
R19	Standby liquid control	624			X	
R21	Standby liquid control	624				X
R3 - north	HPCI	542				X
R3 - south	HPCI	542				X
R6	HPCI	563			X	X
R9	HPCI	547				X
R11	HPCI	532				
R47	HPCI	532				X
R47	HPCI	532				X

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown*	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SSZ-4A	PSC (ring header)	525				X
SSZ-5A	PSC (ring header)	525				X
SSX-6A	PSC (ring header)	525				X
SSX-7A	PSC (ring header)	525				X
SSZ-8A	PSC (ring header)	525				X
R2A	Fire Protection	601				X
R3A	Fire Protection	601				X
R4	Fire Protection	601				X
R42	EECW	605				X
SS1-A	Recirculation	556			X	
SS1-B	Recirculation	556			X	
SS2-A	Recirculation	558			X	
SS2-B	Recirculation	558			X	

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown*	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SS3-A(295°)	Recirculation	564			X	
SS3-A(335°)	Recirculation	564			X	
SS3-B(115°)	Recirculation	564			X	
SS3-B(154°)	Recirculation	564			X	
SS4-A	Recirculation	570			X	
SS4-B	Recirculation	570			X	
SS5-A(262°)	Recirculation	581			X	
SS5-B(325°)	Recirculation	581			X	
SS5-B(35°)	Recirculation	581			X	
SS5-B(98°)	Recirculation	581			X	
SS6-A	Recirculation	568			X	
SS6-B	Recirculation	568			X	
SS7	Recirculation	564			X	
SS8	Recirculation	564			X	

*Modifications to this Table due to changes in high radiation areas should be submitted to the NRC as part of the next license amendment.

3.7.C **Secondary Containment**

1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.

4.7.C **Secondary Containment**

1. Secondary containment surveillance shall be performed as indicated below:

   a. A preoperational secondary containment capability test shall be conducted by isolating the reactor building and placing two standby gas treatment system filter trains in operation. Such test shall demonstrate the

3.7.C Secondary Containment

2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:

   a. The reactor shall be made subcritical and Specification 3.3.A shall be met.

   b. The reactor shall be cooled down below 212°F and the reactor coolant system vented.

   c. Fuel movement shall not be permitted in the reactor zone.

   d. Primary containment integrity maintained.

3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.

4.7.C Secondary Containment

capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system inleakage rate of not more than 12,000 cfm.

b. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system inleakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.

2. After a secondary containment violation is determined the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

3.9 AUXILIARY ELECTRICAL SYSTEM

Applicability

Applies to the auxiliary electrical power system.

Objective

To assure an adequate supply of electrical power for operation of those systems required for safety.

Specification

A. Auxiliary Electrical Equipment

A reactor shall not be started up (made critical) from the cold condition unless four units 1 and 2 diesel generators are operable, both 161-kV transmission lines, two common station service transformers and one cooling tower transformer are operable, and the requirements of 3.9.A.4 through 3.9.A.7 are met.

A reactor shall not be started up (made critical) from the Hot Standby Condition unless all of the following conditions are satisfied:

1. At least one off-site 161-kV transmission line and its common transformer are available and capable of automatically supplying auxiliary power to the shutdown boards.

2. Three units 1 and 2 diesel generators shall be operable.

3. An additional source of power consisting of one of the following:

   a. A second 161-kV transmission line and its

4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective

Verify the operability of the auxiliary electrical system.

Specification

A. Auxiliary Electrical Equipment

1. Diesel Generators

   a. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue for at least a one-hour period at 75% of rated load or greater.

      During the monthly generator test the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps shall be demonstrated, and the diesel starting time to reach rated voltage and speed shall be logged.

   b. Once per operating cy- a test will be conducted to demonstrate the emergency diesel generators will start and accept emergency load within

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9.A Auxiliary Electrical Equipment	4.9.A Auxiliary Electrical Equipment

common transformer and cooling tower transformer capable of supplying power to the shutdown boards.

b. A fourth operable units 1 and 2 diesel generator.

4. Buses and Boards Available

a. Start buses 1A and 1B are energized.

b. The units 1 and 2 4-kV shutdown boards are energized.

c. The 480-V shutdown boards associated with the unit are energized.

d. Undervoltage relays operable on start buses 1A and 1B and 4-kV shutdown boards, A, B, C, and D.

5. The 250-Volt unit and shutdown board batteries and a battery charger for each battery and associated battery boards are operable.

6. Logic Systems

a. Common accident signal logic system is operable.

b. 480-V load shedding logic system is operable.

7. There shall be a minimum of 103,300 gallons of diesel fuel in the standby diesel generator fuel tanks.

the specified time sequence.

c. Once a month the quantity of diesel fuel available shall be logged.

d. Each diesel generator shall be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.

e. Once a month a sample of diesel fuel shall be checked for quality. The quality shall be within the acceptable limits specified in Table 1 of the latest revision to ASTM D975 and logged.

2. D.C. Power System - Unit Batteries (250-Volt) Diesel Generator Batteries (125-Volt) and Shutdown Board Batteries (250-Volt)

a. Every week the specific gravity and the voltage of the pilot cell, and temperature of an adjacent cell and overall battery voltage shall be measured and logged.

b. Every three months the measurements shall be made of voltage of each cell to nearest 0.1 volt, specific gravity of each cell, and temperature of every fifth cell. These measurements shall be logged.

c. A battery rated discharge (capacity) test shall be performed and the voltage, time, and output current measurements shall be logged at intervals not to exceed 24 months.

3.10.A Refueling Interlocks	4.10.A Refueling Interlocks
refueling interlocks shall be operable. b. A sufficient number of control rods shall be operable so that the core can be made subcritical with the strongest operable control rod fully withdrawn and all other operable control rods fully inserted, or all directional control valves for remaining control rods shall be disarmed electrically and sufficient margin to criticality shall be demonstrated. c. If maintenance is to be performed on two control rod drives they must be separated by more than two control cells in any direction. d. An appropriate number of SRM's are available as defined in specification 3.10.A.	
6. Any number of control rods may be withdrawn or removed from the reactor core providing the following conditions are satisfied: a. The reactor mode switch is locked in the "refuel" position. The refueling interlock which prevents more than one control rod from	3. With the mode selection switch in the refuel or shutdown mode, no control rod may be withdrawn until two licensed operators have confirmed that either all fuel has been removed from around that rod or that all control rods in immediately adjacent cells have been fully inserted and electrically disarmed.

3.10.A Refueling Interlocks

being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.

B. Core Monitoring

1. During core alterations, except as in 3.10.B.2, two SRM's shall be operable, in or adjacent to any quadrant where fuel or control rods are being moved. For an SRM to be considered operable, the following shall be satisfied:

   a. The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.)

   b. The SRM shall have a minimum of 3 cps with all rods fully inserted in the core, if one or more fuel assemblies are in the core.

2. During a complete core removal, the SRM's shall have an initial minimum count rate of 3 cps prior to fuel removal, with all rods fully inserted and rendered electrically inoperable. The count rate will diminish during fuel removal. Individual control rods outside the periphery of the then existing fuel matrix may be electrically armed and moved for maintenance after all fuel in the cell containing (controlled by) that control rod have been removed from the reactor core.

4.10.A Refueling Interlocks

B. Core Monitoring

Prior to making any alterations to the core the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for response.

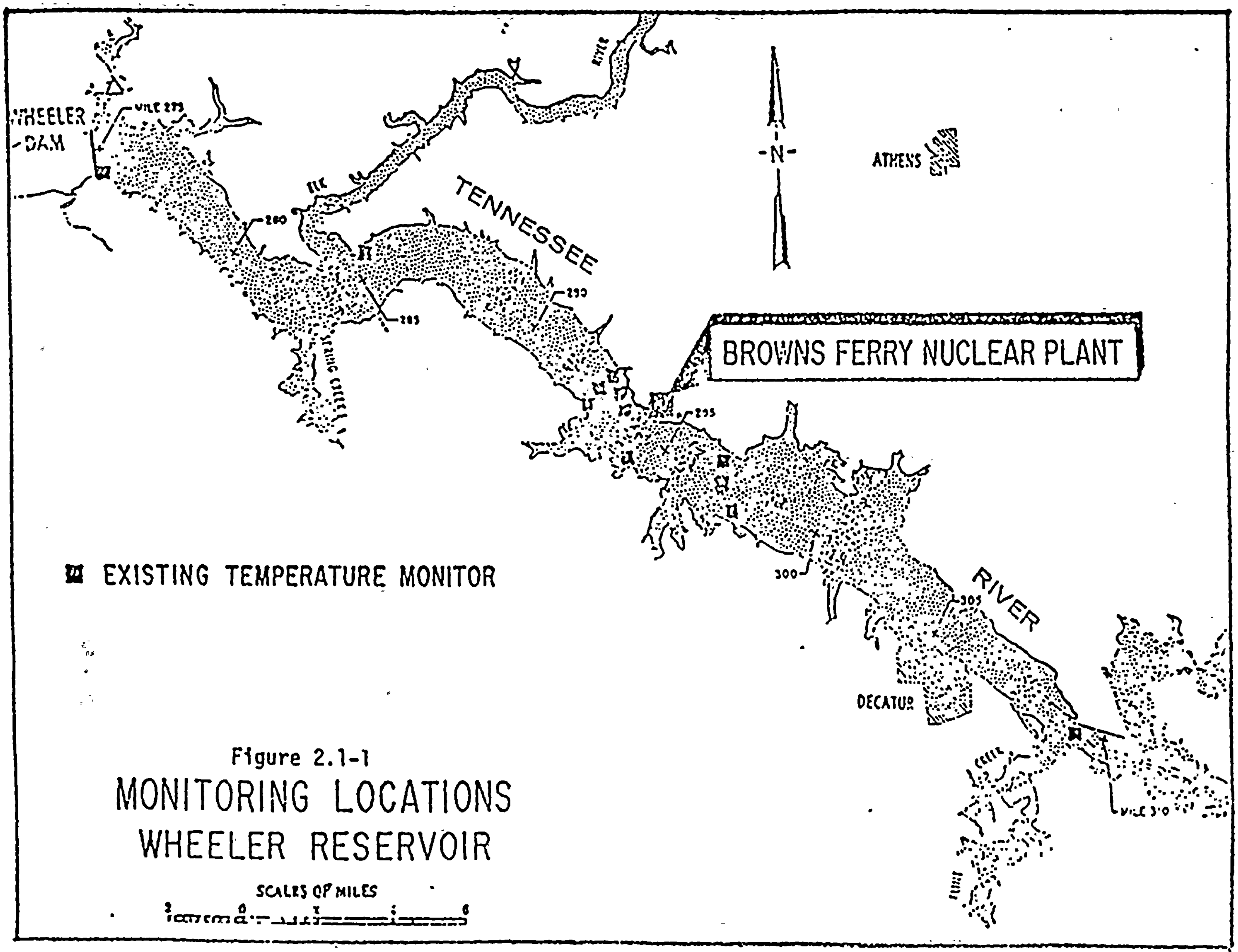
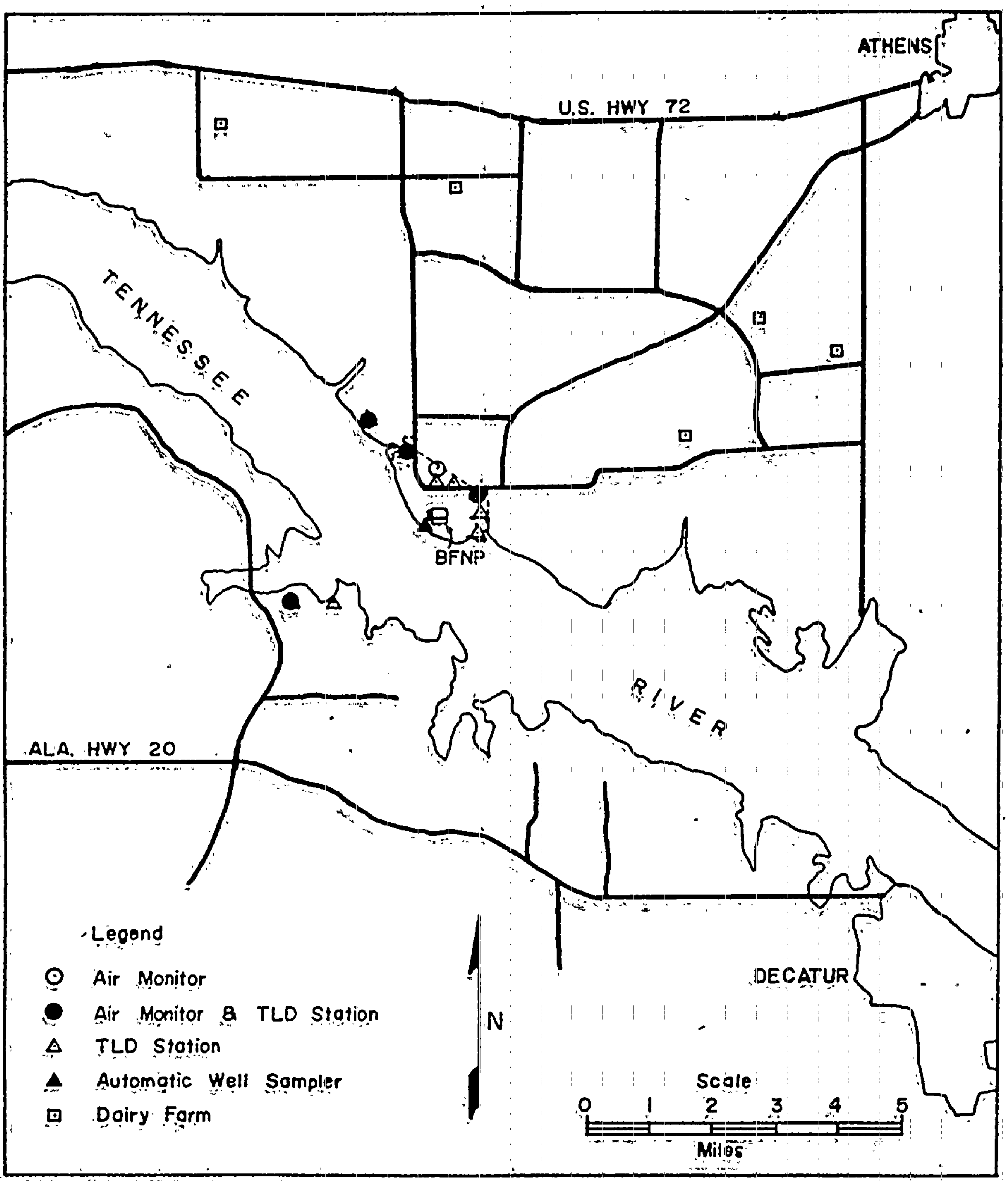


Figure 2.1-1
MONITORING LOCATIONS
WHEELER RESERVOIR

SCALES OF MILES

Figure 4.2-1

LOCAL MONITORING STATIONS
BROWNS FERRY NUCLEAR PLANT



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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 40
License No. DPR-52

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

   A. The applications for amendments by Tennessee Valley Authority (the licensee) dated August 2, 1978 and August 11, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 40, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 16, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 40

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-250

Revise Appendix A as follows:

Remove the following pages and replace with identically numbered pages:

33/34
35/36
51/52
73/74
75/76
113/114
131/132
205/206
207/208
241/242
293/294
303/304

Revise Appendix B as follows:

Remove the following page and replace with identically numbered page:

41/42

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

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Min. No. of Operable Inst. Channels Per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable: Shutdown	Refuel(7)	Startup/Hot Standby	Run	Action(1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤ 120/125 Indicated on scale	X(22)	X (22)	X	(5)	1.A
3	Inoperative			X	X	(5)	1.A
	APRM (16)						
2	High Flux	See Spec. 2.1.A.1				X	1.A or 1.B
2	High Flux	≤ 15% rated power (13)		X(21)	X(17)	(15)	1.A or 1.B
2	Inoperative			X(21)	X(17)	X	1.A or 1.B
2	Downscale	≥ 3 Indicated on Scale		(11)	(11)	X(12)	1.A or 1.B
2	High Reactor Pressure	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14)	≤ 2 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14)	≥ 538" above vessel zero		X	X	X	1.A
2	High Water Level in Scram Discharge Tank	≤ 50 Gallons	X	X(2)	X	X	1.A

TABLE 3.1.A (Continued)

Min. No. of Operable Inst. Channels Per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable: Refuel(7)	Startup/Hot Standby	Run	Action(1)
4	Main Steam Line Isolation Valve Closure	≤ 10% Valve Closure	X(3)(6)	X(3)(6)	X(6)	1.A or 1.C
2	Turbine Cont. Valve Fast Closure	Upon trip of the fast acting solenoid valves	X(4)	X(4)	X(4)	1.A or 1.D
4	Turbine Stop Valve Closure	≤ 10% Valve Closure	X(4)	X(4)	X(4)	1.A or 1.D
2	Turbine Control Valve - Loss of Control Oil Pressure	≥ 550 psig	X(4)	X(4)	X(4)	1.A or 1.D
2	Turbine First Stage Pressure Permissive	≤ 154 psig	X(18)	X(18)	X(18)	(19)
2	Turbine Condenser Low Vacuum	≥ 23 In. Hg. Vacuum	X(3)	X(3)	X	1.A or 1.C
2	Main Steam Line High Radiation (14)	≤ 3X Normal Full Power Background (20)	X(9)	X(9)	X(9)	1.A or 1.C

NOTES FOR TABLE 3.1.A

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable instrument channels per trip system cannot be met for both trip systems, the appropriate actions listed below shall be taken.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

   B. Reduce power level to IRM range and place mode switch in the Startup/Hot Standby position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30% of rated.

2. Scram discharge volume high bypass may be used in shutdown or refuel to bypass scram discharge volume scram with control rod block for reactor protection system reset.

3. Bypassed if reactor pressure < 1055 psig and mode switch not in run.

4. Bypassed when turbine first stage pressure is less than 154 psig.

5. IRM's are bypassed when APRM's are onscale and the reactor mode switch is in the run position.

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   A. Mode switch in shutdown

   B. Manual scram

   C. High flux IRM

   D. Scram discharge volume high level

   E. APRM 15% scram

8. Not required to be operable when primary containment integrity is not required.

9. Not required if all main steamlines are isolated.

head is not bolted to the vessel.

11. The APRM downscale trip function is only active when the reactor mode switch is in run.

12. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.

13. Less than 14 operable LPRM's will cause a trip system trip.

14. Channel shared by Reactor Protection System and Primary Containment and Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

15. The APRM 15% scram is bypassed in the Run Mode.

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system.

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

18. Operability is required when normal first-stage pressure is below 30% ($\leq$ 154 psig).

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required.

20. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in primary coolant.

21. The APRM High Flux and Inoperative Trips do not have to be operable in the Refuel Mode if the Source Range Monitors are connected to give a non-coincidence, High Flux scram, at $\leq 5 \times 10^5$ cps. The SRM's shall be operable per Specification 3.10.B.1. The removal of eight (8) shorting links is required to provide non-coincidence high-flux scram protection from the Source Range Monitors.

22. The three required IRM's per trip channel is not required in the Shutdown or Refuel Modes if at least four IRM's (one in each core quadrant) are connected to give a non-coincidence, High Flux scram. The removal of four (4) shorting links is required to provide non-coincidence high-flux scram protection from the IRM's.

3.2.B Core and Containment Cooling Systems - Initiation & Control

4.2.B Core and Containment Cooling Systems - Initiation & Control

are required to be operable shall be considered operable if they are within the required surveillance testing frequency and there is no reason to suspect that they are inoperable.

C. Control Rod Block Actuation

The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.C.

DELETE
Now covered by note 7.C.

C. Control Rod Block Actuation

Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.C.

System logic shall be functionally tested as indicated in Table 4.2.C.

D. Off-Gas Post Treatment Isolation Function

1. Off Gas Post Treatment Monitors

(a) Except as specified in (b) below, both off-gas post treatment radiation monitors shall be operable during reactor operation. The isolation function trip settings for the monitors shall be set at a value not to exceed the equivalent of the stack release limit specified in specification 3.8.B.1.

D. Off-Gas Post Treatment Isolation Functions

1. Off-Gas Post Treatment Monitoring System

Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.D.

System logic shall be functionally tested as indicated in Table 4.2.D.

3.2.D Off-Gas Post Treatment Isolation Functions

(b) From and after the date that one of the two off-gas post treatment radiation monitors is made or found to be inoperable, continued reactor power operation is permissible during the next seven days, provided that the inoperable monitor is tripped in the downscale position. One radiation monitor may be out of service for four hours for functional test and/or calibration without the monitor being in a downscale tripped condition.

(c) Upon the loss of both off-gas post treatment radiation monitors, initiate an orderly shutdown and shut the mainsteam isolation valves or the off-gas isolation valve within 10 hours.

E. Drywell Leak Detection

The limiting conditions of operation for the instrumentation that monitors drywell leak detection are given in Table 3.2.E.

F. Surveillance Instrumentation

The limiting conditions for the instrumentation that provides surveillance information readouts are given in Table 3.2.F.

G. Control Room Isolation

The limiting conditions for instrumentation that isolates the control room and initiates the control room emergency pressurization systems are given in Table 3.2.G.

4.2.D Off-Gas Post Treatment Isolation Function

E. Drywell Leak Detection

Instrumentation shall be calibrated and checked as indicated in Table 4.2.E.

F. Surveillance Instrumentation

Instrumentation shall be calibrated and checked as indicated in Table 4.2.F.

G. Control Room Isolation

Instrumentation shall be calibrated and checked as indicated in Table 4.2.G.

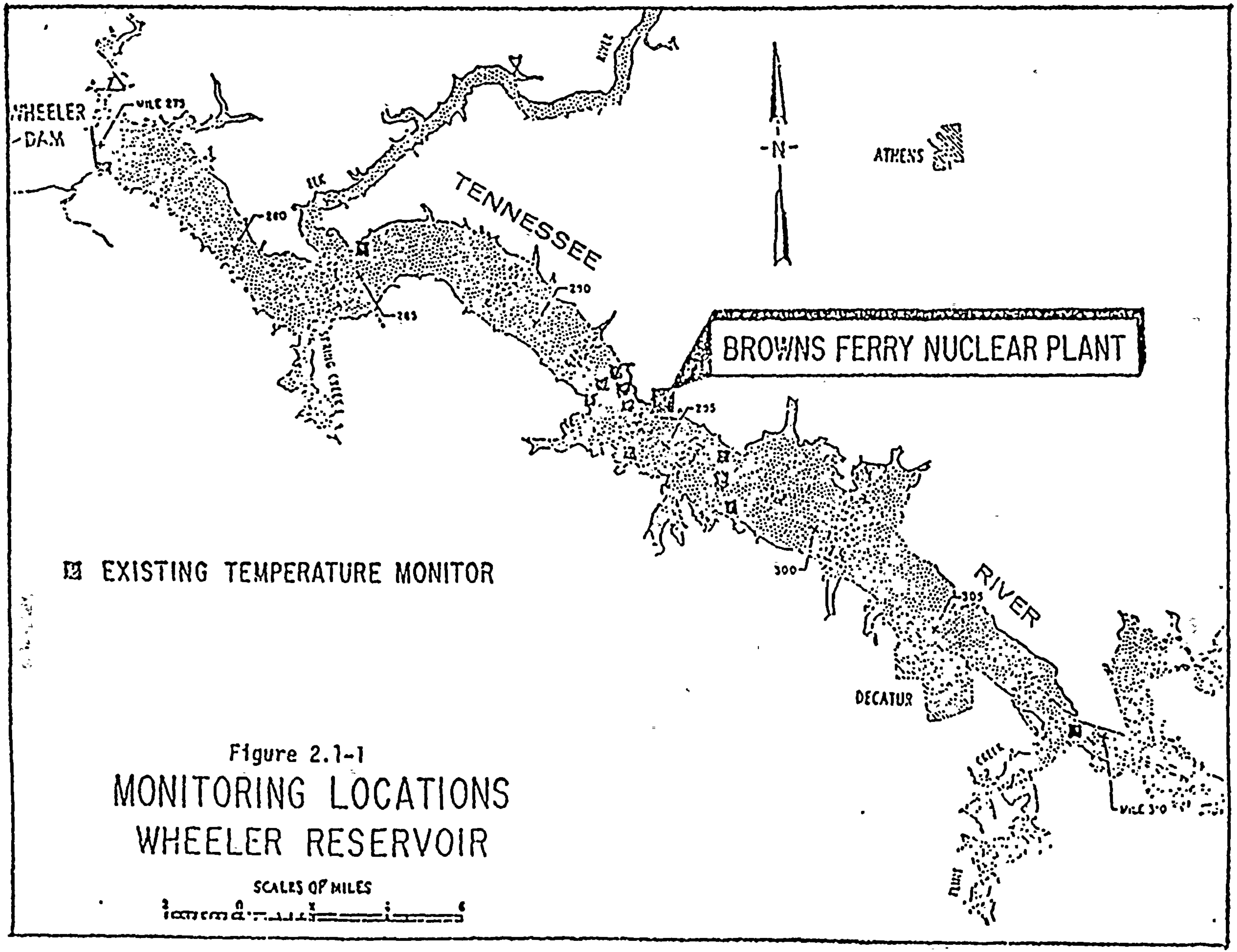
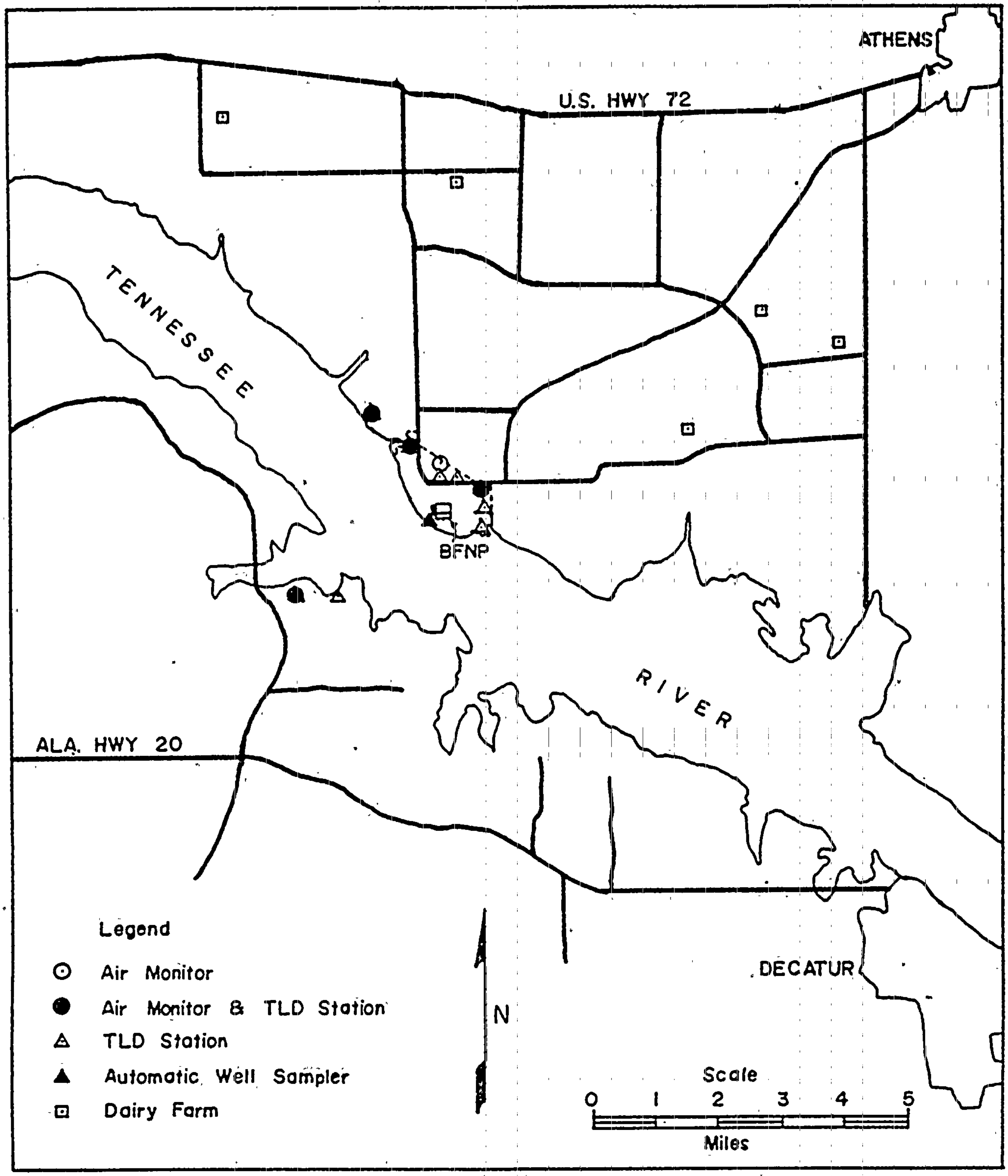


Figure 2.1-1
MONITORING LOCATIONS
WHEELER RESERVOIR

Figure 4.2-1

LOCAL MONITORING STATIONS

BROWNS FERRY NUCLEAR PLANT



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

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

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

   a. SRM and IRM

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

      (2) Power supply voltage low.

