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

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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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

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. 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					
2	High Flux	≤ 15% rated power				X	1.A o
2	Inoperative	(13)		X(21)	X(17)	(15)	1.A o
2	Downscale	≥ 3 Indicated on Scale		X(21)	X(17)	X	1.A o
				(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			Action(1)
			Refuel(7)	Startup/Hot Standby	Run	
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 tc e operable when the reacto 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% ($\leq$ 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 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)	≤ 0.66W + 42% (2)
2(1)	APRM Upscale (Startup Mode) (8)	≤ 12%
2(1)	APRM Downscale (9)	≥ 3%
2(1)	APRM Inoperative	(10_b)
1(7)	RBM Upscale (Flow Bias)	≤ 0.66W + 41% (2) for two recirculation loop operation ≤ 0.66W + 37.7%(2) for one recirculation loop operation
1(7)	RBM Downscale (9)	≥ 3%
1(7)	RBM Inoperative	(10_c)
3(1)	IRM Upscale (8)	≤ 108/125 of full scale
3(1)	IRM Downscale (3)(8)	≥ 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)	≤ 1×10^5 counts/sec.
2(1)(6)	SRM Downscale (4)(8)	≥ 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	≤ 10% difference in recirculation flows
2(1)	Flow Bias Upscale	≤ 110% recirculation flow
1(1)	Rod Block Logic	N/A
2(1)	RSCS Restraint (PS-35-61A & PS-35-61B)	147 psig turbine first stage pressure (approximately 30% power)

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM, IRM, and APRM (Startup mode), blocks need not be operable in "Run" mode, and the APRM (Flow biased) and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition last longer than seven days