      (3) Circuit boards not in circuit.

   b. APRM

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

      (2) Less than 14 LPRM inputs.

      (3) Circuit boards not in circuit.

   c. RBM

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

      (2) Circuit boards not in circuit.

      (3) RBM fails to null.

      (4) Less than required number of LPRM inputs for rod selected.

11. Detector traverse is adjusted to 114 $\pm$ 2 inches, placing the detector lower position 24 inches below the lower core plate.

TABLE 3.2.D
OFF-GAS POST TREATMENT ISOLATION INSTRUMENTATION

Min. No. Operable (1)	Function	Trip Level Setting	Action (2)	Remarks
2	Off-Gas Post Treatment Monitor	Note 3	A or B	1. 2 upscales, or 1 downscale and 1 upscale, or 2 down-scales will isolate off-gas line.
1	Off-Gas Post Treatment Isolation	Note 3	B	1. One trip system with auto transfer to another source

NOTES:

1. Whenever the minimum number operable cannot be met, the indicated action shall be taken.
2. Action
   A. Refer to Section 3.2.D.1.b
   B. Refer to Section 3.2.D.1.c
3. Trip setting to correspond to Specification 3.2.D.1.a

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 1.06. 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. Two RBM channels are provided and only one of these may be bypassed from the console, for maintenance and/or testing, provided that this condition does not last longer than 24 hours in any thirty day period. This time period is only 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

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 1.06.

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.

If the IRM channels are in the worst condition of allowed bypass, the sealing arrangement is such that for unbypassed IRM channels, a rod block signal is generated before the detected neutrons flux has increased by more than a factor of 10.

A downscale indication is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two post treatment off-gas radiation monitors are provided and, when their trip point is reached, cause an isolation of the off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip or both have a downscale trip.

Both instruments are required for trip but the instruments are set so that any instruments are set so that the instantaneous stack release rate limit given in Specification 3.8 is not exceeded.

Four radiation monitors are provided for each unit which initiate Primary Containment Isolation (Group 6 isolation valves) Reactor Building Isolation and operation of the Standby Gas Treatment System. These instrument channels monitor the radiation in the Reactor zone ventilation exhaust ducts and in the Refueling Zone.

Trip setting of 100 mr/hr for the monitors in the Refueling Zone are based upon initiating normal ventilation isolation and SGTS operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the SGTS.

Flow integrators and sump fill rate and pump out rate timers are used to determine leakage in the drywell. A system whereby the time interval to fill a known volume will be utilized to provide a backup. An air sampling system is also provided to detect leakage inside the primary containment (See Table 3.2.E).

3/4.3 BASES:

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 SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit, (ie, MCPR given by Specification 3.5.k or LHGR of 18.5 for 7x7 or 13.4 for 8x8) During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is normally the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform these functions may be designated by the plant superintendent to perform these functions.

Scram Insertion Times

The control rod system is designated to bring the reactor subcritical at the rate fast enough to prevent fuel damage: ie, to prevent the MCPR from becoming less than 1.06. The limiting power transient is given in Reference 1. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification provide the required protection, and MCPR remains greater than 1.06.

On an early BWR, some degradation of control rod scram performance occured during plant startup and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path: i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model drive with a modified (larger screen size) internal filter which is less prone to plugging. Data has been documented by surveillance reports in various operating plants. These include Oyster Creek, Monticello, Dresden 2 and Dresden 3. Approximately 5000 drive tests have been recorded to date.

Following identification of the "plugged filter" problem, very frequent scram tests were necessary to ensure proper performance. However, the more frequent scram tests are now considered totally unnecessary and unwise for the following reasons:

1. Erratic scram performance has been identified as due to an obstructed drive filter in type "A" drives. The drives in BFNP are of the new "B" type design whose scram performance is unaffected by filter condition.

2. The dirt load is primarily released during startup of the reactor when the reactor and its systems are first subjected to flows and pressure and thermal stresses. Special attention and measures are now being taken to assure cleaner systems. Reactors with drives identical or similar (shorter stroke, smaller piston areas) have operated through many refueling cycles with no sudden or erratic changes in scram performance. This preoperational and startup testing is sufficient to detect anomalous drive performance.

3. The 72-hour outage limit which initiated the start of the frequent scram testing is arbitrary, having no logical basis other than quantifying a "major outage" which might reasonably be caused by an event so severe as to possibly affect drive performance. This requirement is unwise because it provides an incentive for shortcut actions to hasten returning "on line" to avoid the additional testing due a 72-hour outage.

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown*	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible Dur Normal Operat
R9 - north	RCIC	564			X	
R9 - south	RCIC (ring hdr)	564			X	
R1 upper	Condensate S&S (ring header)	548				X
R1 lower	Condensate S&S (ring header)	548				X
R2 - north	Condensate S&S (ring header)	548				X
R2 - west	Condensate S&S (ring header)	548				X
R3 - east	Condensate S&S (ring header)	548				X
R3 - west	Condensate S&S (ring header)	548				X
R4 - north	Condensate S&S (ring header)	548		X		X
R4 - east	Condensate S&S (ring header)	548		X		X
R5 upper	Condensate S&S (ring header)	548		X		X
R5 lower	Condensate S&S (ring header)	555		X		X

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers In High Radiation Area During Shutdown*	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SSZ-1	PSC (ring hdr)	525				X
SSX-2	PSC (ring hdr)	525				X
SSX-3	PSC (ring hdr)	525				X
SSZ-4	PSC (ring hdr)	525				X
SSZ-5	PSC (ring hdr)	525				X
SSX-6	PSC (ring hdr)	525				X
SSX-7	PSC (ring hdr)	525				X
SSZ-8	PSC (ring hdr)	525				X
SSZ-1A	PSC (ring hdr)	525				X
SSX-2A	PSC (ring hdr)	525				X
SSX-3A	PSC (ring hdr)	525				X
SSZ-4A	PSC (ring hdr)	525				X
SSZ-5A	PSC (ring hdr)	525				X
SSX-6A	PSC (ring hdr)	525				X
SSX-7A	PSC (ring hdr)	525				X
SSZ-8A	PSC (ring hdr)	525				X

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown*	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
R33	EECW	605				X
R1 upper	RBCCW	615				X
R1 lower	RBCCW	615				X
R2 upper	RBCCW	615				X
R2 lower	RBCCW	615				X
R3 upper	RBCCW	615				X
R3 lower	RBCCW	615				X
R4 upper	RBCCW	615				X
R4 lower	RBCCW	615				X
SS1-A	Recirculation	556			X	
SS1-B	Recirculation	556			X	
SS2-A	Recirculation	558			X	

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown*	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SS2-B	Recirculation	558			X	
SS3-A(295°)	Recirculation	564			X	
SS3-A(335°)	Recirculation	564			X	
SS3-B(115°)	Recirculation	564			X	
SS3-B(154°)	Recirculation	564			X	
SS4-A	Recirculation	570			X	
SS4-B	Recirculation	570			X	
SS5-A(262°)	Recirculation	581			X	
SS5-A(325°)	Recirculation	581			X	
SS5-B(35°)	Recirculation	581			X	
SS5-B(98°)	Recirculation	581			X	
SS6-A	Recirculation	568			X	
SS6-B	Recirculation	568			X	
SS7	Recirculation	564			X	
SS8	Recirculation	564			X	

*Modifications to this Table due to changes in high radiation areas should be submitted to the NRC as part of the next license amendment.

3.7.C Secondary Containment

2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:

   a. The reactor shall be made subcritical and Specification 3.3.A shall be met.

   b. The reactor shall be cooled down below 212°F and the reactor coolant system vented.

   c. Fuel movement shall not be permitted in the reactor zone.

   d. Primary containment integrity maintained.

3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.

4.7.C Secondary Containment

capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system inleakage rate of not more than 12,000 cfm.

b. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system inleakage rate of not more than 12,000 cfm. shall be demonstrated at each refueling outage prior to refueling.

2. After a secondary containment violation is determined the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

7. **Secondary Containment**

4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:

   a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.

   b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones.

Primary Containment Isolation Valves

1. During reactor power operation, all isolation valves listed in Table 3.7.A and all reactor coolant system instrument line flow check valves shall be operable except as specified in 3.7.D.2.

4.7.C **Secondary Containment**

D. **Primary Containment Isolation Valves**

1. The primary containment isolation valves surveillance shall be performed as follows:

   a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

   b. At least once per quarter:

      (1) All normally open power operated isolation valves (except for the main steam line power-operated isolation valves) shall be fully closed and reopened.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9.A Auxiliary Electrical Equipment	4.9.A Auxiliary Electrical Equipment

common transformer and cooling tower transformer capable of supplying power to the shutdown boards.

b. A fourth operable units 1 and 2 diesel generator.

4. Buses and Boards Available

a. Start buses 1A and 1B are energized.

b. The units 1 and 2 4-kV shutdown boards are energized.

c. The 480-V shutdown boards associated with the unit are energized.

d. Undervoltage relays operable on start buses 1A and 1B and 4-kV shutdown boards, A, B, C, and D.

5. The 250-Volt unit and shutdown board batteries and a battery charger for each battery and associated battery boards are operable.

6. Logic Systems

a. Common accident signal logic system is operable.

b. 480-V load shedding logic system is operable.

7. There shall be a minimum of 103,300 gallons of diesel fuel in the standby diesel generator fuel tanks.

the specified time sequence.

c. Once a month the quantity of diesel fuel available shall be logged.

d. Each diesel generator shall be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.

e. Once a month a sample of diesel fuel shall be checked for quality. The quality shall be within the acceptable limits specified in Table 1 of the latest revision to ASTM D975 and logged.

2. D.C. Power System - Unit Batteries (250-Volt) Diesel Generator Batteries (125-Volt) and Shutdown Board Batteries (250-Volt)

a. Every week the specific gravity and the voltage of the pilot cell, and temperature of an adjacent cell and overall battery voltage shall be measured and logged.

b. Every three months the measurements shall be made of voltage of each cell to nearest 0.1 volt, specific gravity of each cell, and temperature of every fifth cell. These measurements shall be logged.

c. A battery rated discharge (capacity) test shall be performed and the voltage, time, and cutout current measurements shall be logged at intervals not to exceed 24 months.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9.A Auxiliary Electrical Equipment	4.9.A Auxiliary Electrical Equipment

3. Logic System

a. Both divisions of the common accident signal logic system shall be tested every 6 months to demonstrate that it will function on actuation of the core spray system of each reactor to provide an automatic start signal to all 4 units 1 and 2 diesel generators.

b. Once every 6 months, the condition under which the 480-Volt load shedding logic system is required shall be simulated using pendant test switches and/or pushbutton test switches to demonstrate that the load shedding logic system would initiate load shedding signals on the diesel auxiliary boards, reactor MOV boards, and the 480-Volt shutdown boards.

4. Undervoltage Relays

a. Once every 6 months, the condition under which the undervoltage relays are required shall be simulated with an undervoltage on start buses 1A and 1B to demonstrate that the diesel generators will start.

b. Once every 6 months, the conditions under which the undervoltage relays are required shall be simulated with an undervoltage on each shutdown board to demonstrate that the associated diesel generator will start.

c. The undervoltage relays which start the diesel generators from start buses 1A and 1B and the 4-kV shutdown boards, shall be calibrated annually for trip and reset and the measurements logged.

3.10.A Refueling Interlocks

3. The fuel grapple hoist load switch shall be set at $\leq$ 1,000 lbs.

4. If the frame-mounted auxiliary hoist, the monorail-mounted auxiliary hoist, or the service platform hoist is to be used for handling fuel with the head off the reactor vessel, the load limit switch on the hoist to be used shall be set at $<$ 400 lbs.

5. A maximum of two non-adjacent control rods may be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance, provided the following conditions are satisfied:

   a. The reactor mode switch shall be locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being performed. All other

4.10.A Refueling Interlocks

control rods are fully inserted and have had their directional control valves electrically disarmed, it is sufficient to demonstrate that the core is subcritical with a margin of at least 0.38 Δk at any time during the maintenance. A control rod on which maintenance is being performed shall be considered inoperable.

3.10.A Refueling Interlocks

refueling interlocks shall be operable.

b. A sufficient number of control rods shall be operable so that the core can be made subcritical with the strongest operable control rod fully withdrawn and all other operable control rods fully inserted, or all directional control valves for remaining control rods shall be disarmed electrically and sufficient margin to criticality shall be demonstrated.

c. If maintenance is to be performed on two control rod drives they must be separated by more than two control cells in any direction.

d. An appropriate number of SRM's are available as defined in specification 3.10.A.

6. Any number of control rods may be withdrawn or removed from the reactor core providing the following conditions are satisfied:

a. The reactor mode switch is locked in the "refuel" position. The refueling interlock which prevents more than one control rod from

4.10.A Refueling Interlocks

3. With the mode selector switch in the refuel or shutdown mode, no control rod may be withdrawn until two licensed operators have confirmed that either all fuel has been removed from around that rod or that all control rods in immediately adjacent cells have been fully inserted and electrically disarmed.

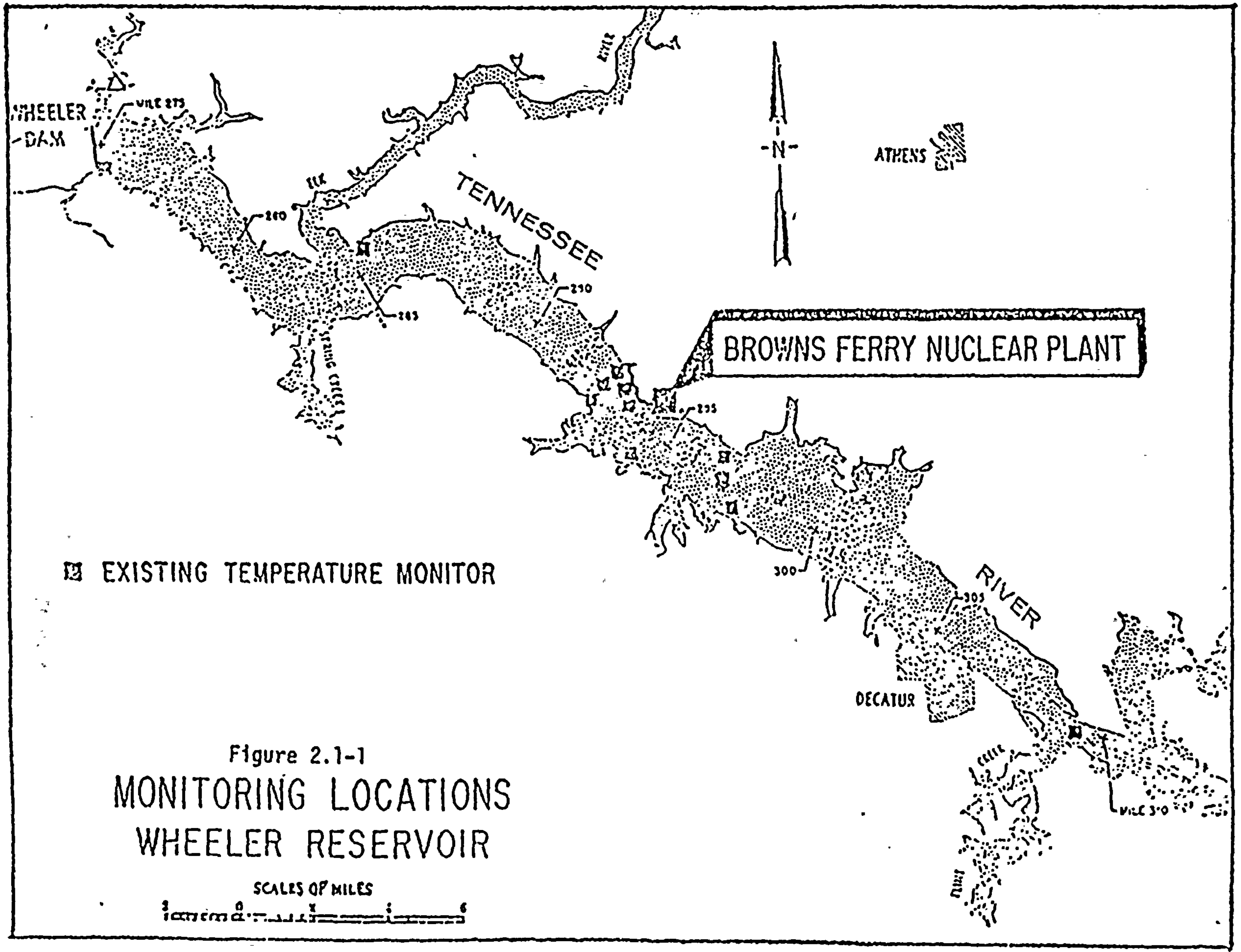
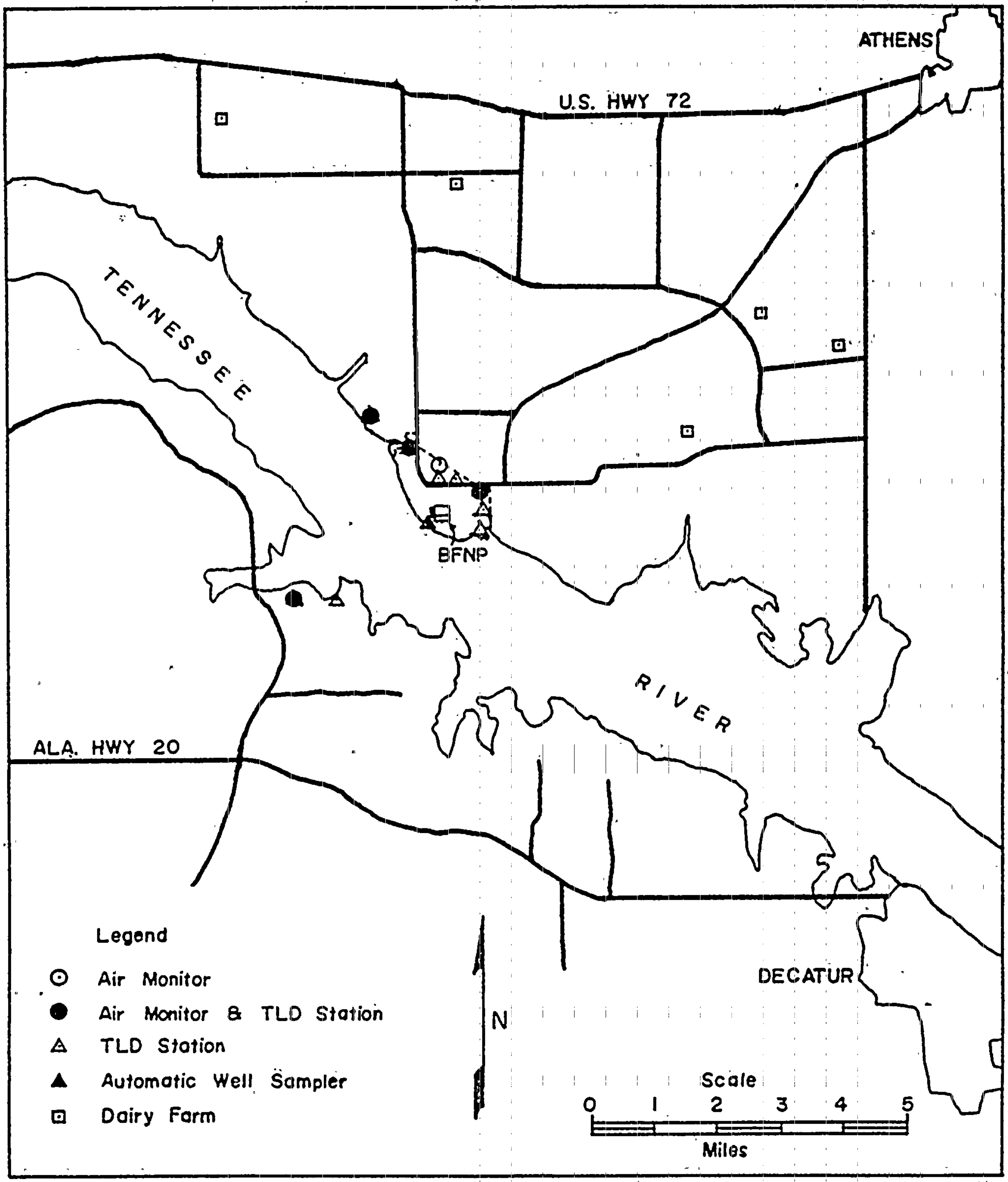


Figure 2.1-1
MONITORING LOCATIONS
WHEELER RESERVOIR

Figure 4.2-1

LOCAL MONITORING STATIONS

BROWNS FERRY NUCLEAR PLANT



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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 17
License No. DPR-68

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

   A. The applications for amendments by Tennessee Valley Authority (the licensee) dated August 2, 1978 and August 11, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility License No. DPR-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. 17, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 16, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 17

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

Remove the following pages and replace with identically numbered pages:

32
35
50
77
78
110
134
218
251
252
318
335

Revise Appendix B as follows:

Remove the following page and replace with identically numbered page:

42

Marginal lines indicate changed areas.

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Min. No. of Operable Inst. Channels Per Trip System (1)	Trip Function	Trip Level Setting	Shut-down	Modes in Which Function Must Be Operable Refuel (7)	Startup/Hot Standby	Run	Action(1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤ 120/125 Indicated on scale	X(22)	X (22)	X	(5)	1.A
3	Inoperative			X	X	(5)	1.A
	APRM (16)						
2	High Flux	See Spec. 2.1.A.1				X	1.A or 1.B
2	High Flux	≤ 15% rated power		X (21)	X(17)	(15)	1.A or 1.B
2	Inoperative	(13)		X (21)	X(17)	X	1.A or 1.B
2	Downscale	≥ 3 Indicated on Scale		(11)	(11)	X(12)	1.A or 1.B
2	High Reactor Pressure	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14)	≤ 2 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14)	≥ 538" above vessel zero		X	X	X	1.A
2	High Water Level in Scram Discharge Tank	≤ 50 Gallons	X	X(2)	X	X	1.A
4	Main Steam Line Isolation Valve Closure	≤ 10% Valve Closure		X(3) (6)	X(3) (6)	X(6)	1.A or 1.C
2	Turbine Cont. Valve Fast Closure	Upon trip of the fast acting solenoid valves		X(4)	X(4)	X(4)	1.A or 1.D

12. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.

13. Less than 14 operable LPRM's will cause a trip system trip.

14. Channel shared by Reactor Protection System and Primary Containment and Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

15. The APRM 15% scram is bypassed in the Run Mode.

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system.

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

18. Operability is required when reactor thermal power is below 30% (high-pressure turbine first-stage pressure (≤ 154 psig).

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required.

20. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in the primary coolant.

21. The APRM High Flux and Inoperative Trips do not have to be operable in the Refuel Mode if the Source Range Monitors are connected to give a non-coincidence, High Flux scram, at $\leq 5 \times 10^5$ cps. The SRM's shall be operable per Specification 3.10.B.1. The removal of eight (8) shorting links is required to provide non-coincidence high-flux scram protection from the Source Range Monitors.

22. The three required IRM's per trip channel is not required in the Shutdown or Refuel Modes if at least four IRM's (one in each core quadrant) are connected to give a non-coincidence, High Flux scram. The removal of four (4) shorting links is required to provide non-coincidence high-flux scram protection from the IRM's.

3.2 PROTECTIVE INSTRUMENTATION

B. Core and Containment Cooling Systems - Initiation & Control

The limiting conditions for operation for the instrumentation that initiates or controls the core and containment cooling systems are given in Table 3.2.B. This instrumentation must be operable when the system(s) it initiates or controls are required to be operable as specified in Section 3.5.

C. Control Rod Block Actuation

The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.C.

DELETE
Now covered by Note 7.C.

4.2 PROTECTIVE INSTRUMENTATION

B. Core and Containment Cooling Systems - Initiation & Control

Instrumentation shall be functionally tested, claibrated and checked as indicated in Table 4.2.B.

System logic shall be functionally tested as indicated in Table 4.2.B.

Whenever a system or loop is made inoperable because of a required test or calibration, the other systems or loops that are required to be operable shall be considered operable if they are within the required surveillance testing frequency and there is no reason to suspect that they are inoperable.

C. Control Rod Block Actuation

Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.C.

System logic shall be functionally tested as indicated in Table 4.2.C.

NOTES FOR TABLE 3.2.C

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

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

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

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

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

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

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

7. The following operational restraints apply to the RBM only:
   a. Both RBM channels are bypassed when reactor power is ≤ 30%.
   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. An RBM channel may be out of service for testing and/or maintenance provided this condition does not last longer than 24 hours in any thirty day period.
   d. If minimum conditions for Table 3.2.C are not met, administrative controls shall be immediately imposed to prevent control rod withdrawal.

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

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

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

    a. SRM and IRM

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

        (2) Power supply voltage low.

        (3) Circuit boards not in circuit.

    b. APRM

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

        (2) Less than 14 LPRM inputs.

        (3) Circuit boards not in circuit.

    c. RBM

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

        (2) Circuit boards not in circuit.

        (3) RBM fails to null.

        (4) Less than required number of LPRM inputs for rod selected.

11. Detector traverse is adjusted to 114 ± 2 inches, placing the detector lower position 24 inches below the lower core plate.

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 1.05. 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. Two RBM channels are provided and only one of these may be bypassed from the console, for maintenance and/or testing provided that this condition does not last longer than 24 hours in any thirty day period. This time period is only 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

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

of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (i.e., MCPR -1.27 or LHGR = 13.4). During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is normally the responsibility of the Nuclear Engineer to identify these limting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform these functions may be designated by the plant superintendent to perform these functions.

C. <u>Scram Insertion Times</u>

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.05. The limiting power transient is that resulting from that of Rod Withdrawal Error (RWE).

Analysis of this transient shows that the negative reactivity rates resulting from the scram (FSAR Figure N3.6-9) with the average response of all the drives as given in the above specification, provide the required protection, and MCPR remains greater than 1.05.

On an early BWR, some degradation of control rod scram performance occurred during plant startup and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

TABLE 3.6.H

SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	System	Elevation	Snubbers in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SSX-7A	PSC (ring hdr)	525				X
SSZ-8A	PSC (ring hdr)	525				X
R24	EECW	605			X	
SS1-A	Recirculation	556			X	
SS1-B	Recirculation	556			X	
SS2-A	Recirculation	558			X	
SS2-B	Recirculation	558			X	
SS3-A(295°)	Recirculation	564			X	

3.7 CONTAINMENT SYSTEMS

C. Secondary Containment

1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.

4.7 CONTAINMENT SYSTEMS

C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

   a. A preoperational secondary containment capability test shall be conducted by isolating the reactor building and placing two standby gas treatment system filter trains in operation. Such test shall demonstrate the capability to maintain 1/4 inch of water vacuum under calm wind (<5 mph) conditions with a system inleakage rate of not more than 12,000 cfm.

3.7 CONTAINMENT SYSTEMS

2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:

a. The reactor shall be made subcritical and Specification 3.3.A shall be met.

b. The reactor shall be cooled down below 212°F and the reactor coolant system vented.

c. Fuel movement shall not be permitted in the reactor zone.

d. Primary containment integrity maintained.

4.7 CONTAINMENT SYSTEMS

b. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (<5 mph) conditions with a system inleakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.

2. After a secondary containment violation is determined the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

3.9 AUXILIARY ELECTRICAL SYSTEM

2. Three unit 3 diesel generators shall be operable.

4.9 AUXILIARY ELECTRICAL SYSTEM

d. Each diesel generator shall be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.

e. Once a month a sample of diesel fuel shall be checked for quality. The quality shall be within the acceptable limits specified in Table 1 of the latest revision to ASTM D975 and logged.

2. D.C. Power System - Unit Batteries (250-Volt) and Diesel Generator Batteries (125-Volt)

   a. Every week the specific gravity and the voltage of the pilot cell, and temperature of an adjacent cell and overall battery voltage shall be measured and logged.

3.10 CORE ALTERATIONS

6. Any number of control rods may be withdrawn or removed from the reactor core providing the following conditions are satisfied: . .

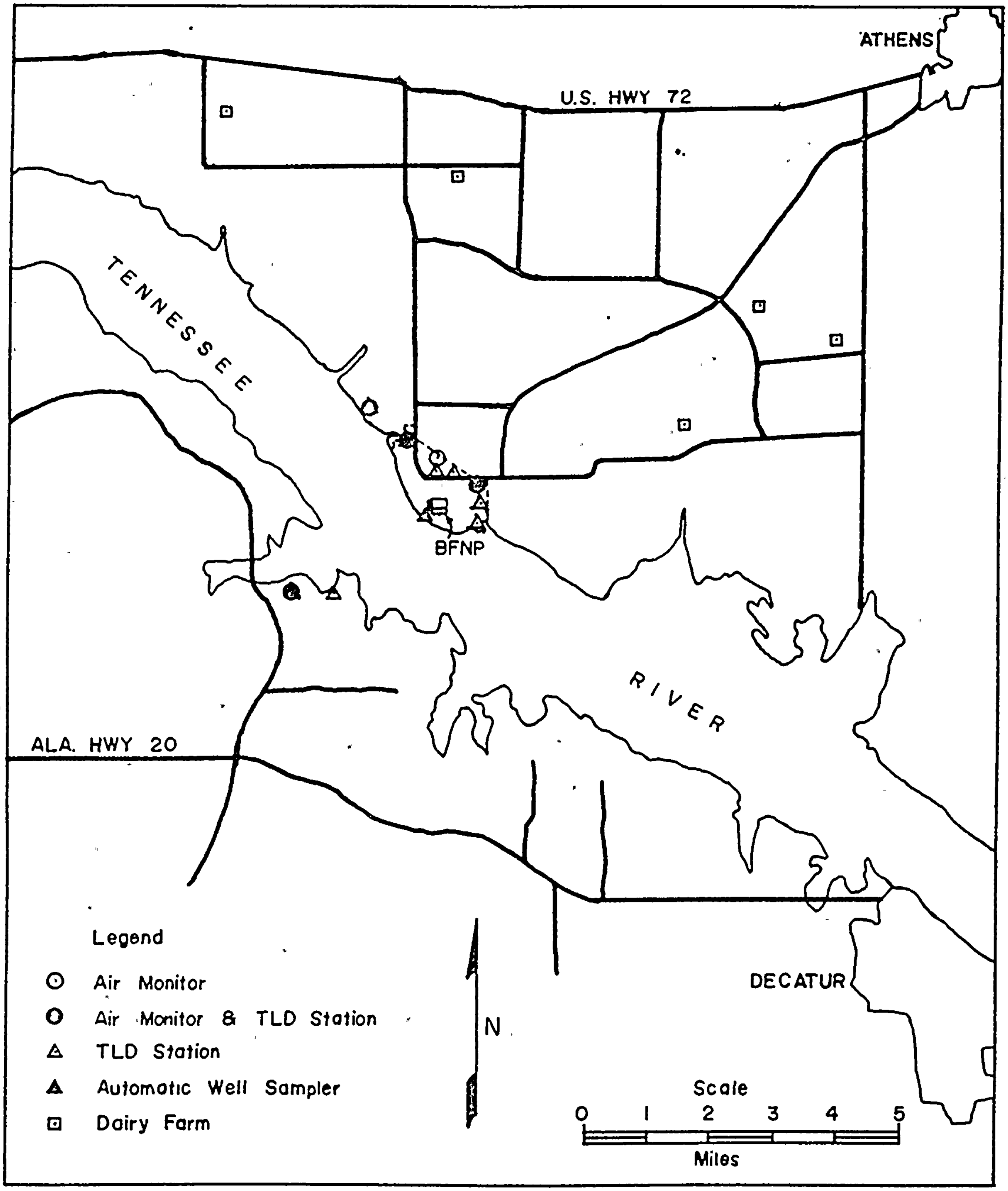
   a. The reactor mode switch is locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.

4.10 CORE ALTERATIONS

3. With the mode selector switch in the refuel or shutdown mode, no control rod may be withdrawn until two licensed operators have confirmed that either all fuel has been removed from around that rod or that all control rods in immediately adjacent cells have been fully inserted and electrically disarmed.

Figure 4.2-1

LOCAL MONITORING STATIONS
BROWNS FERRY NUCLEAR PLANT



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 17 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

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

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

1.0 Introduction

By letter dated August 11, 1978 (TVA BFNP TS 114), the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. The proposed amendments and revised Technical Specifications would (1) permit the average power range monitor (APRM) system to be inoperable in the refuel mode, provided the source range monitors (SRMs) are connected to give a non-coincidence, high flux scram and (2) in the refuel and shutdown modes only, permit less than three intermediate range monitors (IRMs) per trip channel to be operable-provided at least four IRMs (one in each core quadrant) are connected to give a non-coincidence, high flux scram. The present Technical Specifications require that a minimum of three IRMs per trip channel be operable at all times (i.e., shutdown as well as startup and operation).

The reason for this request is to allow the interchange of the fission chambers in the current APRM system with reduced radiation exposure to the operating personnel and with reduced handling and movement of fuel. This can be achieved by removing many LPRMs simmultaneously rather than in sequence. The sequential removal would leave the APRM system operable but the simultaneous removal would not.

In a separate letter dated August 2, 1978 (TVA BFNP TS 112), TVA requested five changes to the Technical Specifications, all of which are administrative in nature. The changes would: (1) clarify an ambiguious portion of the Technical Specifications related to the rod block monitor system, (2) remove reference to an obsolete 1968 version of an ASTM procedure, (3) modify the list of snubbers that are required to be operable, (4) change one of the four locations from which milk samples are routinely collected and (5) remove a specification for additional test of secondary containment that only applied to the first operating cycle for each Browns Ferry unit.

2.0 Discussion

As described in Section 7.5 of the Final Safety Analysis Report (FSAR) for the Browns Ferry Nuclear Plant (BFNP), the Neutron Monitoring System consists of six major subsystems: (a) the Source Range Monitor (SRM) subsystem, (b) the Intermediate Range Monitor (IRM) subsystem, (c) the Local Power Range Monitor (LPRM) subsystem, (d) the Average Power Range Monitor (APRM) subsystem, (e) the Rod Block Monitor (RBM) subsystem and (f) the Traversing In-Core Probe (TIP) subsystem. The IRM subsystem monitors neutron flux from the upper portion of the SRM range to the lower portion of the Power Range Monitoring Subsystems.

The IRM system normally consists of eight moveable miniature chambers with two such chambers in each core quadrant. No more than one of the IRMs in each quadrant may be bypassed. The eight IRM channels are divided into two IRM sub-systems and at least one IRM from each sub-system must reach 120/125 of full scale to initiate a reactor scram. The IRM system is nominally designed for protection in the startup mode and analyses (FSAR, Section 14.5.3) have been performed showing that the system adequately prevents fuel damage due to rod withdrawal errors postulated to occur during startup.

The APRM subsystem provides a continuous indication of average reactor power from a few percent to 125% of rated reactor power. The subsystem has six APRM channels, each of which uses input signals from a number of LPRM channels. Three APRM channels are associated with each of the trip systems of the Reactor Protection System.

The APRM system which consists of a number of stationary fission chambers dispersed throughout the core, is normally required to be operable in the refuel mode with a high flux scram setpoint corresponding to 15% rated power.

Because the APRM response is actually the combined response of a number of individual fission chambers located throughout the core, the APRM primarily provides protection for core-wide transient power increases which might occur in the run mode (above 15% rated power). Also, in the startup mode the APRM provides backup protection to the IRM system against localized power increases which might result from postulated rod withdrawal errors.

Although the IRM system as described above is required by the current Technical Specifications to be operable in both the shutdown and refuel modes, no specific event has been analyzed in the Plant FSAR which takes credit for a scram initiated by the IRM system with a given setpoint or number of bypassed instruments. Similarly, the APRM is required to operate normally in the refuel mode, but no transient or accident taking credit for an APRM initiated scram, and postulated to occur in the refuel mode has been analyzed in the Plant FSAR. As discussed in the evaluation which follows, there is only one event which the staff can postulate - namely, an operator bypassing the interlocks and withdrawing a second control rod adjacent to one which is already withdrawn - for which the IRM/APRM subsystems are required to provide safety protection in the refuel and shutdown modes.

Section 14.5.3 of the Browns Ferry FSAR discusses the events that could result directly in positive reactivity insertions, including control rod removal error during refueling and fuel assembly insertion error during refueling. Section 7.6 of the FSAR describes the refueling interlocks that prevent an inadvertent criticality during refueling operations and that are designed to back up procedural core reactivity controls during refueling operations. Section 3.10 of the Browns Ferry Nuclear Plant Technical Specifications lists the restrictions that apply during core alterations to ensure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.

When the mode switch is in REFUEL, only one control rod can be withdrawn. Selection of a second rod initiates a rod block thereby preventing the withdrawal of more than one rod at a time. The Refueling Interlocks, in combination with core nuclear design and refueling procedures, prevent inadvertent criticality. The nuclear characteristics of the core assure that the reactor is subcritical even when the highest worth control rod is fully withdrawn. Refueling procedures are written to avoid situations in which inadvertent criticality is possible. The combination of refueling interlocks for control rods and the refueling platform provide redundant methods of preventing inadvertent criticality even after procedural violations when the mode switch is in REFUEL position. The interlocks on hoists provide yet another method of avoiding inadvertent criticality.

During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time. The maintenance is performed with the mode switch in the "refuel" position to provide the refueling interlocks normally available during refueling operations. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the refueling interlock on the first control rod which prevents more than one control rod from being withdrawn at the same time. The present Technical Specifications permit bypassing the refueling interlock with the requirement that an adequate shutdown margin be demonstrated or that all remaining control rods have their directional control valves electrically disarmed to ensure that inadvertent criticality cannot occur during this maintenance. The adequacy of the shutdown margin is verified by demonstrating that the core is shut down by a margin of 0.38 percent Δk with the strongest operable control rod fully withdrawn, or that at least 0.38% Δk shutdown margin is available if the remaining control rods have had their directional control valves disarmed. Disarming the directional control valves does not inhibit control rod scram capability.

3.0 Evaluation

3.1 APRM-IRM Systems

We have reviewed the plant Technical Specifications and the nuclear design characteristics of the fuel. We have concluded that a local criticality during shutdown or refueling operations could only occur through violation of technical specifications such as an operator error in withdrawing a control rod for maintenance, adjacent to a previously withdrawn rod.

Although such operator errors are not likely to occur, they are not impossible. We have therefore considered the applicant's request for proposed modifications to the SRM, IRM and APRM systems in terms of the impact on the protection against postulated local criticality which could occur while the mode selection switch is in the refuel or shutdown positions.

The most severe test of the adequacy of the modified IRM and SRM systems would be the withdrawal (for maintenance) of a control rod near the edge of the reactor core face adjacent to a previously withdrawn rod. Because the proposed Technical Specifications allow one IRM in each core quadrant to be bypassed, the IRM nearest the pair of withdrawn rods was assumed to be bypassed.

Because the modified IRM system would initiate a reactor scram when any IRM reaches the trip set point, the modified system will actuate a scram at an earlier time during the withdrawal of the second rod than would the normal system. The normal system would require trips in each IRM subsystem.

We conclude that the redundant independent. IRM instruments connected to give non-coincident scrams provide better protection against fuel damage due to a localized power increase than does the APRM system with its 15% scram setpoint. Beacuse the IRM instruments are independent in the modified IRM system, the IRM will be its own backup. The IRM scram setpoint will be 120/125 of the lowest IRM scale which corresponds to very low flux levels. Although the flux level at the second nearest IRM (the backup IRM) would be low throughout the rod withdrawal event, it will be high enough to scram the reactor at a lower flux level than with the present arrangement using the APRM monitors. We therefore, conclude that the licensee's proposal for the IRM system modification results in a system that is more sensitive to possible operator errors during core modifications than is the present arrangement and therefore the proposed modification is acceptable.

In addition, the SRM system would be connected to scram the reactor at a level of 5 x 10^5 counts per second. Although the SRM is not considered safety grade equipment, the licensee has proposed to provide the SRM scram function, and we believe this is desirable as an additional backup to the IRM system.

A concern which was raised during the NRC review was what technique(s) will be provided to assure that the reconfiguration of the SRM's and IRM's to the non-coincidence trip mode is in fact accomplished prior to removing the APRM protection. By letter dated November 13, 1978, the licensee has agreed to the following administrative controls. The procedures related to maintenance of detectors ("Browns Ferry Nuclear Plant-Instrument Maintenance Instructions") will be reviewed, and revised as necessary, to include: (1) a specific reference to the Technical Specification Table 3.1.A and associated Notes 21 and 22, which indicate that the SRM's/IRM's must be re-configured to provide non-coincidence high flux scram protection, and (2) a specific procedural step which requires that verification will be made that the appropriate shorting links have been removed prior to maintenance on IRM/LPRM detectors. These controls provide adequate assurance that the reconfiguration of the SRMs and IRMs will be accomplished prior to removing the APRM protection.

Due to the interwoven design of the shorting link system, clarification of the notes to Table 3.1.A is needed. The following sentence should be added to Note 21: "The removal of eight (8) shorting links is required to provide non-coincidence high-flux scram protection from the Source Range Monitors". The following sentence should be added to Note 22: "The removal of four (4) shorting links is required to provide non-coincidence high-flux scram protection from the IRM's".

As is proposed by the licensee for Unit No. 3, the Technical Specifications for Units Nos. 1 and 2 should include in Note 21 to Table 3.1.A that the scram setpoint is $\leq 5 \times 10^5$ CPS.

To summarize, we find that the modification TVA has proposed for the Browns Ferry IRM systems is acceptable. The modified systems will be more sensitive to the flux perturbations resulting from the worst postulated transient than the present arrangement. Furthermore, as discussed previously, the redundant and independent IRM instruments which will comprise the modified IRM systems will provide protection against inadvertent criticality in the refuel mode equivalent to or better than the present APRM system. Inoperability of the APRM with the modified IRM in place is therefore acceptable for the refuel mode.

As described in the "Discussion" above, Section 3.10 of the Technical Specifications includes restrictions on withdrawal of control rods during core alterations. As an additional backup to the neutron monitoring instrumentation, we have proposed, and the licensee has accepted, an addition to the surveillance requirements in Section 4.10 of the Technical Specifications to require that no control rod may be withdrawn for maintenance until two licensed operators have confirmed that there is no fuel in the cell controlled by the particular control rod or that all immediately adjacent control rods are fully inserted and electrically disarmed. This requirement, in conjunction with the more sensitive IRM system, will insure that there is no possibility of inadvertent criticality during core modifications.

In summary we conclude that the proposed changes to the licensee's Technical Specifications do not involve an increase in the probability of a transient or accident but in fact should reduce the consequences of such events. The proposed changes do not involve a reduction in safety margin. No change in a safety limit or a safety limit margin is involved. We therefore conclude that the proposed changes to the Browns Ferry Technical Specifications with respect to the APRM and IRM systems are acceptable and do not involve a significant hazards consideration.

3.2 Snubbers

Table 3.6.H of the Browns Ferry Technical Specifications contains a list of "Shock Suppressors (snubbers)" that are required to be operable to protect the primary coolant system or other safety related components. Section 3.6.H.6 of the Technical Specifications states that: "Snubbers may be added to safety-related systems without prior license amendment to Table 3.6.H provided that a revision to Table 3.6.H is included with a subsequent license amendment request". TVA proposes to add three snubbers to Table 3.6.H on the Fire Protection System. They also propose to delete the two snubbers that were formerly on the control rod drive (CRD) line since the CRD return line has been capped at the reactor vessel and rerouted to the reactor water cleanup return line as part of the modifications to reduce the potential for cracking in the CRD return line. The line-and thus the snubbers-are no longer present in the system. TVA also proposes to delete four snubbers from Table 3.6.H on the condensate bypass line, since this line is a non-critical system (i.e., not classified as a safety-related system) and failure of this by-pass line will not cause damage to a critical system. We conclude that the proposed changes to Table 3.6.H are acceptable.

3.3 ASTM Procedure

Section 4.9.A.3 of the Technical Specifications requires that a sample of diesel fuel shall be analyzed once a month and that the quality shall be within the acceptable limits specified in an obsolete 1968 version of ASTM procedure D975. This ASTM procedure is under revision. In lieu of referring to the specific version of the ASTM procedure (which is subject to the periodic revisions) TVA has proposed to change the Technical Specifications to read: "The quality shall be within the acceptable limits specified in Table 1 of the latest revision to ASTM D975 and logged". Since the most recent revision to this standard method of analysis reflects the current best judgement of the country's experts who are on the various ASTM committees, the most recent edition of the standard is the one that should be used as the "referee method" rather than the edition in effect when the plant was under construction. We conclude that the proposed change to the Technical Specification is acceptable.

3.4 Rod Block Monitors

Control rod block functions are provided to prevent excessive control rod withdrawal so that the safety limit minimum critical power ratio is not violated. Two rod block monitor (RBM) channels are provided. The current Technical Specifications and the Bases therefore (Section 3.2.C.2) state that: "The minimum number of operable instrument channels specified in Table 3.2.C for the Rod Block Monitor may be reduced by one in one of the trip systems for maintenance and/or testing, provided that this condition does not last longer than 24 hours in any thirty day period". TVA proposes to relocate this requirement in the Technical Specifications, adding it as part of "Note 7" to Table 3.2.C and rewording it to be more specific. The revised wording will be: "Two RBM channels are provided and only one of these may be out of service for testing and/or maintenance provided this condition does not last longer than 24 hours in any thirty day period". This is not a change to the requirements in the Technical Specifications but simply a change in wording of the requirement and its location in the Technical Specifications. We conclude that the proposed action is an improvement in phraseology and is acceptable.

3.5 Secondary Containment Testing

Section 4.7.C.b of the Technical Specifications required additional tests of secondary containment during the first operating cycle of each of the three Browns Ferry units to supplement the other specified tests which are conducted throughout the life of the plants. All three Browns Ferry units have completed their first operating cycle and the additional tests specified in Section 4.7.C.b. TVA, therefore, proposes to delete this requirement, since it is no longer applicable. We conclude that the proposed deletion is acceptable.

3.6 Milk Sample Locations

As part of the environmental radiological monitoring program at the Browns Ferry Nuclear Plant, TVA collects and analyzes a number of samples. The Browns Ferry Nuclear Plant Environmental Technical Specifications state that "milk shall be collected....from at least four farms in the vicinity of the plant..." and that"...any location from which milk can no longer be obtained may be dropped from the surveillance program. The NRC shall be notified in writing that milk-producing animals are no longer present at that location. An additional milk sampling location will then be added to the program..." (Section 4.2.3.b).

As of May 15, 1978, milk is no longer available from the dairy farm located approximately four miles north of Browns Ferry Nuclear Plant. The milk producing animals have been sold and removed from the farm. A dairy farm located approximately five miles north of the plant has been added to the monitoring program.

We have reviewed the meteorological data and deposition factors for the Browns Ferry plant and conclude that the new sample location is acceptable.

4.0 Environmental Considerations

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

5.0 Conclusion

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

Dated: November 16, 1978

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 44 to Facility Operating License No. DPR-33, Amendment No. 40 to Facility Operating License No. DPR-52, and Amendment No. 17 to Facility Operating License No. DPR-68 issued to Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit Nos. 1, 2 and 3, (the facility) located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

These amendments change the Technical Specifications to (1) permit the average power range monitor system to be inoperable in the refuel mode, provided the source range monitors are connected to give a non-coincidence, high flux scram; (2) permit less than three intermediate range monitors (IRMs) per trip channel to be operable in the shutdown or refuel modes, provided at least four IRMs (one in each core quadrant) are connected to give a non-coincidence, high flux scram; (3) clarifies ambiguous portions of the Technical Specifications related to the rod block monitor system; (4) removes reference to an obsolete 1968 version of an ASTM procedure; (5) modifies the list of snubbers that are required to be operable; (6) removes a specification for additional tests of secondary containment that only applied during the first fuel cycle for each Browns Ferry Unit, and (7) changes one of the four locations where milk samples are collected.

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

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

For further details with respect to this action, see (1) the applications for amendments dated August 2, 1978 and August 11, 1978, (2) Amendment No. 44 to License No. DPR-33, Amendment No. 40 to License No. DPR-52, and Amendment No. 17 to License No. DPR-68, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Athens Public Library, South and Forrest, Athens,

Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 16 day of November 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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

Distribution
Docket
ORB #3
Local PDR
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SEPTEMBER 2 1 1978

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

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

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 42, 39 and 16 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant Units Nos. 1, 2 and 3. These amendments consist of changes to the Technical Specifications in response to your request of December 2, 1977, supplemented by letters dated December 20, 1977, May 24, May 26, June 30, August 2, August 10, and September 1, 1978.

These amendments authorize you to increase the storage capacity of each of the Browns Ferry spent fuel pools from 1080 to 3471 fuel assemblies.

Copies of the related Safety Evaluation, Environmental Impact Appraisal and the Notice of Issuance and Negative Declaration also are enclosed.

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 42 to DPR-33
2. Amendment No. 39 to DPR-52
3. Amendment No. 16 to DPR-68
4. Safety Evaluation
5. Environmental Impact Appraisal
6. Notice and Negative Declaration

MCW2002020219

*SEE PREVIOUS YELLOW FOR CONCURRENCES

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

SEPTEMBER 2 2 1978

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

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 42, 39 and 16 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant Units Nos. 1, 2 and 3. These amendments consist of changes to the Technical Specifications in response to your request of December 2, 1977, supplemented by letters dated December 20, 1977, May 24, May 26, June 30, August 2, August 10, and September 1, 1978.

These amendments authorize you to increase the storage capacity of each of the Browns Ferry spent fuel pools from 1080 to 3471 fuel assemblies.

Copies of the related Safety Evaluation, Environmental Impact Appraisal and the Notice of Issuance and Negative Declaration also are enclosed.

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 42 to DPR-33
2. Amendment No. 39 to DPR-52
3. Amendment No. 16 to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures:
see next page

OFFICE➤	ORB#3	ORB#3	OELD	ORB#3	AD/E&P/DOR	
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 39
License No. DPR-52

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

   A. The application for amendments by Tennessee Valley Authority (the licensee) dated December 2, 1977, as supplemented by letters dated December 20, 1977, May 24, May 26, June 30, August 2, August 10, and September 1, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility License No. DPR-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. 39, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid for

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 21, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 39

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

1. Remove page 331 and insert revised page 331.

2. The marginal line indicates the revised area. The overleaf page is provided for convenience.

B. The k_{eff} of the spent fuel storage pool shall be less than or equal to 0.95. Fuel stored in the pool shall not contain more than 15.2 grams of uranium-235 per axial centimeter of fuel assembly.

C. Loads greater than 1000 pounds shall not be carried over spent fuel assemblies stored in the spent fuel pool.

5.6 SEISMIC DESIGN

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

ENVIRONMENTAL IMPACT APPRAISAL

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AN INCREASE IN STORAGE CAPACITY OF

THE SPENT FUEL STORAGE POOLS

TENNESSEE VALLEY AUTHORITY

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

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

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

ENVIRONMENTAL IMPACT APPRAISAL

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AN INCREASE IN STORAGE CAPACITY

FOR THE

SPENT FUEL POOLS

FACILITY OPERATING LICENSES DPR-33, DPR-52, AND DPR-68

TENNESSEE VALLEY AUTHORITY

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

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

1.0 Description of Proposed Action

In their submittal of December 2, 1977, supplemented by letters dated December 20, 1977, May 24, 1978, May 26, 1978, June 30, 1978, August 2, 1978, August 10, 1978, and September 1, 1978, Tennessee Valley Authority (TVA or the licensee) requested amendments to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3 (BFNP). The proposed amendments and changes to the Technical Specifications would authorize TVA to increase the storage capacity of each of the three spent fuel pools (SFP) from 1080 to 3471 spent fuel assemblies.

The modification evaluated in this environmental impact appraisal is the proposal by the licensee to increase the storage capacity of the SFP by replacing the existing spent fuel storage racks with closer spaced racks and to use these new racks for the longer term storage of more spent fuel in the SFP. The increased storage capacity is achieved by using closer spaced racks than those described in Section 10.3 of the Final Safety Analysis Report (FSAR) for BFNP. The present racks have a center-to-center spacing of 11.75 x 6.6 inches whereas the new racks would store spent fuel assemblies on approximately a 6.5 inch center-to-center spacing.

2.0 Need for Increased Storage Capacity

Browns Ferry Units Nos. 1, 2 and 3 achieved initial criticality on August 17, 1973, July 20, 1974 and August 8, 1976, respectively. Units 1 and 2 have completed their first refueling (January and June, 1978).

During these refuelings, 168 spent fuel assemblies were transferred into the Unit 1 SFP and 132 assemblies into the Unit 2 SFP. Unit 3 is scheduled to shutdown for its first refueling in September, 1978 at which time 208 fuel assemblies are scheduled to be replaced. During the refueling outages for Units 1 and 2, TVA removed the six feedwater spargers, removed the cladding from the feedwater nozzles and installed improved feedwater sparger hardware. TVA also rerouted the control rod drive return line to the reactor water cleanup return line and capped the reactor vessel nozzle and the primary containment penetration. In order to complete these modifications, it was necessary to offload the entire core of 764 fuel assemblies into the SFP. During the refueling outage of Unit 3, scheduled for September 8, 1978, TVA plans to cap and reroute the CRD return line, which will require relocation of the entire core into the SFP. During the second refueling outage for Unit 3 scheduled for September 1979, TVA plans to replace the feedwater spargers as has been accomplished in Units 1 and 2; this will again require offloading of the entire core.

As described in Section 10.2 of the Final Safety Analysis Report (FSAR) for the Browns Ferry Nuclear Plant, all three units have a new fuel storage vault located adjacent to each SFP. New fuel has to be loaded into the SFP in order to transfer it into the reactor. Thus, if the new fuel storage vaults are used to store new fuel, as opposed to storing the new fuel in the SFP, each new fuel assembly must be handled twice rather than once to load it into the core. There is only one refueling bridge, which has to be used both to move spent or irradiated fuel into the SFP and to move new fuel into the reactor. To minimize the number of times a fuel assembly has to be handled, TVA is no longer using the new fuel storage vaults. Instead, new fuel is being stored in the SFP directly upon receipt onsite.

In the upcoming refueling of Unit 3, space must be available to store the 764 irradiated fuel assemblies that will be offloaded from the core plus the 208 new replacement fuel assemblies that will be in the SFP. The design storage capacity of each SFP was 1080 fuel assemblies; utilizing 54 of the standard GE 20 element racks. During the fall 1979 refueling outage for Unit 3, space for 1180 fuel assemblies is required (764 spaces for the full core offload, 208 spaces for the spent fuel from the September 1978 refueling and 208 spaces for the new replacement fuel). Under the present fuel handling arrangement, there would be a deficit of 100 storage spaces unless some of the present racks are replaced with higher density storage racks.

The estimated refueling schedules for Units 1, 2 and 3 are shown in Table 1 along with the estimated number of fuel assemblies scheduled to be replaced during each refueling and the cumulative number of spent fuel assemblies in each SFP. Even if new fuel were to be stored

in the vaults rather than in the SFPs - which would extend each refueling outage - it is evident that Unit 3 would lose the capability to discharge a full core after the fall 1979 refueling. The Unit 1 and Unit 2 SFPs are connected by a transfer canal. On the basis of maintaining one-half full core reserve storage in each of the Unit 1 and 2 SFPs, there would no longer be space to offload a full core in the combined pools after the refueling of Unit 2 in the spring of 1981. While the capability to off-load a full core is not required from the standpoint of safety (i.e., to the health and safety of the public), it is desirable from an ecomonic and operational standpoint and to reduce occupational radiation exposures if repairs or modifications are to be made on equipment or piping in or around the reactor vessel (e.g., the modifications to the Browns Ferry units discussed previously, the repairs to the recirculation nozzle safe ends presently performed at Duane Arnold, etc).

Aside from the more immediate need to increase storage capacity in the SFPs to maintain full core offload capability, increased storage capacity is required for continued operation of the plants. Based on the data in Table 1, if the storage capacity of the SFPs is not increased or if alternate storage space for spent fuel from these facilities is not available, Unit 2 would not be able to replace fuel after the spring 1982 refueling and Units 1 and 3 would not be able to replace fuel after the refuelings scheduled for the fall of 1982. Under this scenerio, the units could continue to operate until 1983, at which time the cores would no longer have sufficient reactivity to continue operation and the facility would have to be shutdown.

Another important consideration is the amount of open storage capacity that would be required to permit removal and replacement of the existing racks. None of the new racks can be installed until a portion of the existing racks are removed. Thus, it would not be possible to replace the present racks if they were all filled with spent fuel. The existing racks are about 5 1/2 feet by 2 feet. The minimum size of the new racks is about 7 1/4 feet by 7 1/4 feet. An additional consideration is the need to maintain any racks remaining in the pool or new racks added to the pool in independent seismically supported groups.

The proposed expansion provides storage for all discharges through 1992 for Browns Ferry 1, through 1993 for Browns Ferry 2, and through 1991 for Browns Ferry 3, while maintaining the full core reserve storage capacity. Therefore, storage capacity is extended for about

12 years for each of the units. In addition, five defective fuel assembly storage positions are provided for the storage of leaking or grossly defective fuel assemblies in the event they are required. If reprocessing is not resumed or if the Federal permanent repository or alternate storage facilities are not available by 1990, the units could continue to operate until 1996 (with some intertransfer of spent fuel) by sacrificing the full core discharge capability.

In this environmental evaluation, we have considered the impacts which may result from storage of up to an additional 2391 spent fuel assemblies in each of the three BFNP spent fuel pools on the basis that the spent fuel that is now in the Units 1 and 2 SFPs and the spent fuel to be stored in the pool from future refuelings may remain in the SFPs through at least the year 2000. We have also evaluated the benefits expected to be derived from the proposed and alternative courses of action.

The proposed modification would not alter the external physical geometry of the spent fuel pool or involve modifications to the SFP cooling or purification system. The licenses for Browns Ferry Units Nos. 1 and 2 expire May 10, 2007. The license for Unit No. 3 expires July 31, 2008. The proposed modification does not change the quantity of uranium fuel intended to be used in the reactor over the anticipated operating life of the facility and does not change the rate of generation of spent uranium fuel by the facility. The rate of spent fuel generation and the total quantity of spent fuel generated during the anticipated operating lifetime of the facility remains unchanged as a result of the proposed expansion. The modification will increase the number of spent fuel assemblies that could be stored in the SFP and the length of time that some of the fuel assemblies could be stored in the pool. If the modification is not approved, the amount of uranium used and the amount of spent fuel generated could be reduced from that anticipated when the licenses were issued, since the BFNP will be forced to shut down before the license expiration dates if alternate storage space for the spent fuel is not available.

3.0 Fuel Reprocessing History

Currently, spent fuel is not being reprocessed on a commercial basis in the United States. The Nuclear Fuel Services (NFS) plant at West Valley, New York, was shut down in 1972 for alterations and expansions; on September 22, 1977, NFS informed the Commission that they were withdrawing from the nuclear fuel reprocessing business. The Allied-General Nuclear Services (AGNS) proposed plant in Barnwell, South Carolina is not licensed to operate. The General Electric Company's (GE) Midwest Fuel Recovery Plant in Morris, Illinois, now referred to as Morris Operation (MO), is in a decommissioned condition. Although no plants are licensed for reprocessing fuel, the storage pool at Morris, Illinois and the storage pool at West Valley, New York (on land owned by the State of New York and leased to NFS through 1980) are licensed to store spent fuel. The storage pool at West Valley is not full but NFS is presently not accepting any additional spent fuel for storage, even from those power generating facilities that had contractual arrangements with NFS. Construction of the AGNS

receiving and storage station has been completed. AGNS has applied for - but has not been granted - a license to receive and store irradiated fuel assemblies in the storage pool at Barnwell. Further proceedings on this licensing action have not been scheduled. An application has been received from the Exxon Corporation for construction of a proposed spent fuel storage and reprocessing facility in Tennessee; licensing review of this application is suspended.

4.0 The Plant

The Browns Ferry Nuclear Plant (plant) is described in the Final Environmental Statement (FES) related to operation of the facility issued by the Tennessee Valley Authority on September 1, 1972, the Final Safety Analysis Report (FSAR) for the Browns Ferry Nuclear Plant and the Safety Evaluation Report (SER) of the Browns Ferry Nuclear Plant, Units 1, 2 and 3 issued by the Commission June 26, 1972. Each unit's nuclear steam supply system includes a General Electric Company (GE) single-cycle, forced circulation boiling water reactor (BWR) which generates steam for direct use in a steam turbine. Each unit is licensed to operate at steady state reactor core power levels of 3293 megawatts thermal (MWt). The net electrical output of each unit is about 1065 megawatts (MWe). Pertinent descriptions of principal features of the Plant as it currently exists are summarized below to aid the reader in following the evaluations in subsequent sections of this appraisal.

4.1 Fuel Inventory

The reactor core, which contains 764 fuel assemblies, is refueled each year, with about one-fourth of the core replaced during each refueling period. The assemblies now in use were manufactured by General Electric Corporation. The fuel for the reactor consists of slightly enriched uranium dioxide pellets contained in sealed zircaloy-2 tubes. These fuel rods are assembled into individual fuel assemblies of either 49 (7x7) or 64 (8x8) rods each.

4.2 Plant Water Use

4.2.1 Condenser Circulating Water System

All water required for operation of BFNP is obtained from Wheeler Reservoir, one of TVA's main stream reservoirs on the Tennessee River. The condenser circulating water system is designed to provide a total flow of 1,890,000 gpm to the condensers and a flow of 90,000 gpm to auxiliaries for the three units. No chemical or biocides are used to treat the circulating water system.

Six mechanical draft cooling towers are provided to dissipate waste heat to the atmosphere. Water is pumped through the main condenser and to an open channel going to the towers by three circulating water

pumps for each unit. Water is pumped to each cooling tower by two lift pumps. The system is designed for three possible modes of operation: open, helper, and closed. In the open mode water is drawn into the circulating water pumping station forebay from the reservoir, pumped through the main condenser, and discharged back into the reservoir through a diffuser discharge system consisting of perforated metal pipes which extend across the reservoir channel to diffuse the warmer water from the plant. In the helper mode the water is pumped from the reservoir, through the plant, and into an open channel going to the cooling towers where it is pumped through the towers and is returned to the reservoir through the diffusers. In the closed mode, the water is returned to the intake pumping station from the cooling tower discharge, and water is neither drawn from the reservoir (except for makeup) nor returned to the reservoir (except for blowdown).

4.2.2 Raw Cooling Water System

A Raw Cooling Water System is provided to remove heat from turbine associated equipment and accessories located in and adjacent to the turbine building, from the Reactor Building Closed Cooling Water System heat exchangers and from other reactor associated equipment which utilizes raw cooling water. The Raw Cooling Water System pumps are located in the turbine building and are supplied with river water from the condenser circulating water conduits. Three pumps are provided for each unit with one spare provided to Units 1 and 2 and one spare for Unit 3.

The Raw Cooling Water System furnishes cooling water to the following:

a. Turbine lube oil coolers
b. Generator stator water coolers
c. Generator hydrogen coolers
d. Reactor feed pump turbine oil coolers
e. Service and control air compressors
f. Steam jet air ejector precoolers
g. Generator alternator coolers
h. Air conditioning condensers
i. Recirculation pump M G set coolers
j. Reactor building closed cooling water heat exchangers
k. Other miscellaneous coolers

4.2.3 Raw Service Water System

A Raw Service Water System, consisting of three 50 percent-capacity pumps, supplies river water from the condenser circulating water conduits for yard watering, cooling for miscellaneous plant equipment requiring small quantities of high-pressure cooling water, washdown services in unlimited access areas and provides a means of pressurizing the raw water fire protection system.

4.2.4 Residual Heat Removal (RHR) Service Water System

The RHR Service Water System is a Class 1 system that consists of four pairs of pumps located on the intake structure for pumping raw river water to the heat exchangers in the RHR system and four pumps for supplying water to the Emergency Equipment Cooling Water System.

4.2.5 Emergency Equipment Cooling Water System

The safety objective of the Emergency Equipment Cooling Water System is to provide cooling water to the standby diesel generator, RHR and core spray equipment room environmental coolers, RHR pump seal coolers, and core spray thrust bearing coolers. It also provides an emergency Class 1 cooling water supply for the control room air conditioning chillers, station service air compressors, and reactor building closed cooling water heat exchangers.

4.2.6 Demineralized Water System

A 120,000 gallon-per-day water treatment plant furnishes a supply of high-purity water for makeup of the primary coolant systems, the Reactor Building Closed Cooling Water Systems, the suppression chambers, and the Standby Liquid Control Systems. The water is also used for radioactive decontamination work and preoperational cleaning of reactor and piping systems. In the makeup water treatment plant raw water from the river is passed through a filtration plant and a demineralized water plant. The latter consists of a pair of cation exchangers, a vacuum degasifier, a pair of anion exchangers, and a pair of mixed-bed exchangers. The water produced has a conductivity of less than 1.0 micromho per centimeter at 26°C and a dissolved silica content of less than 0.01 parts per million.

4.2.7 Potable Water and Sanitary Systems

The potable water for use in the plumbing systems is supplied in a 6-inch main by the city of Athens, Alabama. Obtaining water from this state-approved water supply was more economical than constructing and operating both a temporary and permanent purification plant.

All the sewage from the project is collected in a yard sewage system and flows to a treatment plant by gravity. Sewage ejectors, which discharge into the yard system, are provided at the pumping station and gate house. The sewage-treatment plant consists of two 15,000 gallons per day units arranged for parallel flow. Treatment is based on biological oxidation and reduction of sewage solids by additional aerobic digestion, which is accomplished by extended aeration and sedimentation. Effluent from the plant flows through a chlorine contact tank and discharges into the river.

4.3 Reactor Building Closed Cooling Water System

The reactor building closed cooling water system (RBCCWS) provides cooling water to designated auxiliary plant equipment located in the primary and secondary containments. The cooling water is available to the nuclear system auxiliaries under normal and accident conditions. The system consists of pumps, heat exchangers and necessary control and support equipment. The system is used to transfer heat from the SFP heat exchangers as well as a number of other systems such as the reactor recirculation pump and motor, drywell atmosphere cooler, the reactor building equipment drain tank cooler, the drywell equipment drain sump cooler, sample coolers, cleanup recirculating pump cooler, cleanup system and nonregenerative heat exchangers. The RBCCWS in turn transfers the heat to the Raw Cooling Water System as discussed in Section 4.2.2, above, through two heat exchangers. Under normal operation, the system is designed to transfer up to 31.3 x 10^6 BTU/hr with a river water temperature of 90°F.

4.4 Spent Fuel Pool Cooling and Cleanup System

A fuel pool cooling and cleanup system is provided to remove decay heat from spent fuel stored in the fuel pool and to maintain a specified water temperature, purity, clarity and level. The system cools the fuel storage pool by transferring the spent fuel decay heat through heat exchangers to the Reactor Building Closed Cooling Water System. Water purity and clarity in the storage pool, reactor well, and dryer-separator storage pit are maintained by filtering and demineralizing the pool water through a filter demineralizer.

The system for each fuel pool consists of two circulating pumps connected in parallel, two heat exchangers, one common filter-demineralizer subsystem, two skimmer surge tanks, and the required piping, valves, and instrumentation. Each pump has a design capacity equal to or greater than the system design flow rate and is capable of simultaneous operation. Four filter-demineralizers are provided, (one spare unit shared between the three active units) each with a design capacity equal to or greater than the design flow rate for a fuel pool. The pumps circulate the pool water in a closed loop, taking suction from the surge tanks, circulating the water through the heat exchangers and filter-demineralizer and discharging it through diffusers at the bottom of the fuel pool and reactor well. The water flows from the pool surface through skimmer weirs and scuppers to the surge tanks. The fuel pool pumps and heat exchangers are located in the reactor building below the bottom of the fuel pool. The fuel pool filter-demineralizers, which collect radioactive corrosion and fission products, are located in the radwaste building. The fuel pool concrete structure and metal liner are designed to withstand earthquake loads per project seismic requirements as a Class 1 system.

Fuel pool water is continuously recirculated. The heat exchangers are designed to remove the decay heat load of the normal discharge batch of spent fuel. The heat exchangers in the Residual Heat Removal System are used in conjunction with the Fuel Pool Cooling and Cleanup System to supplement pool cooling in the event that a larger than normal amount of fuel is stored in the pool. Makeup water for the system is transferred from the condensate storage tank to the skimmer surge tanks to make up evaporative and leakage losses.

Pool water clarity and purity are maintained by a combination of filtering and ion exchange. The filter-demineralizer maintains total dissolved heavy element content (Cu, Ni, Fe, Hg, etc.) at 0.1 ppm or less with a pH range of 6.0 to 7.5 for compatibility with aluminum fuel racks and other equipment. Particulate material is removed from the circulated water by the pressure precoat filter-demineralizer unit in which finely divided powdered ion exchanger resin serves as a disposable filter medium. The resin is replaced when the pressure drop is excessive or the ion exchange resin is depleted. Backwashing and precoating operations are controlled from the radwaste building. The spent filter medium is flushed from the elements and transferred to the waste backwash receiver tank by backwashing with air and condensate. New ion exchange resin is mixed in a precoat tank and transferred as a slurry by a precoat pump to the filter where the solids deposit on the filter elements. The holding pump maintains circulation through the filter in the interval between the precoating operation and the return to normal system operation.

The SFP Cooling and Cleanup System was designed on the basis that only one of the two pumps and heat exchangers would be needed to remove the decay heat released by the average spent fuel batch discharged from the equilibrium fuel cycle plus the heat being released by the batch discharged at the previous refueling. With one of the pumps operating, flow rate through the system is 600 gpm. This is more than is required for two complete water changes per day of the approximately 51,300 cubic feet volume of the SFP or one change per day of the approximately 106,900 cubic feet of volume in the combined SFP, reactor well and dryer-separator pit. Under the design heat load of 8.8×10^6 BTU/hr (both pumps and heat exchangers in operation), the SFP Cooling and Cleanup System will maintain the temperature of the water below 125°F with the reactor building closed cooling water system temperature at its maximum. If additional cooling is required, the SFP Cooling and Cleanup system can be connected by operator action to the Residual Heat Removal System. With this connection, and allowing the pool water temperature to increase to 150°F, the heat transfer capability is increased to 27.6×10^6 BTU/hr.

4.5 Heat Dissipation to Environment

As discussed in Section 4.2.1, above, the BFNP is designed to discharge the heat from the main condensers and auxiliary cooling systems either directly to Wheeler Reservoir and the Tennessee River (open mode of operation), to the atmosphere through the six mechanical draft cooling towers (closed mode of operation) or partially to both the river and atmosphere. At rated power, the discharge of heat from the main condenser in each unit is about 7.77×10^9 BTU/hr.

4.6 Radioactive Wastes

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid waste that might contain radioactive material. The waste treatment systems are evaluated in the Final Environmental Statement (FES) dated September 1972. There will be no change in the waste treatment systems described in Section 2.4 of the FES because of the proposed modification.

4.7 Purpose of SFPs

The SFPs at BFNP were designed to store spent fuel assemblies prior to shipment to a reprocessing facility. These assemblies may be transferred from the reactor core to the SFP during a core refueling, or to allow for inspection, repair and/or modification to core internals. The latter may require the removal and storage of up to a full core, as was required during the first refuelings of Units 1 and 2 and as is presently required to modify the control rod drive return line for Unit 3. The assemblies are initially intensely radioactive due to their fission product content and have a high thermal output. They are stored in the SFP to allow for radioactive and thermal decay.

The major portion of decay occurs during the first 150-day period following removal from the reactor core. After this period, the assemblies may be withdrawn and placed into a heavily shielded fuel cask for offsite shipment. Space permitting, the assemblies may be stored for an additional period allowing continued fission product decay and thermal cooling prior to shipment.

5.0 Environmental Impacts of Proposed Action

5.1 Land Use

The proposed modification will not alter the external physical geometry of the SFP. The SFP is entirely contained within the existing reactor building structure. No additional commitment of land is required. The SFP was designed to store spent fuel assemblies under water for a

period of time to allow shorter-lived radioactive isotopes to decay and to reduce their thermal heat output. The Commission has never set a limit on how long spent fuel assemblies could be stored onsite. The longer the fuel assemblies decay, the less radioactivity they contain. The proposed modification will not change the basic land use of the SFP. The pool was designed to store the spent fuel assemblies from up to six normal refuelings. The modification would provide storage for up to eighteen normal refuelings. The pool was intended to store spent fuel. This use will remain unchanged by the proposed modification. The proposed modification will make more efficient use of the land already designated for spent fuel storage.

5.2 Water Use

There will be no significant change in plant water usage as a result of the proposed modification. As discussed subsequently, storing additional spent fuel in the SFP will increase the heat load on the SFP cooling system, which is transferred to the Reactor Building Closed Cooling Water System, thence to the plant Raw Cooling Water System and is dissipated in the environment by discharge to Wheeler Reservoir and/or the atmosphere. The modifications will not change the flow rates within any cooling system. As discussed in Section 10.5 of the BFNP Final Safety Analysis Report (FSAR), the design bases for the SFP cooling system was that for a normal refueling cycle the fuel pool cooling system would be capable of maintaining the bulk pool temperature below 125°F. The maximum possible heat load, (i.e., the decay heat of a full core at the end of a full cycle plus the decay heat from fuel discharged at previous refuelings), the fuel pool cooling system in conjunction with the Residual Heat Removal (RHR) system would be capable of maintaining the bulk pool temperature below 150°F. As discussed in Section 4.4, the SFP Cooling and Cleanup System can be connected to the RHR system to increase the cooling capacity. Based on the expected annual refueling cycle, TVA estimates that the peak heat load could be 14.8×10^6 BTU/hr when the 17th annual discharge is moved into the SFP in 1993 or 1994. With the existing storage capacity of 1080 spent fuel assemblies, the peak heat load from 5 annual discharges would be 13.3×10^6 BTU/hr. Thus, TVA's estimate of the incremental heat load from the proposed expansion was 1.5×10^6 BTU/hr resulting from the normal annual refueling cycle. We estimate that the maximum incremental decay heat load could be 2.65×10^6 BTU/hr, increasing from 10.7 to 13.35 BTU/hr. Based on our estimate, the bulk pool water temperature could be increased by 8°F after the 17th annual refueling if the additional heat is not removed by using the RHR system in conjunction with the SFP Cooling and Cleanup System. Our estimates were based on the core operating at 100% power factor, whereas the cumulative capacity factors to date for Units 1, 2, and 3 has only been 38.4%, 31.7% and 77.0%, respectively. By using the RHR system as necessary to supplement the SFP Cooling and Cleanup System, the

bulk SFP water temperature can be maintained below 125°F during normal refuelings and below 150°F in the event it is necessary to off-load a full core. This was the design basis for the SFP as described in the FSAR and evaluated by the staff at the operating license review. We conclude that there will be no significant increase in evaporation rates as a result of the proposed modification and thus no significant increase in the amount of makeup water that will be added to the SFP. The increase in water makeup attributable to the modification because of increased evaporation from the pool will be undetectable in the total plant makeup water requirement.

5.3 Heat Rejection

As discussed in Section 5.2 above and in the accompanying Safety Evaluation, the storage of more spent fuel in the BFNP SFP will slightly increase the decay heat load in the pool water. This increase will be insignificant particularly compared to the heat rejection from the secondary system heat cycle at the main condenser and further does not constitute a net increase of effect on the environment because this heat loss would occur regardless of the location where the spent fuel is stored.

We estimate that the maximum incremental heat load that could be added to the SFP water by increasing the number of stored spent fuel assemblies from 1080 to 3471 will be 2.6×10^6 BTU/hr from the normal annual refuelings and 3.4×10^6 BTU/hr for full core offloads that essentially fill the present and the modified pools. As noted in section 4.5, at rated power, the discharge of heat from the main condenser in each unit is about $7{,}770 \times 10^6$ BTU/hr.

The plant cooling water system will accommodate the additional heat load. The increase of heat load contribution of stored spent fuel to total plant thermal discharge to the environment during normal operation is less than 0.02 percent. The incremental heat load from the SFP will have a negligible incremental impact and is so low that it would not be differentiated in thermal plume measurements. The slight increase in thermal effluents will not effect the plant's capability of complying with the Alabama water quality standards.

5.4 Radiological

5.4.1 Introduction

The potential offsite radiological environmental impacts associated with the expansion of the spent fuel storage capacity were evaluated and determined to be environmentally insignificant as addressed below.

The additional spent fuel which would be stored due to the expansion is the oldest fuel which has not been shipped from the plant. This fuel should have decayed at least five years. During the storage of the spent fuel under water, both volatile and nonvolatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54 which are not volatile. The radionuclides that might be released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90, are also predominately nonvolatile. The primary impact of such nonvolatile radioactive nuclides is their contribution to radiation levels to which workers in and near the SFP would be exposed. The volatile fission product nuclides of most concern that might be released through defects in the fuel cladding are the noble gases (xenon and krypton), tritium and the iodine isotopes.

Experience indicates that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the spent fuel pool water appear to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the spent fuel pool during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFP. During and after refueling, the spent fuel pool cleanup system reduces the radioactivity concentrations considerably. It is theorized that most failed fuel contains small, pinhole-like perforations in the fuel cladding at the reactor operating condition of approximately 800°F. A few weeks after refueling, the spent fuel cools in the spent fuel pool so that fuel clad temperature is relatively cool, approximately 180°F. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the gap between pellets and clad, thereby tending to retain the fission products within the gap. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months.

5.4.2 Effect of Fuel Failure on the SFP

Experience indicates that there is little radionuclide leakage from Zircaloy clad spent fuel stored in pools for over a decade. The predominance of radionuclides in the spent fuel pool water appears

to be radionuclides that were present in the reactor coolant system prior to refueling (which become mixed with water in the spent fuel pool during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFP. During and after refueling, the spent fuel pool cleanup system reduces the radioactivity concentrations considerably.

Operators at several reactors have discharged, stored, and/or shipped relatively large numbers of Zircaloy-clad fuel which developed defects during reactor exposures, e.g., Ginna, Oyster Creek, Nine Mile Point, and Dresden Unit Nos. 1 and 2. Several hundred Zircaloy-clad assemblies which developed one or more defects in-reactor are stored in the GE-Morris pool without need for isolation in special cans. Detailed analysis of the radioactivity in the pool water indicates that the defects are not continuing to release significant quantities of radioactivity. Normal radioactivity concentrations in the Morris pool water are about 3×10^{-4} μCi/ml which is near the maximum desired concentration for occupational exposure considerations in bathing and culinary uses. The radioactivity concentrations rose to 2×10^{-3} μCi/ml during a month when the water cleanup system was removed from service.

Based on the operational reports submitted by the licensees and discussions with the operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the Morris Operation (MO) pool (formerly Midwest Recovery Plant) at Morris, Illinois, or at Nuclear Fuel Services' (NFS) storage pool at West Valley, New York. Spent fuel has been stored in these two pools which, while it was in a reactor, was determined to have significant leakage and was, therefore, removed from the core. After storage in the onsite spent fuel pool, this fuel was later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from this fuel in the offsite storage facility.

A recent Battelle Northwest Laboratory (BNL) report, "Behavior of Spent Nuclear Fuel in Water Pool Storage: (BNWL-2256 dated September 1977), states that radioactivity concentrations may approach a value up to 0.5 μCi/ml during fuel discharge in the SFP. After the refueling, the SFP ion exchange and filtration units will reduce and maintain the pool water in the range of 10^{-3} to 10^{-4} μCi/ml.

In handling defective fuel, the BNL study found that the vast majority of failed fuel does not require special handling and is stored in the same manner as intact fuel. Two aspects of the defective fuel account for its favorable storage characteristics. First, when a fuel rod perforates in-reactor, the radioactive gas inventory is released to the reactor primary coolant. Therefore, upon discharge, little additional gas release occurs. Only if the failure occurs by mechanical damage in the basin are radioactive gases released in detectable amounts, and this type of damage is extremely rare. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels. The second favorable aspect is the inert character of the uranium oxide pellets in contact with water. This has been demonstrated in laboratory studies and also by casual observations of pellet behavior when broken rods are stored in pools.

5.4.3 Radioactive Material Released to Atmosphere

With respect to gaseous releases, the only significant noble gas isotope attributable to storing additional assemblies for a longer period of time would be Krypton-85. As discussed previously, experience has demonstrated that after spent fuel has decayed 4 to 6 months, there is no significant release of fission products from defected fuel. However, we have conservatively estimated that an additional 102 curies per year of Krypton-85 may be released from the three units when the modified pools are completely filled. This increase would result in an additional total body dose of less than 0.005 mrem/year to an individual at the site boundary. This dose is insignificant when compared to the approximately 100 mrem/year that an individual receives from natural background radiation. The additional total body dose to the estimated population within a 50-mile radius of the plant is less than 0.005 man-rem/year. This is small compared to the fluctuations in the annual dose this population would receive from natural background radiation. Under our conservative assumptions, these exposures represent an increase of less than 0.5% of the exposures from the plant evaluated in the FES for the individual and the population (Table 2.4-3). Thus, we conclude that the proposed modification will not have any significant impact on exposures offsite.

Assuming that the spent fuel will be stored onsite for several years, Iodine-131 releases from spent fuel assemblies to the SFP water will not be significantly increased because of the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings.

Storing additional spent fuel assemblies should not increase the bulk water temperature during normal refuelings above the 125°F used in the design analysis. Therefore, it is not expected that there will be any significant change in the annual release of tritium or iodine as a result of the proposed modification from that previously evaluated in the FES.

Most airborne releases from the plant result from leakage of reactor coolant which contains tritium and iodine in higher concentrations than the spent fuel pool. Therefore, even if there were a slightly higher evaporation rate from the spent fuel pool, the increase in tritium and iodine released from the plant as a result of the increase in stored spent fuel would be small compared to the amount normally released from the plant and that which was previously evaluated in the FES. If levels of radioiodine become too high, the air can be diverted to charcoal filters for the removal of radioiodine before release to the environment. In addition, the plant radiological effluent Technical Specifications, which are not being changed by this action, restrict the total releases of gaseous activity from the plant including the SFP.

5.4.4 Solid Radioactive Wastes

The concentration of radionuclides in the pool is controlled by the filter-demineralizers and by decay of short-lived isotopes. The activity is high during refueling operations while reactor coolant water is introduced into the pool and decreases as the pool water is processed through the filter-demineralizer. The increase of radioactivity, if any, should be minor because the additional spent fuel to be stored is relatively cool, thermally, and radionuclides in the fuel will have decayed significantly.

While we believe that there should not be an increase in solid radwaste due to the modification, as a conservative estimate, we have assumed that the amount of solid radwaste may be increased by 48 cubic feet of resin a year from the demineralizer (twelve additional resin beds/year) for each unit. The annual

average amount of solid waste shipped from Browns Ferry 1, 2 and 3 for 1975 to 1977 is about 42,000 cubic feet per year. If the storage of additional spent fuel does increase the amount of solid waste from the SFP purification systems by about 144 cubic feet per year, the increase in total waste volume shipped would be less than 0.4% and would not have any significant environmental impact.

The present spent fuel racks to be removed from the SFP are contaminated and will be disposed of as low level waste. The licensee has estimated that about 5,000 cubic feet of solid radwaste will be removed from the SFP of each unit because of the proposed modification. Therefore, the total waste shipped from the plant should be increased by less than 1% per year when averaged over the lifetime of the plant. This will not have any significant environmental impact.

5.4.5 Radioactivity Released to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the plant as a result of the proposed modification. The amount of radioactivity on the SFP filter-demineralizer might slightly increase due to the additional spent fuel in the pool but this increase of radioactivity should not be released in liquid effluents from the plant.

The demineralizer resins are periodically flushed with water to the condensate phase separator tank. The water used to transfer the spent resin is decanted from the tank and returned to the liquid radwaste system for processing. The soluble radioactivity will be retained on the resins. If any activity should be transferred from the spent resin to this flush water, it would be removed by the liquid radwaste system.

Leakage from the SFP is collected in the Reactor Building floor drain sumps. This water is transferred to the liquid radwaste system and is processed by the system before any water is discharged from the plant.

5.4.6 Occupational Exposures

We have reviewed the licensee's plan for the removal, crating and disposal of the low density racks and the installation of the high density racks with respect to occupational radiation exposure. The occupational exposure for the entire operation is estimated by the licensee to be about 32 man-rem for Units 1 and 2 and about 8 man-rem for Unit 3. We consider this to be a conservative estimate based on the occupational exposures recorded at over two dozen other facilities that have increased the storage capacity of their spent fuel pools. This operation is expected to be a small fraction of the total annual man-rem burden from occupational exposure at this facility.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by the licensee and by utilizing relevant assumptions for occupancy times and for dose rates in the spent fuel pool area from radionuclide concentrations in the SFP water. The spent fuel assemblies themselves contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. The occupational radiation exposure resulting from the proposed action represents a negligible burden. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation exposure burden at this facility. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

5.4.7 Impact of Other Pool Modifications

As discussed above, the additional environmental impacts in the vicinity of Browns Ferry 1, 2 and 3 resulting from the proposed modification are very small fractions (less than 1%) of the impacts evaluated in the Browns Ferry 1, 2 and 3 FES. These additional impacts are too small to be considered anything but local in character.

Based on the above, we conclude that a SFP modification at any other facility should not significantly contribute to the environmental impact of the proposed action at Browns Ferry 1, 2 and 3 and that the Browns Ferry 1, 2 and 3 modification should not contribute significantly to the environmental impact of any other facility.

5.4.8 Evaluation of Radiological Impact

As discussed above, the proposed modification does not significantly change the radiological impact evaluated in the FES.

5.5 Nonradiological Effluents

There will be no change in the chemical or biocidal effluents from the plant a result of the proposed modification.

The only potential offsite nonradiological environmental impact that could arise from this proposed action would be additional discharge of heat to the atmosphere and to the Tennessee River. Storing spent fuel in the SFP for a longer period of time will add more heat to the SFP water. The spent fuel pool heat exchangers

are cooled by the reactor building cooling water system which in turn is cooled by the plant Raw Cooling Water System. An evaluation of the augmented spent fuel storage facility was made to determine the effects of the increased heat generation on the plant cooling water systems, and ultimately, on the environment.

As discussed in the staff's Safety Evaluation, the maximum incremental heat load that will be added by use of the proposed rack modification is that from unloading a full core which would fill the pool. The maximum calculated heat generation rate in this case would be about 3.4×10^6 Btu/hr.

The total heat load on the environment from BFNP used in the evaluation in the FES was 7.8×10^9 Btu/hr per unit. The incremental heat load attributable to the proposed modification would be less than 0.02% of the total heat rejection rate. Compared to the existing heat load, which was evaluated in the FES and has been evaluated by continuing environmental monitoring porgrams, the additional thermal impact from the proposed modification will be negligible.

5.6 Impacts on the Community

The new storage racks will be fabricated offsite and shipped to the plant. No environmental impacts on the environs outside the spent fuel storage building are expected during removal of the existing racks and installation of the new racks. The impacts within this building are expected to be limited to those normally associated with metal working activities and fuel handling operations. No significant environmental impact on the community is expected to result from the fuel rack conversion or from subsequent operation with the increased storage of spent fuel in the SFP.

5.7 Transportation and Handling

Delivery of material for the new high density storage racks and disposal of the existing racks for off-site burial will involve truck and/or rail transportation activity. The number of such shipments will be less than would be required to ship the spent fuel offsite at this time. By deferring offsite shipment of spent fuel, a number of factors can be considered that will reduce the overall environmental impact: More fuel might be loaded per shipping cask, reducing the number of miles in transport; a lighter shipping cask may be used, reducing the tonnage in transport; and the reduced radiation level of spent fuel will further reduce the already minimal environmental impact of spent fuel shipments which are covered by the Final Environmental Statement.

6.0 Environmental Impact of Postulated Accidents

Although the new high density racks will accommodate a larger inventory of spent fuel, we have determined that the installation and use of the racks will not change the radiological consequences of a postulated fuel handling accident in the SFP area from those values reported in the FES for Browns Ferry 1, 2 and 3 dated September 1972. The Commission's Safety Evaluation assessed fuel handling accidents; there is no change in fuel handling operations as a result of this proposed modification.

Additionally, the NRC staff has under way a generic review of load handling operations in the vicinity of spent fuel pools to determine the likelihood of a heavy load impacting fuel in the pool and, if necessary, the radiological consequences of such an event. The Technical Specifications are being changed to prohibit loads greater than 1000 pounds (approximately the weight of a fuel assembly, channel and associated load handling equipment) from being transported over spent fuel in the SFP. We have concluded that the likelihood of a heavy load handling accident is sufficiently small that the proposed modification is acceptable and no additional restrictions on load handling operations in the vicinity of the SFP are necessary while our generic review is under way.

7.0 Alternatives

In regard to this licensing action, the NRC staff has considered the following alternatives; (1) reprocessing the spent fuel, (2) shipment of spent fuel to a separate fuel storage facility, (3) shipment of spent fuel to another reactor site, (4) lengthening the fuel cycles, (5) reducing plant power factors through energy conservation and (6) ceasing operation of the facility. These alternatives are considered in turn.

The total cost associated with the project for all three Browns Ferry units is expected to be about $19 million in 1977 dollars. This estimate includes the following five categories of expense:

1. Project management, design, quality assurance, and licensing.
2. Materials, tooling, and hardware fabrication.
3. Removal, installation, and transportation.
4. Contingency allowance.
5. Allowance for funds used during construction.

This equates to about $2650 for each of the additional 7173 storage spaces that would be provided by the proposed modification.

7.1 Reprocessing of Spent Fuel

As discussed earlier, none of the three commercial reprocessing facilities in the U.S. is currently operating. The General Electric Company's Midwest Fuel Recovery Plant at Morris, Illinois is in a decommissioned condition. On September 22, 1976, Nuclear Fuel Services, Inc. (NFS) informed the Nuclear Regulatory Commission that they were "withdrawing from the nuclear fuel reprocessing business." The Allied-General Nuclear Services (AGNS) reprocessing plant received a construction permit on December 18, 1970. In October 1973, AGNS applied for an operating license for the reprocessing facility; construction of the reprocessing facility is essentially complete but no operating license has been granted. On July 3, 1974, AGNS applied for a materials license to receive and store up to 400 MTU of spent fuel in the onsite storage pool, on which construction has also been completed but hearings with respect to this application have not yet commenced and no license has been granted.

In 1976, Exxon Nuclear Company, Inc. submitted an application for a proposed Nuclear Fuel Recovery and Recycling Center (NFRRC) to be located at Oak Ridge, Tennessee. The plant would include a storage pool that could store up to 7,000 MTU in spent fuel. Licensing review of this application is suspended.

On April 7, 1977, the President issued a statement outlining his policy on continued development of nuclear energy in the U.S. The President stated that: "We will defer indefinitely the commercial reprocessing and recycling of the plutonium produced in the U.S. nuclear power programs. From our own experience, we have concluded that a viable and economic nuclear power program can be sustained without such reprocessing and recycling."

On December 23, 1977 the Nuclear Regulatory Commission announced that it would order the termination of the now-pending fuel cycle licensing actions involving GESMO (Docket No. RM-50-5), Barnwell Nuclear Fuel Plant Separation Facility, Uranium Hexafluoride Facility and Plutonium Product Facility (Docket No. 50-332, 70-1327 and 70-1821), the Exxon Nuclear Company, Inc. Nuclear Fuel Recovery and Recycling Center (Docket No. 50-564), the Westinghouse Electric Corporation Recycle Fuels Plants (Docket No. 70-1432), and the Nuclear Fuel Services, Inc. West Valley Reprocessing Plant (Docket No. 50-201). The Commission also announced that it would not at this time consider any other applications for commercial facilities for reprocessing spent fuel, fabricating mixed-oxide fuel, and related functions. At this time, any considerations

of these or comparable facilities has been deferred for the indefinite future. Accordingly, the Staff considers that shipment of spent fuel to such facilities for reprocessing is not a viable alternative to the proposed expansion of the BFNP spent fuel pool especially when considered in the relevant time frame - i.e., from now until 1980 - when expanded capacity at BFNP will be needed.

The licensee had intended to reprocess the spent fuel to recover and recycle the uranium and plutonium in the fuel. Due to a change in national policy and circumstances beyond the licensee's control, reprocessing of the spent fuel is not an available option at this time. Even if the governmental policy were changed tomorrow to allow reprocessing of spent fuel, the current backlog of spent fuel and the time it would take to bring adequate reprocessing capacity on line would require that current spent fuel be stored somewhere for up to another 10 years.

7.2 Independent Spent Fuel Storage Facility

An alternative to expansion of onsite spent fuel pool storage is the construction of new "independent spent fuel storage installations" (ISFSI). Such installations could provide storage space in excess of 1,000 MTU of spent fuel. This is far greater than the capacities of onsite storage pools. Fuel storage pools at GE Morris and NFS are functioning as ISFSIs although this was not the original design intent. Likewise, if the receiving and storage station at AGNS is licensed to accept spent fuel, it would be functioning as an ISFSI until the reprocessing facility is licensed to operate. The license for the GE facility at Morris, Illinois was amended on December 3, 1975 to increase the storage capacity to about 750 MTU: as of August 30, 1978, 310 MTU was stored in the pool in the form of 1196 spent fuel assemblies. An application for an 1100 MTU capacity addition is pending. Present schedule calls for completion in 1980 if approved. However by motion dated November 8, 1977 General Electric Company requested the Atomic Safety and Licensing Board to suspend indefinitely further proceedings on this application. This motion was granted. The staff has discussed the status of storage space at MO with GE personnel. We have been informed that GE is primarily operating the MO facility to store either fuel owned by GE (which had been leased to utilities on an energy basis) or fuel which GE had previously contracted to reprocess. We were informed that the present GE policy is not to accept spent fuel for storage except for that fuel for which GE has a previous commitment. In response to the Commission's requests for justification for the requested increase in storage capacity at MO, G.E. described the space being reserved for various utilities. No space was listed as being reserved for Browns Ferry spent fuel. The NFS facility has capacity for about 260 MTU, with approximately 170 MTU presently stored in the pool. The storage pool at West Valley, New York, is on land owned by the State of New York and leased to NFS thru 1980. Although the storage pool at West Valley is not full, since NFS withdrew from the fuel reprocessing business, correspondence we have received indicated that they are not at present accepting

additional spent fuel for storage even from the reactor facilities with which they had contracts. The status of the storage pool at AGNS was discussed above.

The original core loading for each of the Browns Ferry Units and the reloads have been supplied by General Electric Company. Under terms of TVA's former contract with GE, the latter was required to remove and reprocess discharged spent fuel. In the absence of an operating reprocessing facility in this country and the recent national policy to defer reprocessing, TVA has reached agreement with GE to store the spent fuel onsite until there is a better resolution of national policy on reprocessing and interim and permanent storage of spent fuel. On April 29, 1977, the President issued "The National Energy Plan"; Chapter VI outlined the plan for Coal, Nuclear and Hydroelectric Power. In discussing the program to "develop techniques for long-term storage of spent fuel", it was noted that "improved methods of storing spent fuel will enable most utilities at least to double their current storage capacity without constructing new facilities." The basis for the current Department of Energy (DOE) policy is that if storage space is or can be made available, spent fuel should be stored onsite until it can be shipped directly to the permanent Federal repository which the President has directed DOE to develop.

With respect to construction of new ISFSIs, Regulatory Guide 3.24, "Guidance on the License Application, Siting, Design, and Plant Protection for an Independent Spent Fuel Storage Installation," issued in December 1974, recognizes the possible need for ISFSIs and provides recommended criteria and requirements for water-cooled ISFSIs. Pertinent sections of 10 CFR Parts 19, 20, 30, 40, 51, 70, 71 and 73 would also apply.

The staff has estimated that at least five years would be required for completion of an independent fuel storage facility. This estimate assumes one year for preliminary design; one year for preparation of the license application, Environmental Report, and licensing review in parallel with one year for detail design; two and one-half years for construction and receipt of an operating license; and one-half year for plant and equipment testing and startup.

Industry proposals for independent spent fuel storage facilities are scarce to date. In late 1974, E. R. Johnson Associates, Inc. and Merrill Lynch, Pierce, Fenner and Smith, Inc. issued a series of joint proposals to a number of electric utility companies having nuclear plants in operation or contemplated for operation, offering to provide independent storage services for spent nuclear fuel. A paper on this proposed project was presented at the American Nuclear Society meeting in November 1975 (ANS Transactions, 1975 Winter Meeting, Vol. 22, TANSAO 22-1-836, 1975). In 1974, E. R. Johnson Associates estimated their construction cost at about $20 million.

Several licensees have evaluated construction of a separate independent spent fuel storage facility and have provided cost estimates. In 1975, Connecticut Yankee, for example, estimated that to build an independent facility with a storage capacity of 1,000 MTU (BWR and/or PWR assemblies) would cost approximately $54 million and take about 5 years to put into operation. Commonwealth Edison estimated the construction cost to build a fuel storage facility at about $10,000 per fuel assembly. To this would be added the costs for maintenance, operation, safeguards, security, interest on investment, overhead, transportation and other costs.

On December 2, 1976, Stone and Webster Corporation submitted a topical report requesting approval for a standard design for an independent spent fuel storage facility. No specific locations were proposed, although the design is based on location near a nuclear power facility. No estimated costs for fuel storage were included in the topical report.

TVA evaluated construction of an independent spent fuel storage facility. No specific costs were cited, but the licensee noted that "an independent facility would possibly require acquisition of additional land and would necessarily require construction of a spent fuel pool with associated containment, purchase of heat removal systems, shipping cask and spent fuel transportation system, plus operational and security personnel whereas the proposed modification requires only the installation of spent fuel storage racks". TVA concluded that it would obviously be much more expensive to construct an independent storage facility than to implement the proposed modification.

On a short-term basis (i.e., prior to 1983) an independent spent fuel storage installation does not appear to be a viable alternative based on cost or availability in time to meet the licensee's needs. In addition, constructing an ISFSI would have a greater environmental impact than the proposed action. A new or expanded facility would require additional land use and constructing considerable equipment and structures, whereas installing new racks at Browns Ferry requires only the small amount of material necessary to construct the racks and the modest personnel exposure during installation. Based on our own evaluation, we estimate it would cost at least twice as much per assembly to construct an ISFSI.

In the long-term, the U. S. Department of Energy (USDOE) is modifying its program for nuclear waste management to include design and evaluation of a retrievable storage facility to provide Government storage at central locations for unreprocessed spent fuel rods. The pilot plant is expected to be completed by late 1985 or 1986. It is estimated that the long-term storage facility will start accepting commerical spent fuel in the time frame of 1990 to 1993. The design is based on storing the spent fuel in a retrievable condition for a minimum of 25 years. The criteria for acceptance is that the spent fuel must have decayed a minimum of ten years so it can be stored in dry condition without need for forced air circulation. As an interim alternative to the long term retrievable storage facility, on October 18, 1977, USDOE announced a new "spent nuclear fuel policy". USDOE will determine industry interest in providing interim fuel storage services on a contract basis. If adequate private storage services cannot be provided, the Government will provide interim fuel storage facilities. It was announced by USDOE at a public meeting held on October 26, 1977, that this interim storage is expected to be available in the 1981-1982 time frame. USDOE thru their Savannah River Operations Office is preparing a conceptual design for a possible spent fuel storage pool of about 5000 MTU capacity. DOE has requested, but has not received, Congressional authorization for design and construction of this interim spent fuel storage facility. Based on our discussions with USDOE personnel, it appears that the earliest such a pool could be licensed to accept spent fuel would be about 1983. The interim facility(s) would be designed for storage of the spent fuel under water. USDOE stated that it was their intent to not accept any spent fuel that had not decayed a minimum of five (5) years.

As indicated in the President's energy policy statement of April 29, 1977, the preferred solution to the spent fuel storage program is to have the nuclear power plants store their spent fuel on-site until the government long term storage facility is operable, which is now estimated to be about 1990 to 1993. For those nuclear power plants that cannot store the spent fuel on-site until the permanent long-term storage facility is available, USDOE intends to provide limited interim storage facilities.

7.3 Storage at Another Reactor Site

TVA has 14 nuclear facilities under construction. Watts Bar 1 and 2, which are the most advanced in construction, along with Sequoyah 1 and 2 and Yellow Creek 1 and 2 are PWRs. PWR fuel assemblies are much larger than BWR fuel assemblies. Different racks than those proposed in the design for these facilities would have to be installed to store spent fuel from Browns Ferry. Like BFNP, Phipps Bend 1 and 2 and Hartsville 1, 2, 3 and 4 are BWRs. The earliest construction is estimated to be completed on any of these facilities is late 1982 (Hartsville 1). The Browns Ferry Unit No. 3 SFP will be essentially full after the refueling scheduled for September 1982. TVA is planning to increase the spent fuel storage capacity at most of these facilities compared to that proposed in the original design. This proposed action is necessary to provide onsite storage of spent fuel from the specific facility until the Federal permanent repository is available. Considering the uncertainty in the time when another BWR facility may be available in the TVA system and the transportation costs associated with moving spent fuel between facilities, storage of spent fuel from Browns Ferry in another TVA facility is a possible alternative to the proposed action but would be more expensive, offer no environmental benefits and is very unlikly to be available before it would be necessary to shutdown one or more of the Browns Ferry units.

Storage of spent fuel at another reactor facility outside the TVA system would be physically possible but is not considered a realistic alternative. Most operating reactors in the United States are experiencing shortages in spent fuel storage capacity and could not efficiently provide storage space for other plants. Furthermore, no current power plants are licensed to receive spent fuel from offsite. Storage of BFNP spent fuel at another reactor facility is, therefore, not considered a viable alternative.

According to a survey conducted and documented by the former Energy Research and Development Administration, up to 27 of the operating nuclear power plants will lose the ability to refuel during the period 1977-1986 without additional spent fuel storage pool expansions or access to offsite storage facilities. Thus, the licensee cannot assuredly rely on any other power facility

to provide additional storage capability except on a short-term emergency basis. If space were available in another reactor facility, it is unlikely that the cost would be less than storage onsite as proposed.

7.4 Lengthening the Fuel Cycle

The present fuel cycles for light water reactors was based on the premise that spent fuel would be reprocessed and the fissionable material recovered and recycled. With the change in national policy to a "throw-away" cycle, the industry is evaluating higher initial loadings, higher burnups, recycling of low burnup fuel assemblies and extension of times between refuelings. These types of changes are not an immediate potential alternative. To obtain data to support higher burnups will require exposure of experimental fuel in reactors for several years. The lead time for design and procurement of core reloads is one to two years. In the long run, redesigning the fuel cycle can extend the time between refuelings by 50 to 100%. While the number of fuel assemblies that would be replaced during each refueling are increased, the total number of spent fuel assemblies generated over the lifetime of the facility would be reduced. In planning fuel cycles, however, there are other factors that have to be taken into consideration other than just minimizing the number of spent fuel assemblies to be generated. Utilities normally try to schedule refuelings during the spring and fall to avoid having the facility down during peak load periods. The Commission and National Codes (e.g., the ASME Boiler and Pressure Vessel Code) require periodic tests and inspections of components and systems; to reduce the cost of replacement power, it is prudent to schedule the tests and inspections that require an extended plant shutdown to coincide with a refueling outage.

TVA is conducting a technical feasibility study on the use of an 18 month fuel cycle, in place of the current annual cycle, for Browns Ferry Units 1 and 2. If these results are favorable, TVA will evaluate 18 month cycles as a planning basis for all Browns Ferry units. This study is based on designing for the same burnup as with the present fuel cycles (i.e., average exposure of 26,000 MWD/MTU at 23 KW/KgU). Preliminary results indicate that on an 18 month cycle, 272 fuel assemblies would be replaced at each refueling compared to 204 assemblies used for design purposes with the present fuel cycle. If 204 fuel assemblies are replaced annually, at the end of 4 years, 816 spent fuel assemblies would be generated. If 272 assemblies are replaced every 18 months, 816 spent assemblies would be generated in 4 1/2 years. If the Commission were to approve the proposed action to increase the storage capacity of the SFP's to 3471 assemblies each, discharges at the annual cycle rate will fill the SFP's, less reserve for one full core (764 assemblies), in

thirteen cycles (years). Similarly, an 18 month cycle would fill the pools in ten cycles (15 years), adding up to three years to the time when the pools would be filled to the point that the units would have to shutdown. If the technology is developed to support higher burnups, and heat fluxes, the generation of spent fuel would be further reduced.

Extending the fuel cycle is a promising and very likely alternative in the near future. It is not an alternative that can be implemented now. Considering the long lead times on core design and procurement and the present state of technology, the potential reduction in spent fuel generation is not sufficient to obviate the need for the proposed action.

7.5 Reduced Plant Output

If a nuclear facility's electrical output is reduced, the amount of spent fuel generated can be reduced. During 1978, the cumulative capacity factors for units 1, 2 and 3 has been 76.0, 37.1 and 79.2, respectively. Unit No. 2 shutdown for refueling on March 18, 1978. Because of the low capacity factor, only 132 fuel assemblies were replaced rather than the 168 that had been scheduled to be replaced. Nuclear plants are usually base-loaded because of their lower costs of generating a unit of electricity compared to older plants in the system. Reducing the plant output to reduce spent fuel generation is not an economical use of the resources available. The total production costs remain essentially constant, irrespective of plant output, so at a reduced plant output, the unit cost of electricity is increased proportionately. If the full output of the plant is required to meet load demands on the system and TVA is forced to be reduce output because of spent fuel storage restrictions, then TVA would be required to purchase replacement power or operate less cost-efficient fossil units. In either case, the cost to TVA customers would be increased.

7.6 Shutdown of Facility

Storage of spent fuel from Browns Ferry Units 1, 2 and 3 in the existing racks is possible but only for a short period of time. As discussed above, if expansion of the SFP capacity is not approved and if an alternate storage facility is not located, Browns Ferry Units 1, 2 and 3 would only be able to replace a partial core load at the refuelings now scheduled for September 1982 for Unit 1, March 1982 for Unit 2 and September 1982 for Unit 3. Thus, all three units would have to be shutdown in 1983 or 1984 due to a lack of spent fuel storage facilities. Adoption of the 18 month fuel cycle could delay the shutdown for another year. The need for the BFNP has been previously justified. Shutdown of the three Browns Ferry units would result in the cessation of almost 3300 megawatts of electrical energy production.

The licensee in their submittal of December 2, 1977 stated that replacement power (if available at all) is expected to cost an average of at least 16 mills per kilowatt-hour greater than the cost of generation from the Browns Ferry reactors. Shutting down one reactor is estimated to result in additional costs of at least $9 million per month. Replacement of the generating capability that would be lost by shutting down the Browns Ferry reactors would be many times more expensive than the proposed modification.

7.7 Summary of Alternatives

In summary, alternatives (1) and (2) described above (reprocessing and shipment to an existing storage facility) are not presently available to the licensee. Alternative (3) (shipment to another TVA nuclear facility) cannot be made available in time to meet the licensee's needs. Alternative (5) (reducing plant output) is available but would be more expensive than the proposed modification and does not offer any advantages in terms of environmental impacts. Alternative (4) (lengthening the fuel cycle) is being evaluated and probably will be adopted; depending on the development of technial supporting data on higher burnups, this could reduce the amount of spent fuel generated over the next 15 years by 12 to 20%; however, this alternative cannot be implemented now and cannot be used to substitute for the immediate short term need for additional storage capacity. The alternative of ceasing operation of the facility would be much more expensive than the proposed action because of the need to provide replacement power. In addition to the economic advantages of the proposed action, we have determined that the expansion of the storage capacity of the spent fuel pool for BFNP would have a negligible environmental impact. Accordingly, deferral or severe restriction of the proposed action would result in substantial harm to the public interest.

The proposed modifications accomplish the design objective of providing the required storage capacity while at the same time making more efficient use of the existing facilities at BFNP and minimizing costs of capital, environmental effects, and resources committed. None of the alternatives available presently would provide the storage capacity required to support continued operation of BFNP and none result in lower overall costs. The only alternatives presently available are a plant shutdown, or reduced plant output, which are economically not viable. Offsite storage alternatives, should they become available, would require relatively high capital expenditures. Environmental costs and resources committed for the proposed

modifications are minimal and in general would result regardless of where the spent fuel would be stored. The proposed modifications have advantages in several areas such as land use and increased time for decay prior to shipment.

8.0 Evaluation of Proposed Action

8.1 Unavoidable Adverse Environmental Impacts

8.1.1 Physical Impacts

As discussed above, expansion of the storage capacity of the SFP would not result in any significant adverse environmental impacts on the land, water, air or biota of the area.

8.1.2 Radiological Impacts

As discussed in Section 5.4, expansion of the storage capacity of the SFP will not create any significant additional radiological effects. The additional total body dose that might be received by an individual or the estimated population within a 50-mile radius is less than 0.005 mrem/yr and 0.005 man-rem/yr, respectively. These exposures are small compared to the fluctuations in the annual dose this population receives from background radiation and represent an increase of less than 0.5% of the exposures from the plant evaluated in the FES. The total occupational exposure of workers during removal of the present storage racks and installation of the new racks is estimated by the licensee to be about 40 man-rem for the three units. This is a small fraction of the total man-rem burden from occupational exposure at the plant. Operation of the plan with additional spent fuel in the SFP is not expected to increase the occupational radiation exposure by more than one percent of the present total annual occupational exposure at this facility.

8.2 Relationships Between Local Short-Term Use of Man's Environment and the Maintenance and Enhancement of Long-Term Productivity

Expansion of the storage capacity of the SFP, which would permit the plant to continue to operate until at least 1995, when offsite storage facilities are expected to be available for interim or long-term storage of spent fuel, will not change the evaluation in the FES.

8.3 Irreversible and Irretrievable Commitments of Resources

8.3.1 Water, Land and Air Resources

The proposed action will not result in any significant change in the commitments of water, land and air resources as identified in the FES. No additional allocation of land would be made; the land area now used for the SFP would be used more efficiently by reducing the spacings between fuel assemblies.

8.3.2 Material Resources

Under the proposed modification, the present storage racks in the SFP will be replaced by new fuel storage modules. The new modules will be fabricated stainless steel structures composed of fuel storage tubes, which are made by forming an outer tube and an inner tube of 304 stainless steel which encapsulate plates of Boral on each side of the tube. The Boral consists of a B4C-Al matrix bonded between two layers of aluminum. The inner and outer tubes are welded together. The completed storage tubes are fastened together by angles welded along the corners and attached to a base plate to form storage modules. Spent fuel assemblies are stored both within the tubes and in the spaces between the tubes. Two module sizes will be used in the Browns Ferry SFPs, a 13 x 13 module that will store a total of 169 fuel assemblies (84 in tubes and 85 in spaces outside the tubes) and a 13 x 17 module that will store 221 assemblies. Each SFP will contain fourteen of the 13 x 13 modules and five of the 13 x 17 modules when all of the existing storage racks are replaced with the new high density racks.

Storage will be provided for canned defective fuel and used control rods in each SFP. There will be five extra positions in each pool for storage of defective fuel. Control rod storage will be provided by supplying 20 permanent storage locations in the Units 1 and 2 SFP's and 18 locations in the Unit 3 SFP, and an aggregate of 370 temporary storage locations.

The arrangement of the high density fuel storage system for the spent fuel pools is shown in Figures 1, 2 and 3. The relatively small quantities of material resources being committed would not significantly foreclose the alternatives with respect to other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity. The principal material resources that will be consumed by the proposed modification together with estimated annual domestic consumption are indicated below.

Material	Browns Ferry Modification Quantity (lbs.)	Annual U.S. Consumption (lbs.)
304 Stainless Steel	1.12×10^6	2.82×10^{11}
Boron Carbide	2.71×10^4	3 to 9×10^5
Aluminum	1.25×10^5	8×10^9

Stainless steel and aluminum are readily available in abundant supply. The amount of stainless steel and aluminum required for fabrication of the new racks is a small amount of these resources consumed annually in the United States. Also, the 13 existing aluminum racks which have been removed from the Unit 3 SFP are available as scrap to off-set the net usage. Boron is also available in abundant supply. Boron carbide is primarily used in the nuclear industry. There has been a limited requirement for this material, primarily in high density spent fuel pool storage racks. The material could be made available in much greater quantities if there were a demand for it. We conclude that the amount of material required for the new Browns Ferry racks will not create a significant impact on other potential uses for the materials and does not represent a significant irreversible commitment of material resources.

The longer term storage of spent fuel assemblies withdraws the unburned uranium from the fuel cycle for a longer period of time. Its usefulness as a resource in the future, however, is not changed. The provision of longer onsite storage does not result in any cumulative effects due to plant operation since the throughput of materials does not change. Thus, the same quantity of radioactive material will have been produced when averaged over the life of the plant. This licensing action would not constitute a commitment of resources that would affect the alternatives available to other nuclear power plants or other actions that might be taken by the industry in the future to alleviate fuel storage problems. No other resources need be allocated because the design characteristics of the SFP remain unchanged.

We conclude that the expansion of the SFP at the Browns Ferry facility does not constitute a commitment of either material or nonmaterial resources that would tend to significantly foreclose the alternatives available with respect to any other individual licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity.

8.4 Commission Policy Statement Regarding Spent Fuel Storage

On September 16, 1975, the Commission announced (40FR42801) its intent to prepare a generic environmental impact statement on handling the storage of spent fuel from light water reactors. In this notice, the Commission also announced its conclusion that it would not be in the public interest to defer all licensing actions intended to ameliorate a possible shortage of spent fuel storage capacity pending completion of the generic environmental impact statement. The draft statement was issued for comment on March 17, 1978, (Draft Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel" NUREG-0404, March 1978).

The Commission directed that in the consideration of any such proposed licensing action, among other things, the following five specific factors should be applied, balanced, and weighed in the context of the required environmental statement or appraisal:

1. Is it likely that the licensing action proposed here would have a utility that is independent of the utility of other licensing actions designed to ameliorate a possible shortage of spent fuel capacity?

   A reactor core for BFNP contains 764 fuel assemblies. Typically, the reactor is refueled annually. Each refueling replaces about 1/4 of the core. The SFP was designed on the basis that a fuel cycle would be in existence that would only require storage of spent fuel for a year or two prior to shipment to a reprocessing facility. Initially, sufficient racks were installed to store 1080 spent fuel assemblies (1.4 cores), which was a typical design basis for BWRs in the late sixties and early seventies. When BFNP was designed, a SFP storage capacity for 1.4 cores was considered adequate. This provided for complete unloading of the reactor even if the spent fuel from a previous refueling were in the pool. While not required from the standpoint of safety considerations, it is a desirable engineering practice to reserve space in the SFP to receive an entire reactor core, should this be necessary to inspect or repair core internals or because of other operational considerations. This is the situation which has or will exist at all three Browns Ferry Units. During the first refuelings of Units 1 and 2 in the fall of 1977 and spring of 1978, respectively, TVA had to unload the complete cores from these units to accomplish the modifications discussed in Section 2.0 of this Appraisal. Unit 3 was shutdown for refueling on September 8, 1978. During this outage, TVA plans to off-load the full core to

modify the control rod drive return line. TVA also plans to off-load the full core from Unit 3 during the fall 1979 refueling shutdown to permit modifications to the feedwater nozzles. During this fall 1979 shutdown, TVA will need storage space for 1180 fuel assemblies in the Unit 3 SFP, including space for new fuel. With the existing racks only providing storage space for 1080 fuel assemblies, there would be an excess of 100 fuel assemblies that could not be stored in the SFP. Aside from the more immediate need to increase the storage capacity of the SFP's to provide space for core off-loads, if expansion of the SFP capacity is not approved and if it is not possible to implement one or more of the alternatives discussed in Section 7., the connecting pools for Units 1 and 2 would be filled after the refuelings of Units 1 and 2 in September 1982 and March 1982, respectively. Similarly, the separate Unit 3 pool would be filled to the point where it would only be possible to replace about 1/3 of the normal core reload in the refueling scheduled for September 1982. If the SFP's were full and the reactors could not be refueled. Units 1, 2 and 3 would have to shutdown in the fall of 1983, the spring of 1983 and early 1984 respectively. Even if DOE obtains Congressional authorization in FY79 to construct an interim storage basin as discussed in Section 7., the facility will not be operational prior to 1984. Storage of spent fuel from the Browns Ferry Units in the onsite spent fuel pools is the only reasonable alternative to allow the plant to continue to operate until the permanent Federal repository is available.

The proposed licensing action (i.e., installing new racks of a design that permits storing more assemblies in the same space) would provide the licensee with additional flexibility which is desirable even if adequate offsite storage facilities hereafter become available to the licensee.

We have concluded that a need for additional spent fuel storage capacity exists at BFNP which is independent of the utility of other licensing actions designed to ameliorate a possible shortage of spent fuel capacity.

2. Is it likely that the taking of the action here proposed prior to the preparation of the generic statement would constitute a commitment of resources that would tend to significantly foreclose the alternatives available with respect to any other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity?

With respect to this proposed licensing action, we have considered commitment of both material and nonmaterial resources. The material resources considered are those to be utilized in the expansion of the SFP. The nonmaterial resources are primarily the labor and talent needed to accomplish the proposed modification.

The increased storage capacity of the BFNP spent fuel pool was also considered as a nonmaterial resource and was evaluated relative to proposed similar licensing actions at other nuclear power plants, fuel reprocessing facilities and fuel storage facilities. We have determined that the proposed expansion in the storage capacity of the SFP is only a measure to allow for continued operation and to provide operational flexibility at the facility, and will not affect similar licensing actions at other nuclear power plants. Similarly, taking this action would not commit the NRC to repeat this action or a related action in 1994, at which time the modified pools are estimated to be full if no fuel is removed.

Preparation of the generic statement was initiated in the fall of 1975. The draft statement, NUREG-0404 was issued in March 1978. As discussed in Section 2.0, there is an immediate need to increase the storage capacity of the SFP's to permit repairs to be made to the facilities. Even if this were not the case, it is necessary to install the permanent racks prior to the 1980 refuelings because of space restrictions. Issuance of the final generic statement and Commission action on the statement is not expected to be completed prior to this time.

We conclude that the expansion of the SFP at BFNP prior to issuance of the final generic statement, does not constitute a commitment of either material or nonmaterial resources that would tend to significantly foreclose the alternatives available with respect to any other individual licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity.

3. Can the environmental impacts associated with the licensing action here proposed be adequately addressed within the context of the present application without overlooking any cumulative environmental impacts?

Potential nonradiological and radiological impacts resulting from the fuel rack conversion and subsequent operation of the expanded SFP at this facility were considered by the staff.

No environmental impacts on the environs outside of the spent fuel storage building are expected during removal of the existing racks and installation of the new racks. The impacts within this building are expected to be limited to those normally associated with metal working activities and to the occupational radiation exposure to the personnel involved.

The potential nonradiological environmental impact attributable to the additional heat load in the SFP was determined to be negligible compared to the existing thermal effluents from the facility.

We have considered the potential radiological environmental impacts associated with the expansion of the SFP and have concluded that they would not result in radioactive effluent releases that significantly affect the quality of the human environment during either normal operation of the expanded SFP or under postulated fuel handling accident conditions.

As listed in NUREG-0020. there are presently 68 facilities that have or are proposing to increase the storage capacity of their onsite SFPs. Because of the limited number of vendors supplying high density storage racks, there has been a "cumulative impact" in terms of the time required to fabricate new racks. Since no significant environmental impact has been identified with any individual licensing action to increase onsite storage capacity, there is no cumulative environmental impact.

4. Have the technical issues which have arisen during the review of this application been resolved:

   This Environmental Impact Appraisal and the accompanying Safety Evaluation respond to the questions concerning health, safety and environmental concerns. The only significant technical issue which arose in connection with this application was the swelling noted in the Monticello racks and this has been resolved with the licensee.

5. Would a deferral or severe restriction on this licensing action result in substantial harm to the public interest?

   We have evaluated the alternatives to the proposed action, including storage of the additional spent fuel offsite and ceasing power generation from the plant when the existing SFP is full. We have determined that there are significant economic advantages associated with the proposed action and that expansion of the storage capacity of the SFP will have a negligible environmental impact. Deferral or severe restriction of the action here proposed would result in increased costs to TVA customers and potential shortage of needed electrical energy. We conclude that deferral or severe restriction of the proposed action would result in substantial harm to the public interest.

9.0 Benefit-Cost-Balance

This section summarizes and compares the cost and the benefits resulting from the proposed modification to those that would be derived from the selection and implementation of each alternative. Table 2 presents a tabular comparison of these costs and benefits. The benefit that would be derived from seven of these alternatives would be the continued operation of the plant and production of electrical energy - if the alternative is available. With the present storage capacity of the SFPs, only two alternatives, (other than the proposed action) - lengthening the fuel cycle and reduction in plant output - offer the potential to extend the time at which the plant would be forced to shutdown. As shown in Table 2, reactor shutdown and subsequent storage of fuel in the reactor vessel results in the cessation of electrical energy production. While this would have the "benefit" of eliminating thermal, chemical and radiological releases from the plant, these effluents have been evaluated in the FES and it has been determined that the environmental impacts of these releases are not significant. Therefore, there would be no significant environmental benefit in their cessation.

From examination of the table, it can be seen that the most cost-effective alternative is the proposed spent fuel pool modification. As evaluated in the proceeding sections, the environmental impacts associated with the proposed modification would not be significantly changed from those analyzed in the Final Environmental Statement related to operation of the Browns Ferry Nuclear Plant issued on September 1, 1972.

10.0 Basis and Conclusion for not Preparing an Environmental Impact Statement

We have reviewed this proposed facility modification relative to the requirements set forth in 10 CFR Part 51 and the Council of Environmental Quality's Guidelines, 40 CFR 1500.6 and have applied, weighed, and balanced the five factors specified by the Nuclear Regulatory Commission in 40 FR 42801. We have determined that the proposed license amendment will not significantly affect the quality of the human environment and that there will be no significant environmental impact attributable to the proposed action other than that which has already been predicted and described in the Final Environmental Statement for the facility dated September 1972. Therefore, the staff has found that an environmental impact statement need not be prepared, and that pursuant to 10 CFR 51.5(c), the issuance of a negative declaration to this effect is appropriate.

Dated: September 21, 1978

TABLE 1

REFUELING SCHEDULES

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

BASIS: ANNUAL REFUELINGS

BROWNS FERRY 1

Refueling Date	Number of Fuel Assemblies Discharged	Cumulative Number of Fuel Assemblies in SFP
Sept. 1977	168	168
Sept. 1978	220	388
Sept. 1979	196	584
Sept. 1980	196	780
Sept. 1981	204	984
Sept. 1982	200	1184
Sept. 1983	200	1384
Sept. 1984	200	1584
Sept. 1985	200	1784
Sept. 1986	200	1984
Sept. 1987	200	2184
Sept. 1988	200	2384
Sept. 1989	200	2584
Sept. 1990	200	2784
Sept. 1991	200	2984
Sept. 1992	200	3184
Sept. 1993	200	3384
Sept. 1994	200	3584*

*Units 1 and 2 have separate spent fuel pools. However, they are connected so that fuel can be transferred between the two pools. After the refueling of Unit 2 in March 1994, there would be 123 storage spaces left in the Unit 2 SFP. The refueling of Unit 1 in September 1994 is contingent on using 113 of the 123 spaces in the Unit 2 SFP.

TABLE 1 (Continued)

BROWNS FERRY 2

Refueling Date	Number of Fuel Assemblies Discharged	Cumulative Number of Fuel Assemblies in SFP
March 1978	132	132
March 1979	220	352
March 1980	196	548
March 1981	196	744
March 1982	204	948
March 1983	200	1148
March 1984	200	1348
March 1985	200	1548
March 1986	200	1748
March 1987	200	1948
March 1988	200	2148
March 1989	200	2348
March 1990	200	2548
March 1991	200	2748
March 1992	200	2948
March 1993	200	3148
March 1994	200	3348

BROWNS FERRY 3

Refueling Date	Number of Fuel Assemblies Discharged	Cumulative Number of Fuel Assemblies in SFP
Sept. 1978	208	208
Sept. 1979	208	416
Sept. 1980	188	604
Sept. 1981	200	804
Sept. 1982	200	1004
Sept. 1983	200	1204
Sept. 1984	200	1404
Sept. 1985	200	1604
Sept. 1986	200	1804
Sept. 1987	200	2004
Sept. 1988	200	2204
Sept. 1989	200	2404
Sept. 1990	200	2604
Sept. 1991	200	2804
Sept. 1992	200	3004
Sept. 1993	200	3204
Sept. 1994	200	3404

TABLE 2

SUMMARY OF COST-BENEFITS

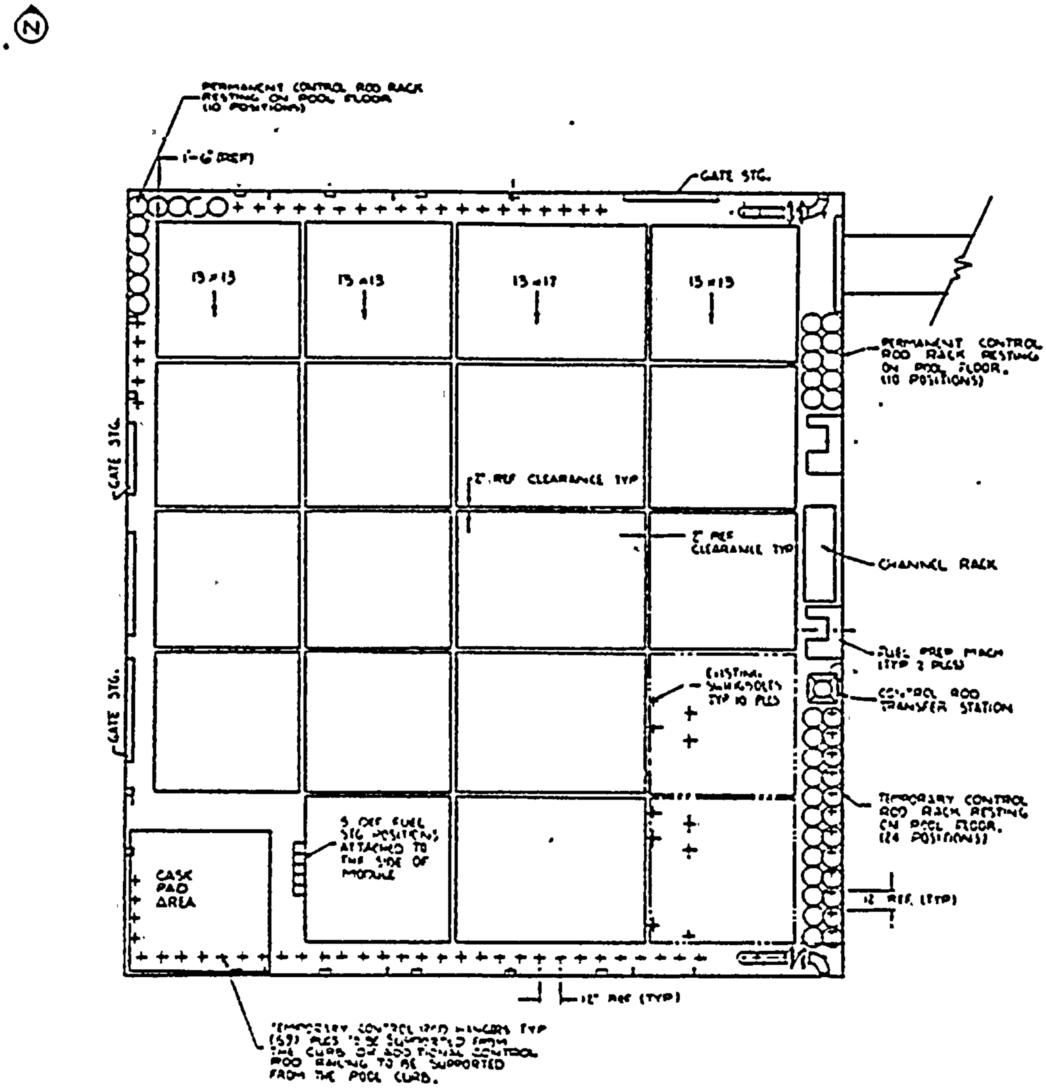
Alternative		Cost	Benefit
Reprocessing of Spent Fuel	>	$10,000/assembly	Continued operation of BFNP and production of electrical energy. This alternative is not available either now or in the foreseeable future.
Increase storage capacity of BFNP		$1,825/assembly ($2650 for each additional storage space)	Continued operation of BFNP and production of electrical energy.
Construction and storage at Independent Facility	>	$4,000/assembly	Continued operation of BFNP and production of electrical energy. There have been proposals - but no applications - for on-site and AFR storage facilities. This alternative could not be available within the next six years.
Storage at Reprocessor's Facility*		$3,000 to $6,000/ assembly plus shipping costs to facility and annual operating costs	Continued operation of BFNP and production of electrical energy. This alternative is not available now or in the foreseeable future.
Storage of Other Nuclear Plants		Comparable to storage at BFNP	Continued operation of BFNP and production of electrical energy. However, this alternative is not available.
Lengthening Fuel Cycle		$1,000 per storage space saved**	Continued operation of BFNP and production of electrical energy. Not available now but will probably be implemented in near future.
Reduction in Plant Output		See below for replacement power costs. Amount of replacement power required would	Continued operation of plant and production of electrical energy - but at higher unit cost.

TABLE 2 (Continued)

Alternative	Cost	Benefit
	depend on the reduction in plant output.	
Reactor Shutdown	Replacement power costs are estimated to be as much as $324 million/year if all three units are shutdown plus $30 million/year for maintenance and security of the plant.	No significant benefit since there is no significant environmental impact associated with plant operation.

*Since NFS and MO are not accepting spent fuel for storage, cost range reflects prices that were quoted in 1972 to 1974. GE estimates that if they were to accept spent fuel today on a temporary basis until a utility could locate other storage space, it would probably be at the rate of $30,000 per MTU, which equates to about $6,000 per BWR assembly. Transportation of the spent fuel would add about $2,000 per assembly.

**Based on estimated R&D costs, differential fuel costs and costs for revised ECCS and reload analyses.

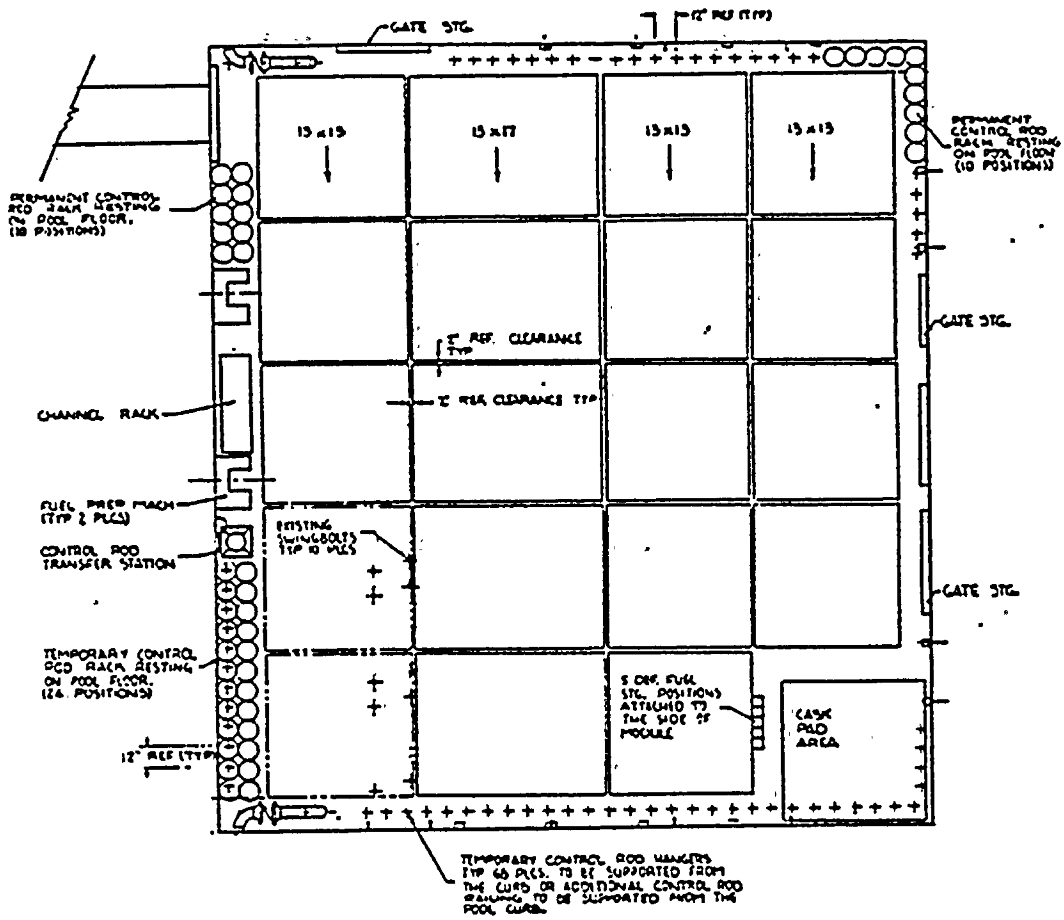


Fuel Storage	3471
Defective Fuel Storage	5
Control Rod Storage	
Perm. Racks	20
Temp. Rack	24
Temp. Hangers	69
	113

STORAGE ARRANGEMENT

BFNP - UNIT 1

FIGURE 1

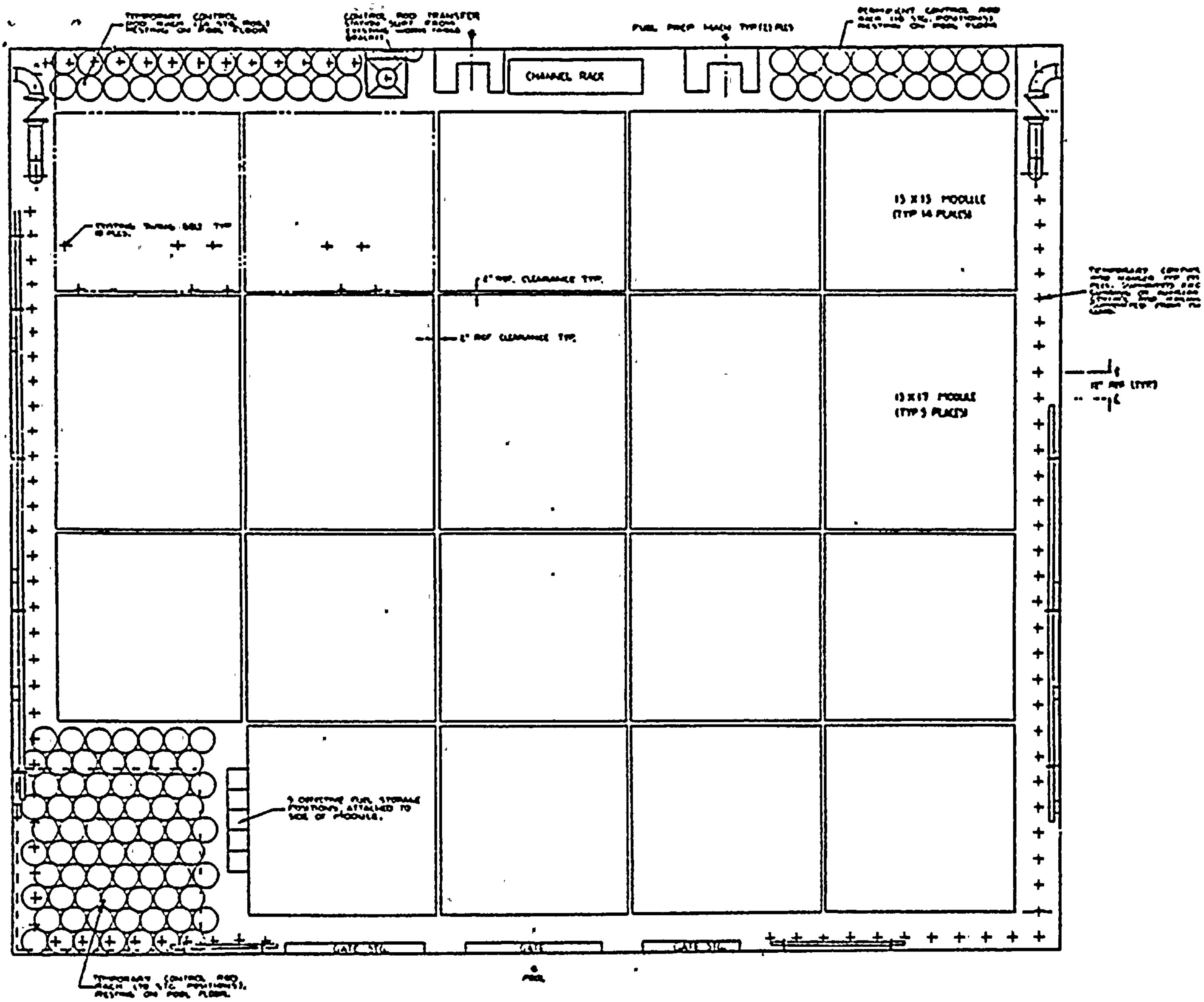


Fuel Storage	3471
Defective Fuel Storage	5
Control Rod Storage	
Perm. Racks	20
Temp. Rack	24
Temp. Hangers	68
	112

STORAGE ARRANGEMENT

BFNP - UNIT 2

FIGURE 2



Fuel Storage	3471
Defective Fuel Storage	5
Control Rod Storage	
Perm. Rack	18
Temp. Racks	94
Temp. Hangers	91
	203

STORAGE ARRANGEMENT

BFNP - UNIT 3

FIGURE 3

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 16 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

In their submittal of December 2, 1977, supplemented by letters dated December 20, 1977, May 24, 1978, May 26, 1978, June 30, 1978, August 2, 1978, August 10, 1978 and September 1, 1978, Tennessee Valley Authority (TVA or the licensee) requested amendments to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant (BFNP), Units Nos. 1, 2 and 3, respectively. The requested amendments would authorize up to 3471 spent fuel or new fuel assemblies to be stored in each of the three onsite spent fuel pools (SFP) by removing the 54 storage racks that are presently in each pool and replacing them in stages with 19 new racks which are designed for closer center-to-center spacing of the spent fuel assemblies. These amendments would increase the amount of spent fuel that could be stored in each SFP from 1080 to 3471 assemblies.

Notice of Proposed Issuance of these Amendments to Facility Operating Licenses No. DPR-33, DPR-52 and DPR-68 was published in the FEDERAL REGISTER on January 9, 1978 (43 FR 1412).

2.0 Discussion

The proposed amendments would modify the single sentence in paragraph 5.5.B of the Technical Specifications on "Fuel Storage" which now states that the k_{eff} of the spent fuel pool shall be less than or equal to 0.90 for normal conditions and 0.95 for abnormal conditions. As revised, the requirement will state that the k_{eff} of the spent fuel storage pool shall be less than or equal to 0.95. A similar change has been approved for 33 other facilities over the past six years and has been determined by experience to provide an adequate margin of safety. We proposed, and the licensee accepted, a requirement to limit the fuel loading on assemblies stored in the SFP.

Our review and evaluation considered the following:

1. Structural and material considerations
2. Criticality considerations
3. Spent fuel pool cooling capacity
4. Fuel handling and installation of the modified spent fuel racks
5. Occupational radiation exposure and radioactive waste treatment

3.0 Evaluation

3.1 Criticality Considerations

3.1.1 Criticality Discussion

The proposed spent fuel assembly racks are to be made up of alternating stainless steel containers. Thus, there will be only one container wall between adjacent spent fuel assemblies. Each container wall is to have a core of Boral sandwiched between 0.036 inch inside and 0.090 inch outside stainless steel containers. The containers will be about 14 feet long and will have a square cross section with an outer dimension of 6.653 inches and a total wall thickness of 0.2015 inches. The nominal pitch between fuel assemblies will be 6.563 inches.

The Boral core is made up of a central segment of a 0.056 inch thick dispersion of boron carbide in aluminum. This central segment is clad on both sides with 0.010 inches of aluminum. TVA states that the minimum homogeneous concentration of the boron-ten isotope will be 0.013 grams per square centimeter of the Boral plate. This is equivalent to 0.78×10^{21} boron-ten atoms per square centimeter. These Boral plates are to be sealed between two stainless steel containers, by welding.

3.1.2 Criticality Analyses

The TVA fuel pool criticality calculations are based on an unirradiated BWR fuel assembly with no burnable poison and a fuel loading of 15.2 grams of uranium-235 per axial centimeter of fuel assembly.

The General Electric Company (GE) performed the criticality analyses for TVA. GE made the calculations with the MERIT Monte Carlo program with cross sections which were processed from ENDF/B-IV data. The accuracy of this calculational method was assessed by using it to calculate the following experiments: (1) thermal reactor benchmark experiments TRX-1 through 4 of the Cross Section Evaluation Work Group; (2) the Babcock and Wilcox UO_2 critical assemblies; and (3) the Oyster Creek BWR experiments with boron curtains. From this qualification program, GE determined that this calculational method underpredicts k_{eff} by 0.5 per cent Δk (0.005k).

GE used these computer programs to calculate the neutron multiplication factor for an infinite array of fuel assemblies in the nominal storage lattice at 20°C with the minimum boron concentration in the Boral, i.e., 0.013 grams of boron-ten per square centimeter and to calculate the k_∞ for the minimum possible pitch [i.e., 6.503 inches] and found it to be 0.87.

GE then calculated the k_∞'s for the following conditions: (1) increasing the temperature to 65°C; (2) increasing the lattice pitch; (3) locating every four fuel assemblies as close together as possible; and (4) reducing the density of the water. GE found that all of these changes resulted in a decrease in k_∞.

Because of the alternating lattice design, wherein there will be only one storage container for every two fuel assemblies, there will be spaces on the periphery of the rack modules which will not have Boral plates. Thus it will be possible for two rack modules to be put together so that adjacent fuel assemblies will not have a Boral plate between them. GE calculated the effect of these missing Boral plates for the minimum attainable gap between rack modules and found that it would not increase the maximum k_∞ of 0.87. GE also analyzed the situation where a fuel assembly is moved as close as possible to an unpoisoned location on the periphery of a filled storage rack and found that the neutron multiplication factor would not increase above 0.90.

TVA also states the following:

> "The presence of the neutron absorber material in the fabricated fuel storage module will be verified at the reactor storage-pool site by use of a neutron source and neutron detectors. There will be a permanent record of all test results that will provide a comparison between the test results for each Boral sheet and the neutron absorption rate taken where there is no Boral sheet. A significant increase in the neutron absorption rates will verify the presence of Boral. Module subcriticality calculations have demonstrated $k_{eff} < 0.95$ at 95% confidence level with any four complete Boral sheets missing. A module will be accepted unless measurements indicate that five or more Boral sheets are not present."

3.1.3 Criticality Evaluation

GE's use of discrete fuel pins in its calculational model for the MERIT Monte Carlo program should result in a more precise value for k_∞. By assuming new, unirradiated fuel with no burnable poison or control rods, these calculations yield the maximum neutron multiplication factor that could be obtained throughout the life of the fuel assemblies. This includes the effect of the plutonium which is generated during the fuel cycle. We conclude that acceptable methods of analyses have been used in the criticality determinations.

The NRC acceptance criteria for the criticality aspects of high density fuel storage racks is that the neutron multiplication factor in spent fuel pools shall be less than or equal to 0.95, including all uncertainties, under all conditions throughout the life of the racks. This 0.95 acceptance criterion is based on the overall uncertainties associated with the calculational methods. We have concluded that this provides sufficient margin to preclude criticality in fuel pools. Accordingly, there is a technical specification which limits the neutron multiplication factor, k_{eff}, in spent fuel pools to a maximum of 0.95.

Since the neutron multiplication factor in spent fuel pools is not a quantity which is measured with good accuracy, the only available value is a calculated one. To preclude any unreviewed increase, or increased uncertainty, in the calculated value of the neutron multiplication factor which could raise the actual k_{eff} in the fuel pool above 0.95 without being detected, a limit on the maximum fuel loading is also required. Accordingly, we find that the proposed high density storage racks will meet the NRC criteria when the fuel loading in the assemblies described in these submittals is limited to 15.2 grams or less of uranium-235 per axial centimeter of fuel assembly.

We conclude that TVA proposed quality assurance program to test the neutron attenuation of each tube in each rack will detect if there are any Boral plates missing from the prescribed locations in the fabricated fuel storage modules.

3.1.4 Criticality Summary

We find that when any number of the fuel assemblies, which TVA described in these submittals, which have no more than 15.2 grams of uranium-235 per axial centimeter of fuel assembly, are loaded into the proposed racks, the k_{eff} in the fuel pool will be less than the 0.95 limit. We also find that in order to preclude the possibility of the k_{eff} in the fuel pool from exceeding this 0.95 limit without being detected, it is necessary, pending an NRC review, to prohibit the use of these high density storage racks for fuel assemblies that contain more than 15.2 grams of uranium-235 per axial centimeter of fuel assembly. On the basis of our evaluation, and the k_{eff} and fuel loading limits stated above we conclude that the health and safety of the public will not be endangered by the use of the proposed racks.

3.2 SPENT FUEL COOLING

3.2.1 Discussion of Cooling System

The licensed thermal power for each of the three Browns Ferry Reactors is 3293 MWt. TVA is presently refueling these plants annually, but it is studying an 18 month refueling cycle. In the annual cycle, about 204 of the 764 fuel assemblies in the core are replaced. In the 18 month cycle, the number replaced would go up to 272. TVA assumed an 8 day time interval (5 days of preparation and 3 days of unloading time) between reactor shutdown and the time when 204 fuel assemblies were transferred to the spent fuel pool and a 16 day time interval between reactor shutdown and the time a full core offload was completed. For the power history prior to refueling, TVA assumed an energy production of 26,000 MWD/MTU obtained with a continuous energy density of 23 KW/kgu. With these assumptions TVA used the ORIGEN program to calculate the maximum possible heat loads for the modified spent fuel pools. These are graphically shown to be about 14.5 x 10^6 BTU/hr for the annual refueling and about 29 x 10^6 BTU/hr for a full core offload.

As indicated in Table 10.5-1 of the FSAR, the spent fuel pool cooling system for each pool consists of two pumps and two heat exchangers in parallel. Each pump is designed to pump 600 gpm (3 x 10^5 pounds per hour). Also, as stated by TVA in response to our request for additional information each heat exchanger is designed to transfer 4.4 x 10^6 BTU/hr from 125°F fuel pool water to 100°F Reactor Building Closed Cooling System water, which is flowing through the heat exchanger at a rate of 3.75 x 10^5 pounds per hour. For higher heat loads, such as the full core offload, TVA states that the residual heat removal system (RHR), with a capacity of 18.8 x 10^6 BTU/hr, will be operated in parallel with the spent fuel pool cooling system.

In its response to our request for additional information, TVA states that emergency makeup water for the spent fuel pool could be obtained from fire hoses at six stations at approximately 95 gpm from each station.

3.2.2 Cooling Evaluation

Using the method given on pages 9.2.5-8 through 14 of the NRC Standard Review Plan, with the uncertainty factor, K, equal to 0.1 for decay times longer than 10^3 seconds, we calculate that the maximum peak heat load during the seventeenth annual refueling could be 13.4 x 10^6 BTU/hr and that the maximum peak heat load for a full core offload that fills the pool could be 28.4 x 10^6 BTU/hr. This full core offload was assumed to take place one year after the year 1991 (i.e., the nineteenth) annual refueling. We also find that the maximum incremental heat load that could be added by increasing the number of spent fuel assemblies in the pool from, 1,080 to 3,471 will be 3.4 x 10^6 BTU/hr. This is the difference in peak heat loads for full core offloads that essentially fill the present and the modified pools.

The present Technical Specifications (3.10.C) require that the pool water temperature be $\leq 150°F$. We calculate that with both pumps operating, the spent fuel pool cooling system can maintain the fuel pool outlet water temperature below 138°F for a peak annual refueling heat load of 13.4×10^6 BTU/hr. We find that when the RHR system is aligned with the spent fuel pool cooling system, the combined system will have sufficient capacity to keep the spent fuel pool outlet water temperature below 150°F for a full core heat load of 29×10^6 BTU/hr.

Assuming a maximum fuel pool temperature of 150°F, the minimum possible time to achieve bulk pool boiling after any credible accident will be about seven hours. After bulk boiling commences, the maximum evaporation rate will be 58 gpm. We conclude that seven hours provides sufficient time for TVA to establish a 58 gpm make up rate from the fire hoses even if the normal sources of makeup water are not available. We also find that under bulk boiling conditions the temperature of the fuel will not exceed 350°F. This is an acceptable temperature from the standpoint of fuel element integrity and surface corrosion.

3.2.3 Cooling Summary

We find that the present cooling capacities in the spent fuel pools of the Browns Ferry Nuclear Plant will be sufficient to handle the incremental heat loads that will be added by the proposed modifications. We also find that these incremental heat loads will not alter the safety considerations of spent fuel pool cooling from that which we previously reviewed and found to be acceptable. We conclude that there is reasonable assurance that the health and safety of the public will not be endangered by the use of the proposed design.

3.3 Fuel Handling and Installation of Racks

3.3.1 Installation Discussion

There are presently 168 spent fuel assemblies stored in the Unit 1 SFP and 132 spent fuel assemblies in the Unit 2 SFP. After the refueling shutdown of Unit 1 scheduled for November 1978, the Unit 1 SFP will have 388 assemblies in the pool. The present storage capacity of each SFP is 1080 assemblies. The spent fuel presently stored in each pool only occupies one corner and removal of the old racks and installation of new racks could be accomplished without moving these racks over stored spent fuel. The Units 1 and 2 pools are connected by a fuel transfer slot. As discussed later, we are amending the Technical Specifications to prohibit loads greater than 1000 lbs. from being carried over spent fuel stored in the SFP. This would preclude the new or present racks from being carried over spent fuel in the pools. TVA could accomplish the modification with this restriction leaving the spent fuel in the pools (as most other licensees have done). However, as a precautionary measure, TVA states that they will transfer the Unit 2 spent fuel to the Unit 1 pool prior to changing the racks in Unit 2 and vice versa. Thus, the rack changes in these two pools will be done without any fuel assemblies in the pool.

Unit 3 shutdown for refueling on September 8, 1978. During this outage, the entire core is scheduled to be off-loaded to permit modifications to the control rod drive return line. At the completion of the modification, the reactor will be refueled, leaving 208 spent fuel assemblies in the pool. Prior to the refueling shutdown, while the pool was dry and not contaminated by exposure to radioactivity, TVA removed 13 of the 54 existing racks in the pool and installed 4 of the new racks. The existing racks are the standard 20 element BWR racks described in Section 10.3 of the Final Safety Analysis Report (FSAR) for BFNP. There is sufficient space (820 storage locations) in the remaining existing racks to accommodate the entire core of 764 fuel elements. Removing the 13 racks keeps these racks from becoming contaiminated and reduces the volume of low level radioactive waste that would have to be shipped offsite for burial. In accordance with the Commission's objective to maintain occupational radiation exposures as low as reasonably achievable (ALARA), removal and cutting up of these 13 racks and installation of the 4 new racks before spent fuel is transferred into the pool will reduce the total occupational exposure. TVA will not use the new racks for storage of spent fuel until their use is approved by the Commission. Assuming that use of the new racks is authorized, TVA will remove the remaining 41 old racks in the Unit 3 SFP and install 15 additional new racks.
The Standard Technical Specifications for BWRs (Section 3.9.7) limits the weight of loads carried over spent fuel assemblies stored in the SFP racks to 2500 pounds, which is approximately the weight of one assembly with channels plus associated load handling tools. TVA is using lighter load handling tools on the refueling bridges, Accordingly, the Browns Ferry Technical Specifications are being amended to limit the weight of loads carried over spent fuel to 1000 pounds.

3.3.2 Installation Evaluation

The procedures to be followed during removal of the existing racks and installation of the new racks include removal of all spent fuel from the Units 1 and 2 SFPs during the modification and limiting the weight of loads which may be carried over spent fuel stored in the Unit 3 SFP. These actions will prevent an accident which could result in any increased multiplication factor.

3.3.3 Installation Summary

We conclude that there is reasonable assurance that the health and safety of the public will not be endangered by the installation and use of the proposed racks.

3.4 Radiological Considerations

3.4.1 Fuel Handling Accidents

The NRC staff has under way a generic review of load handling operations in the vicinity of spent fuel pools to determine the likelihood of a heavy load impacting fuel in the pool and, if necessary, the radiological consequences of such an event. Because the Technical Specifications prohibit the movement of loads over spent fuel stored in the pools which significantly exceed the weight of a fuel assembly (i.e., the weight of a fuel assembly and grapple hoist) we have concluded that the likelihood of a heavy load handling accident is sufficiently small that the proposed modification is acceptable and no additional restrictions on load handling operations in the vicinity of the SFP are necessary while our review is under way. The present Technical Specifications on the Spent Fuel Cask (Section 3.10.E) provide adequate restrictions on cask movement.

The consequences of fuel handling accidents in the spent fuel pool area are not changed from those presented in the Safety Evaluation (SE) of the Browns Ferry Nuclear Plant issued by the Commission on June 26, 1972.

3.4.2 Occupational Radiation Exposure

We have reviewed the licensee's plan for the removal, crating and disposal of the low density racks and the installation of the high density racks for each unit with respect to occupational radiation exposure. The occupational radiation exposure for this operation is estimated by the licensee to be about 32 man-rem for Units 1 and 2 and about 8 man-rem for Unit 3. We consider this to be a conservative estimate based on the occupational exposures that have been recorded at over two dozen other facilities that have increased the storage capacity of their SFPs. This operation is expected to be performed only once during the lifetime of the plant. It represents a small fraction of the total man-rem burden from occupational exposure at the plant. Based on our review, we conclude the exposure will be as low as is reasonably achievable.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by the licensee on the estimated time required by personnel (e.g., crane operators, riggers, operators, etc.) to accomplish the modification and by utilizing relevant assumptions for occupancy times and for dose rates in the spent fuel area from radionuclide concentrations in the SFP water. The

spent fuel assemblies themselves contribute a negligible amount (less than 1 mr/hr) to dose rates in the pool area because of the depth of water shielding the fuel. The occupational radiation exposure resulting from the additional spent fuel in the pool represents a negligible burden. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification will add less than one percent to the total annual occupational radiation exposure burden at this facility. The small increase in radiation exposure will not affect the licensee's ability to maintain individual occupational doses to as low as is reasonably achievable and within the limits of 10 CFR 20. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

The estimated radiation exposure to off-site personnel is discussed in the accompanying environmental impact appraisal.

3.4.3 Radioactive Waste Treatment

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid wastes that might contain radioactive material. The waste treatment systems were evaluated in the Safety Evaluation (SE) dated June 1972. As discussed in the accompanying environmental impact appraisal, there will be no change in the type of radioactive effluents and no significant change in their amounts. No changes in the waste treatment systems are required to process these effluents. There is no change in our conclusions and evaluation of these systems as described in Section 8.0 of the SE because of the proposed modification.

3.4.4 Summary of Accidents and Radiological Considerations

Our Evaluation supports the conclusion that the proposed modifications to the Browns Ferry Units 1, 2 and 3 Spent Fuel Pools are acceptable because:

(1) The increase in occupational radiation exposure to individuals due to the storage of additional fuel in the SFP would be negligible.

(2) The installation and use of the new fuel racks does not alter the potential occurrence or the consequences of the design basis accident for the SFP, i.e., the rupture of a fuel assembly and subsequent release of the assembly's radioactive inventory within the gap.

(3) The restriction on carrying heavy loads over spent fuel which is being incorporated in the Technical Specifications by this amendment will preclude the likelihood of an accident involving heavy loads in the vicinity of the spent fuel pool.

3.5 Structural and Material Considerations

The current Browns Ferry fuel storage racks have a storage capacity of 1080 fuel assemblies per pool. The proposed SFP modification consists of installation of new fuel storage modules. Each module is composed of fuel storage tubes arranged in 13 X 13 and 13 X 17 arrays. The new system will provide a capacity of up to 3471 fuel assemblies per pool. The new racks will replace the existing fuel storage and control rod storage racks. The new racks are seismic Category I structures.

Control rod storage will be provided by supplying twenty storage locations in BF-1 and BF-2 and eighteen in BF-3 and 370 temporary storage locations. There will be five extra positions in each pool for defective fuel storage. The pool capacity of 3471 fuel assemblies require fourteen modules of 13 X 13 and five modules of 13 X 17.

The fuel storage tube is fabricated by forming an outer and inner sheet of 304 stainless steel sandwiching a core of Boral (clad by aluminum) into a single rectangular tube. The inner and outer walls of the storage tube are welded together at each end, which isolates the Boral from direct contact with fuel pool water. Except for the Boral and aluminum, all structural material used in fabrication of the new modules is type 304 stainless steel.

The module design, material, and fabrication are in accordance with the requirements set forth in Section III, Subsection NF of the ASME Boiler and Pressure Vessel Code. The modules are designed to remain within Code allowed stress limits for both Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) conditions. The modules were analyzed as cantilever beams attached to a rigid base using qualified computer codes to derive loads in a water filled rectangular pool. These loads were derived for horizontal and vertical accelerations specified in the General Electric BWR Systems Department seismic criteria document and the resulting stresses were compared to the allowable stresses. The analysis indicated that the derived loads do not overstress the modules since the Browns Ferry accelerations at the fuel pool elevation are much less than the accelerations for which the analysis has been performed. For instance the OBE peak acceleration is only 0.25g. The virtual mass effect is not critical. The licensee has however established that small

sliding may occur, but limited to about 0.65 inches in the worse case. Added damping due to fluid effects was conservatively neglected.* Stresses due to seismic loading in the three orthogonal directions were combined by the Square Root of the Sum of the Squares Method as outlined in Regulatory Guide 1.92.

The module design is free-standing, transferring shear forces to the pool slab through friction resistance provided by the normal force of the weight of the module through the support columns resting on the pool floor liner. TVA has used a minimum value for the coefficient of friction in the sliding analysis, a value which was verified by recent tests of steel materials.* The coefficient of friction used was sufficient to ensure that only small sliding will occur for earthquake motions corresponding to OBE and SSE. An additional non-linear analysis for sliding was performed to determine relative displacements if the coefficient of friction were less than the minimum value used. This analysis gives added assurance that there should be no interaction between modules as a consequence of the SSE.

The TVA has re-evaluated the fuel pool structural capacity for the High Density Fuel Storage System and has shown that the existing structure is capable of supporting the increased load with an ample margin of safety.

The new racks which TVA proposes to use at Browns Ferry are identical in design and are supplied by the same manufacturer as those which are being furnished for the Monticello Nuclear Generating Plant. Following installation of four of the 13 x 13 racks in the Monticello SFP, swelling was detected in 10 of the 340 tubes. The swelling was caused by leaks in the tubes, which allowed water to enter the tubes. The water resulted in corrosion of the aluminum cladding, which generated hydrogen.

The tubes in the GE racks are about 14 feet long. Under water, there is a differential pressure of about 5.5 psig between the top and bottom of the tubes due to the hydrostatic head of water. The 36 mil stainless steel tube will withstand about 4.5 psig internal pressure before deforming. If there is a leak at the bottom of a tube which allows water to enter, the hydrostatic head of water prevents the hydrogen from escaping through the same hole until the internal pressure is greater than the hydrostatic head and this pressure is greater than that which deforms the tube. To prevent a buildup of hydrogen within the tubes which could cause swelling, the licensee has drilled a hole in the top of the tubes in the four racks at Browns Ferry Unit No. 3 to prevent swelling in these racks.

*Any possible variations in the coefficient of friction have been covered by the fact that the licensee has used in its analysis a conservatively low value for this parameter.

The presence of water within the tubes of the four moludes which will be used in the Unit 3 SFP will cause corrosion of the Boral. The potential extent of the corrosion attach was evaluated based on corrosion data submitted by Brooks and Perkins, the experience and test results with Boral in the Brookhaven Reactor and experience with Boral in military and test reactors. The available corrosion data is adequate to support the conclusion that corrosion and pitting of the Boral is not a safety concern for the near future. The staff is continuing the evaluation of the corrosion behavior of Boral under coupled and crevice conditions for long-term exposures (i.e., 20 to 30 years) to various aqueous environments. Like most metals, the corrosion rate of aluminum in water is comparatively high during the first few days of exposure and then decreases and essentially levels off as a protective oxide film is built up on the metal. Although no swelling of the tubes is expected since the tubes are vented, as a precautionary measure, TVA has committed to store spent fuel from the September 8, 1978 refueling only in the spaces adjacent to tubes. This restriction will apply until Phase II of the rack replacement program is initiated.

TVA also committed to install corrosion test specimens in the Browns Ferry Unit No. 3 SFP that will be periodically removed and examined to check the long-term corrosion behavior of Boral sandwiched between Type 304 stainless steel.

Since the possibility of long term storage of spent fuel exists, we are also generically investigating further the effects of the pool environment on the modules, fuel cladding and pool liner. Our available corrosion data on the materials used in the proposed racks spans over two decades of service in spent fuel pools or similar environments (e.g., shield water systems). Battelle has recently completed an evaluation of the corrosion behavior of spent fuel stored in pools for over 14 years ("Behavior of Spent Nuclear Fuel in Water Pool Storage", BNWL-2256, September 1977). Based upon our evaluation and previous operating experience, we have concluded that at the pool temperature and the quality of the demineralized water, and taking no credit for inservice inspection, there is reasonable assurance that no significant corrosion of the modules, the fuel cladding or the pool liner will occur over the lifetime of the plant. However, if the results of the current generic review indicate that additional protective measures are warranted to protect the modules, the fuel cladding and/or the liner from the effects of corrosion, the necessary steps and/or inspection programs will be required to assure that an acceptable level of safety is maintained. Any conceivable problems which could be uncovered are of a long term nature and warrant no need for immediate concern.

The criteria used in the analysis, design, and construction of the High Density Fuel Storage System to account for the anticipated loadings and postulated conditions that may be imposed on the structures during their service lifetime are in conformance with established criteria, codes, standards, and specifications for seismic Category I components and are designed to maintain the spent fuel assemblies in a safe configuration through all environmental and abnormal loadings. Therefore, we find that the proposed expansion is acceptable from the aspect of mechanical, material, and structural considerations.

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.

Dated: September 21, 1978

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSE AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 42 to Facility Operating License No. DPR-33, Amendment No. 39 to Facility Operating License No. DPR-52 and Amendment No. 16 to Facility Operating License No. DPR-68, issued to Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3, located in Limestone County, Alabama. The amendments are effective as of date of issuance.

The amendments change the Technical Specifications and authorize the licensee to increase the storage capacity of each of the three on-site spent fuel pools to 3471 fuel assemblies.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on January 9, 1978 (43FR1412). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has prepared an environmental impact appraisal for the amendment and has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the action other than that which has already been predicted and described in the Final Environmental Statement for the facility dated September 1, 1972.

For further details with respect to this action, see (1) the application for amendments dated December 2, 1977, as supplemented by letters dated December 20, 1977, May 24, May 26, June 30, August 2, August 10, and September 1, 1978, (2) Amendment No. 42 to License No. DPR-33, Amendment No. 39 to License No. DPR-52, and Amendment No. 16 to License No. DPR-68, (3) the Commission's related Environmental Impact Appraisal and (4) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 21st day of September, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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

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

AUGUST 2 1978

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

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 40, 38 and 14 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. These amendments consist of changes to the Technical Specifications in response to your request of February 24, 1977 as supplemented by your letter of May 23, 1978.

The amendments change the Technical Specifications to lower the reactor low water level setpoint by 20 inches (i.e., from 490" to 470").

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

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 40 to DPR-33
2. Amendment No. 38 to DPR-52
3. Amendment No. 14 to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures: See page 2

Construct

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cc: H. S. Sanger, Jr., Esquire
General Counsel
Tennessee Valley Authority
400 Commerce Avenue
E 11B 33 C
Knoxville, Tennessee 37902

Mr. D. McCloud
Tennessee Valley Authority
303 Power Building
Chattanooga, Tennessee 37401

Mr. William E. Garner
Route 4, Box 354
Scottsboro, Alabama 35768

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 36104

Mr. C. S. Walker
Tennessee Valley Authority
400 Commerce Avenue
W 9D199 C
Knoxville, Tennessee 37902

Athens Public Library
South and Forrest
Athens, Alabama 35611

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U.S. Environmental Protection Agency
Room 645, East Tower
401 M Street, SW
Washington, D.C. 20460

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS Coordinator
345 Courtland Street
Atlanta, Georgia 30308

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 40
License No. DPR-33

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

A. The application for amendments by Tennessee Valley Authority (the licensee) dated February 24, 1977, as supplemented by letter dated May 23, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 40, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 2, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 40

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

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

   55/56
   61/62
   63/64
   111/112

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

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level (6)	≥ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure	100 ± 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D, SW #1)	≥ 470" above vessel zero.	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PS-64-56A-D)	≤ 2 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
2	Instrument Channel - High Radiation Main Steam Line Tunnel (6)	≤ 3 times normal rated full power background	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Low Pressure Main Steam Line	≥ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel - High Flow Main Steam Line	≤ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation

TABLE 3.2.A (Continued)

Minimum No. Operable Per Sys (1)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Main Steam Line Tunnel High Temperature	$\leq$ 200°F	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Reactor Water Cleanup System Floor Drain High Temperature	160 - 180°F	C	1. Above trip setting initiates Isolation of Reactor Water Cleanup Line from Reactor and Reactor Water Return Line.
2	Instrument Channel - Reactor Water Cleanup System Space High Temperature	160 - 180°F	C	1. Same as above
1	Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	$\leq$ 100 mr/hr or downscale	G	1. 1 upscale or 2 downscale will a. Initiate SGTS b. Isolate reactor zone and refuleing floor. c. Close atmosphere control system.
1	Instrument Channel - Reactor Building Ventilation High Radiation Refuleing Zone	$\leq$ 100 mr/hr or downscale	F	1. 1 upscale or 2 downscale will a. Initiate SGTS. b. Isolate refueling floor. c. Close atmosphere control system.
2 (7)(8)	Instrument Channel SGTS Flow - Train A Heaters	Charcoal Heaters $\leq$ 2000 cfm R. H. Heaters $\leq$ 2000 cfm	H and (A or F)	1. Below 2000 cfm, trip setting charcoal heaters will turn on. 2. Below 2000 cfm, trip setting R. H. heaters will shut off.
2 (7)(8)	Instrument Channel SGTS Flow - Train B Heaters	Charcoal Heaters $\leq$ 2000 cfm R.H. Heaters $\leq$ 2000 cfm	H and (A or F)	1. Below 2000 cfm, trip setting charcoal heaters will turn on. 2. Below 2000 cfm, trip setting R.H. heaters will shut off.
2 (7)(8)	Instrument Channel SGTS Flow - Train C Heaters	Charcoal Heaters $\leq$ 2000 cfm R.H. Heaters $\leq$ 2000 cfm	H and (A or F)	1. Below 2000 cfm, trip setting charcoal heaters will turn on. 2. Below 2000 cfm, trip setting R.H. heaters will shut off.

6. Channel shared by RPS and Primary Containment Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

7. A train is considered a trip system.

8. Two out of three SGTS trains required. A failure of more than one will require action A and F.

9. There is only one trip system with auto transfer to two power sources.

TABLE 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero.	A	1. Below trip setting initiated HPCI.
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #1)	≥ 378" above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #2)	≥ 378" above vessel zero.	A	1. Below trip settings in conjunction with drywell high pressure, low water level permissive, 120 sec. del timer and CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184 & 185, SW #1)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LITS-3-52 & 62, SW #1)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.
2	Instrument Channel - Drywell High Pressure (PS-64-58 E-H)	1≤ p≤ 2 psig	A	1. Below trip setting prevents inadvertent operation of containment.spray during accident conditions.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Drywell High Pressure (PS-64-58 A-D, SW #2)	$\leq$ 2 psig	A	1. Above trip setting in conjunction wi low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal.(15).
2	Instrument Channel - Reactor Low Water Level (LS-3-56A, B, C, D)	$\geq$ 470" above vessel zero	A	1. Below trip setting trips recirculation pumps
2	Instrument Channel Reactor High Pressure (PS-3-204 A, B, C, D)	$\leq$ 1120 psig	A	1. Above trip setting trips recirculation pumps
2	Instrument Channel - Drywell High Pressure (PS-64-58A-D, SW #1)	$\leq$ 2 psig	A	1. Above trip setting in conjunction wi low reactor pressure initiates LPCI.
2(16)	Instrument Channel - Drywell High Pressure (PS-64-57A-D)	$\leq$ 2 psig	A	1. Above trip setting in conjunction wi low reactor water level, drywell hig pressure, 120 sec. delay timer and C or RHR pump running, initiates ADS.
2	Instrument Channel - Reactor Low Pressure (PS-3-74 A & B, SW #2) (PS-68-95, SW #2) (PS-68-96, SW #2)	450 psig $\pm$ 15	A	1. Below trip setting permissive for opening CSS and LPCI admission valves.
2	Instrument Channel - Reactor Low Pressure (PS-3-74A & B, SW #1) (PS-68-95, SW #1) (PS-68-96, SW #1)	230 psig $\pm$ 15	A	1. Recirculation discharge valve actuation.

Amendment No. 40

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
1	Instrument Channel - Reactor Low Pressure (PS-68-93 & 94, SW #1)	100 psig $\pm$ 15	A	1. Below trip setting in conjunction with containment isolation signal and both suction valves open will close RHR (LPCI) admission valves.
2	Core Spray Auto Sequencing Timers (5)	$6 \le t \le 8$ secs.	B	1. With diesel power 2. One per motor
2	LPCI Auto Sequencing Timers (5)	$0 \le t \le 1$ sec.	B	1. With diesel power 2. One per motor
1	RHRSW A3, B1, C3, and D1 Timers	$13 \le t \le 15$ sec.	A	1. With diesel power 2. One per pump
2	Core Spray and LPCI Auto Sequencing Timers (6)	$0 \le t \le 1$ sec. $6 \le t \le 8$ sec. $12 \le t \le 16$ sec. $18 \le t \le 24$ sec.	B	1. With normal power 2. One per CSS motor 3. Two per RHR motor
1	RHRSW A3, B1, C3, and D1 Timers	$27 \le t \le 29$ sec.	A	1. With normal power 2. One per pump

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 177.7" (538" above vessel zero) above the top of the active fuel closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Group 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel closes the Main Steam Line Isolation Valves and Main Steam, RCIC, and HPCI Drain Valves (Group 1 and 7). Details of valve grouping and required closing times are given in Specification 3.7. These trip settings are adequate to prevent core uncovery in the case of a break in the largest line assuming the maximum closing time.

The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel (Table 3.2.B) also initiate the RCIC and HPCI, provides input to the

LPCI loop selection logic and trips the recirculation pumps. The low reactor water level instrumentation that is set to trip when reactor water level is 17.7" (378" above vessel zero) above the top of the active fuel (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus the results given above are applicable here also.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. The setting of 200°F for the main steam line tunnel detector is low enough to detect leaks of the order of 15 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.2 FSAR. An alarm, with a nominal set point of 1.5 x normal full power background, is provided also.

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

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
License No. DPR-52

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

   A. The application for amendments by Tennessee Valley Authority (the licensee) dated February 24, 1977, as supplemented by letter dated May 23, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility License No. DPR-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. 38, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

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

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: August 2, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 38

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

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

   55/56
   61/62
   63/64
   111/112

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

TABLE 3.2.A

PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level (6)	≥ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure	100 ± 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D, SW #1)	≥ 470" above vessel zero.	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PS-64-56A-D)	≤ 2 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
2	Instrument Channel - High Radiation Main Steam Line Tunnel (6)	≤ 3 times normal rated full power background	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Low Pressure Main Steam Line	≥ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel - High Flow Main Steam Line	≤ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation

TABLE 3.2.A (Continued)

Minimum No. Operable Per Sys (1)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Main Steam Line Tunnel High Temperature	≤ 200°F	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Reactor Water Cleanup System Floor Drain High Temperature	160 - 180°F	C	1. Above trip setting initiates Isolation of Reactor Water Cleanup Line from Reactor and Reactor Water Return Line.
2	Instrument Channel - Reactor Water Cleanup System Space High Temperature	160 - 180°F	C	1. Same as above
1	Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	≤ 100 mr/hr or downscale	G	1. 1 upscale or 2 downscale will a. Initiate SGTS b. Isolate reactor zone and refuleing floor. c. Close atmosphere control system.
1	Instrument Channel - Reactor Building Ventilation High Radiation Refuleing Zone	≤ 100 mr/hr or downscale	F	1. 1 upscale or 2 downscale will a. Initiate SGTS. b. Isolate refueling floor. c. Close atmosphere control system.
2 (7)(8)	Instrument Channel SGTS Flow - Train A Heaters	Charcoal Heaters ≤ 2000 cfm R. H. Heaters ≤ 2000 cfm	H and (A or F)	1. Below 2000 cfm, trip setting charcoal heaters will turn on. 2. Below 2000 cfm, trip setting R. H. heaters will shut off.
2 (7)(8)	Instrument Channel SGTS Flow - Train B Heaters	Charcoal Heaters ≤ 2000 cfm R.H. Heaters ≤ 2000 cfm	H and (A or F)	1. Below 2000 cfm, trip setting charcoal heaters will turn on. 2. Below 2000 cfm, trip setting R.H. heaters will shut off.
2 (7)(8)	Instrument Channel SGTS Flow - Train C Heaters	Charcoal Heaters ≤ 2000 cfm R.H. Heaters ≤ 2000 cfm	H and (A or F)	1. Below 2000 cfm, trip setting charcoal heaters will turn on. 2. Below 2000 cfm, trip setting R.H. heaters will shut off.

6. Channel shared by RPS and Primary Containment & Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

7. A train is considered a trip system.

8. Two out of three SGTS trains required. A failure of more than one will require action A and F.

9. There is only one trip system with auto transfer to two power sources.

TABLE 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero.	A	1. Below trip setting initiated HPCI.
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #1)	≥ 378" above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #2)	≥ 378" above vessel zero.	A	1. Below trip settings in conjunction with drywell high pressure, low water level permissive, 120 sec. del timer and CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184 & 185, SW #1)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LITS-3-52 & 62, SW #1)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.
2	Instrument Channel - Drywell High Pressure (PS-64-58 E-H)	1≤ p≤ 2 psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.

Unit 2

Amendment No. 38

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Drywell High Pressure (PS-64-58 A-D, SW #2)	≤ 2 psig	A	1. Above trip setting in conjunction wi low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal.(15).
2	Instrument Channel - Reactor Low Water Level (LS-3-56A, B, C, D)	≥ 470" above vessel zero	A	1. Below trip setting trips recirculation pumps
2	Instrument Channel Reactor High Pressure (PS-3-204 A, B, C, D)	≤ 1120 psig	A	1. Above trip setting trips recirculation pumps
2	Instrument Channel - Drywell High Pressure (PS-64-58A-D, SW #1)	≤ 2 psig	A	1. Above trip setting in conjunction wi low reactor pressure initiates LPCI.
2(16)	Instrument Channel - Drywell High Pressure (PS-64-57A-D)	≤ 2 psig	A	1. Above trip setting in conjunction wi low reactor water level, drywell hig pressure, 120 sec. delay timer and C or RHR pump running, initiates ADS.
2	Instrument Channel - Reactor Low Pressure (PS-3-74 A & B, SW #2) (PS-68-95, SW #2) (PS-68-96, SW #2)	450 psig ± 15	A	1. Below trip setting permissive for opening CSS and LPCI admission valves.
2	Instrument Channel - Reactor Low Pressure (PS-3-74A & B, SW #1) (PS-68-95, SW #1) (PS-68-96, SW #1)	230 psig ± 15	A	1. Recirculation discharge valve actuation.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
1	Instrument Channel - Reactor Low Pressure (PS-68-93 & 94, SW #1)	100 psig $\pm$ 15	A	1. Below trip setting in conjunction with containment isolation signal and both suction valves open will close RHR (LPCI) admission valves.
2	Core Spray Auto Sequencing Timers (5)	$6 \leq t \leq 8$ secs.	B	1. With diesel power 2. One per motor
2	LPCI Auto Sequencing Timers (5)	$0 \leq t \leq 1$ sec.	B	1. With diesel power 2. One per motor
1	RHRSW A3, B1, C3, and D1 Timers	$13 \leq t \leq 15$ sec.	A	1. With diesel power 2. One per pump
2	Core Spray and LPCI Auto Sequencing Timers (6)	$0 \leq t \leq 1$ sec. $6 \leq t \leq 8$ sec. $12 \leq t \leq 16$ sec. $18 \leq t \leq 24$ sec.	B	1. With normal power 2. One per CSS motor 3. Two per RHR motor
1	RHRSW A3, B1, C3, and D1 Timers	$27 \leq t \leq 29$ sec.	A	1. With normal power 2. One per pump

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 177.7" (538" above vessel zero) above the top of the active fuel closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Group 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel closes the Main Steam Line Isolation Valves and Main Steam, RCIC, and HPCI Drain Valves (Group 1 and 7). Details of valve grouping and required closing times are given in Specification 3.7. These trip settings are adequate to prevent core uncovery in the case of a break in the largest line assuming the maximum closing time.

The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel (Table 3.2.B) also initiate the RCIC and HPCI, provides input to the

LPCI loop selection logic and trips the recirculation pumps. The low reactor water level instrumentation that is set to trip when reactor water level is 17.7" (378" above vessel zero) above the top of the active fuel (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus the results given above are applicable here also.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. The setting of 200°F for the main steam line tunnel detector is low enough to detect leaks of the order of 15 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.2 FSAR. An alarm, with a nominal set point of 1.5 x normal full power background, is provided also.

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

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
License No. DPR-68

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

   A. The application for amendments by Tennessee Valley Authority (the licensee) dated February 24, 1977, as supplemented by letter dated May 23, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility License No. DPR-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. 14, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 2, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 14

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

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

   57
   64
   65
   108

2. Marginal lines indicate revised area.

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level (6)	≥ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure	100 ± 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D, SW #1)	≥ 470" above vessel zero	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PS-64-56A-D)	≤ 2 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
2	Instrument Channel - High Radiation Main Steam Line Tunnel (6)	≤ 3 times normal rated full power background	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Low Pressure Main Steam Line	≥ 850 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel - High Flow Main Steam Line	≤ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Main Steam Line Tunnel High Temperature	≤ 200°F	B	1. Above trip setting initiates Main Steam Line Isolation.

ENCLOSURE 1

Amendment No. 14

Table 3.2.B

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimun No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level	≥ 470"above vessel zero.	A	1. Below trip setting initiated HPCI.
2	Instrument Channel - Reactor Low Water Level	≥ 470"above vessel zero.	A	1. Below trip setting, associated with LPCI loop selection. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #1)	≥ 378" above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #2)	≥ 378" above vessel zero.	A	1. Below trip settings in conjunction with drywell high pressure, low water level permissive, 120 sec. del timer and CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184 & 185, SW #1)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LITS-3-52 & 62, SW #1)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of of containment spray during accident condition.

Amendment No. 14

Table 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimun No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Drywell High Pressure (PS-64-58 E-H)	1≤p≤ 2 psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.
2	Instrument Channel - Drywell High Pressure (PS-64-58 A-D, SW #2)	≤ 2 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal. (15)
2	Instrument Channel - Reactor Low Water Level (LS-3-56A, B, C, D)	≥ 470" above vessel zero	A	1. Below trip setting trips recirculation pumps
2	Instrument Channel Reactor High Pressure (PS-3-204 A, B, C, D)	≤1120 psig	A	1. Above trip setting trips recirculation pumps
2	Instrument Channel - Drywell High Pressure (PS-64-58A-D, SW #1)	≤ 2 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
2(16)	Instrument Channel - Drywell High Pressure (PS-64-57A-D)	≤ 2 psig	A	1. Above trip setting in conjunction with low reactor water level, drywell high pressure, 120 sec, delay timer and CSS or RHR pump running, initiates ADS.

Amendment No. 14

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 177.7" (538" above vessel zero) above the top of the active fuel closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Group 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel closes the Main Steam Line Isolation Valves and Main Steam RCIC, and HPCI Drain Valves (Group 1 and 7). Details of valve grouping and required closing times are given in Specification 3.7. These trip settings are adequate to prevent core uncovery in the case of a break in the largest line assuming the maximum closing time.

The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel (Table 3.2.B) also initiate the RCIC

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 14 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 letters dated February 24, 1977 (Reference 1) and May 23, 1978 (Reference 2) the Tennessee Valley Authority (TVA) has requested approval for setting the Browns Ferry Unit Nos. 1, 2, and 3 (BF1, BF2, BF3) low water level setpoints at 470 inches above vessel zero. This represents a reduction of 20 inches below the current level. By reducing the low water level setpoint TVA hopes to avoid isolations due to low water level during turbine trips at high power.

The low water level setpoint, which is commonly called the L_2 setpoint, is that reactor water level below which the main steamline isolation valves close, HPCI and RCIC flows are initiated, and the recirculation pumps trip.

Lowering L_2 by 20 inches would mean that those system functions could be initiated later in time during any transient or accident involving reduction in water level. For the Browns Ferry reactors the most severe events involving water level reduction are LOCA, steamline breaks, feedwater pump trip, loss of offsite or auxiliary power, MSIV closure, turbine trip, load rejection, and pressure regulator failure. The worst case LOCA and the most severe of the anticipated transients, the loss of feedwater flow, have been analyzed to determine the effect of the proposed reduction in L_2 on plant safety (References 1 and 2). Our evaluation of these analyses is presented in the following discussion.

2.0 Discussion

2.1 ECCS Performance With the L_2 Setpoint at 470 Inches

To justify that the Browns Ferry ECCS performance will remain acceptable with the new L_2 setpoint, TVA has calculated and provided the maximum changes in LOCA peak clad temperature (PCT) expected to result from the setpoint reduction.

A wide spectrum of break sizes and break locations has been analyzed using approved calculational methods and input. For the large breaks analyzed, reduction of the L_2 setpoint resulted in increases in PCT which were in each case less than 20°F, and for the small breaks the largest increase in PCT was 15°F (Reference 2).

Peak clad temperatures associated with the worst small breaks (less than 1 ft^2) are below the large break values by much more than 15°F (Reference 3 for BF3 and Reference 6 for BF1 and BF2). This means that reduction of L_2, which would involve an increase in small break PCT no more than 15°F, could not cause any small break LOCA to become the worst case.

For each of the three plants the worst break with the current L_2 setpoint is a DBA size break in the suction side of the recirculation line, and the worst single failure is the failure of the LPCI injection valve. In Reference 7, TVA described the extent and result of the large break analyses which were performed to evaluate the possibility that the proposed change in L_2 might affect the nature of the worst break. Based on our review of that information, we concluded that sufficient analyses have been completed to demonstrate that reduction in L_2 by 20 inches would not change the size or location of the worst large break, and that the worst single failure would also remain the same. This conclusion is valid for BF1 and BF2 which are LPCI modified plants and for BF3 which has loop selection logic.

The staff has recently completed a reevaluation (after correction to errors in ECCS model and data) of peak clad temperatures for Browns Ferry Units Nos. 1 and 2 as part of our evaluation of the initial core refuelings (reloads) for these facilities (references 4 and 5). For BF1 and BF2, the maximum PCT is 2151°F.

For BF3 the PCT has been calculated to be 2030°F (Reference 3). Although errors have been identified in both the input data and ECCS evaluation model on which the BF3 PCT is based (Reference 10), we have concluded that these errors have opposite effects on the calculated PCT and that the corrected value would not be significantly above 2030°F. Since the March 10, 1977 Orders to all licensees with BWR facilities, we have completed our evaluation of 16 revised ECCS analyses submitted in response to those Orders. In all but one case, the revised PCT was decreased as a result of the correction in the ECCS model errors. (In the one case, the PCT increased by 3°F) Based on this information, we conclude that when the ECCS analysis for Browns Ferry Unit 3 is revised to account for the model errors, the PCT will probably decrease but certainly will not increase by more than 20°F. Thus, at the very worst, we can conclude that the present PCT for BF3 is no more than 2050°F for the worst break condition.

We conclude that addition of 20°F to the current PCT values adequately represents the effect of the proposed L_2 setpoint reductions. Therefore, the resulting PCT for each of the three Browns Ferry reactors would remain below the 2200°F safety limit. On this basis, we conclude that the proposed reduction in L_2 is acceptable in terms of its possible effect on ECCS performance.

2.2 Effect of Reduction in L_2 on Results of Anticipated Transients

MCPR reductions or LHGR increases during anticipated transients are affected by the L_2 setpoint only through the recirculation pump trip which would occur on low water signal during turbine trips or load rejection transients. However, the analyses performed to determine the MCPR reduction due to these events do not take credit (and not taking credit is conservative) for the recirculation pump trip (References 8 and 9). It has not been necessary, therefore, to determine the effect of the proposed L_2 reduction on MCPR or LHGR limits.

Because the MSIV closure-flux scram event, which demonstrated compliance with the ASME Code requirements on peak vessel pressure, does not take credit for any L_2 trips, this event has not been re-analyzed.

We have considered the possibility that the reduction in L_2 could involve an increase in the release of fission products associated with a break in the steamline outside containment. However, the isolation signal caused by water level below the L_2 setpoint would be preceeded in time by either two or three other independent isolation signals, depending on the break size. Even if a break too small to result in MSIV flows above the high MSIV flow isolation setpoint should occur, isolation would be initiated by either high temperature or high radiation levels in the steamline tunnel before the water level drops to the L_2 setpoint. On this basis we conclude that the potential consequences of postulated steamline breaks will not increase due to the reduction in L_2.

To provide assurance that no anticipated transient would result in uncovery of the top of the active fuel; those transients involving reduction in reactor water inventory have been reviewed. Of the transients of this type mentioned in Section 1 of this report, information in Section 14.54 of the FSAR shows that the most severe is the loss of feedwater flow due to a feedwater pump trip. The feedwater pump trip has been evaluated with the proposed lower L_2 setpoint. The minimum water level which would be reached should such a transient occur would be 60 inches above the top of the active fuel (Reference 2). This represents only a 10 inch decrease from the minimum water level without the change in L_2.

On the basis that MCPR reduction, LHGR, and MSIV-closure-Flux Scram are not sensitive to the L_2 setpoint, we conclude that new analyses of MCPR, LHGR or pressure limits are not necessary. We have also concluded that the consequences of steam line breaks will not increase. Furthermore, on the basis of the evaluation provided by TVA of the most severe water level reduction transient, showing that a feedwater pump trip would not result in uncovery of the top of the active fuel, we conclude that the margin between the minimum water level and the top of the active fuel with the proposed L_2 setpoint is acceptable. These conclusions are valid for all three of the Browns Ferry units which are identical with regard to the transients of interest.

3.0 Evaluation

Based on our review of TVA's analyses of the worst case LOCA and the most severe anticipated transient assuming the proposed L_2 set point, we conclude that the reduction of the low water level set point by 20 inches is acceptable. The change will not involve a significant decrease in safety margins or a significant increase in the probability or consequences of any accident or transient.

4.0 Environmental Considerations

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

5.0 Conclusion

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

Dated: August 2, 1978

References

1. Letter from H. G. Parris of TVA to B. C. Rusche of NRC, February 24, 1977.

2. Letter from J. E. Gilliland of TVA to G. Lear of NRC, May 23, 1978.

3. Supplement No. 8 to the Safety Evaluation Report by the Division of Project Management, Office of Nuclear Reactor Regulation, USNRC, in the Matter of TVA Browns Ferry Nuclear Plant, Units 1, 2 and 3, Docket Nos. 50-259, 50-260 and 50-296.

4. Amendment No. 35 to Facility License No. DPR-33 dated January 10, 1978 approving operation of Browns Ferry Unit No. 1 in cycle 2 and terminating the Commission's Order for Modification of License dated March 11, 1977 based on the acceptability of a revised ECCS analysis.

5. Amendment No. 35 to Facility License No. DPR-52 dated June 21, 1978 approving operation of Browns Ferry Unit No. 2 in cycle 2 and terminating the Order for Modification of License dated March 11, 1977 relating to ECCS reevaluation.

6. Safety Evaluation Report Approving Operation of Browns Ferry Units 1 and 2 with Four of the Six ADS Valves Operable, May 1978.

7. Letter from J. E. Gilliland of TVA to T. A. Ippolito of NRC, July 19, 1978.

8. General Electric "Generic Reload Fuel Application", NEDE 24011P, May 1977.

9. Safety Evaluation for the General Electric Topical Report, "Generic Reload Fuel Application" (NEDE-24011-P), April 1978.

10. Letter, A. Schwencer, NRC, to Godwin Williams, Jr., TVA dated March 11, 1977 transmitting "Order for Modification of License No DPR-68" for Browns Ferry Nuclear Plant, Unit No. 3.

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 40 to Facility Operating License No. DPR-33, Amendment No. 38 to Facility Operating License No. DPR-52 and Amendment No. 14 to Facility Operating License No. DPR-68 issued to Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3, located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

The amendments change the Technical Specifications to lower the reactor low water level setpoint from 490 inches to 470 inches.

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

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated February 24, 1977, as supplemented by letter dated May 23, 1978, (2) Amendment No. 40 to License No. DPR-33, Amendment No. 38 to License No. DPR-52, and Amendment No. 14 to License No. DPR-68, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 2 day of August 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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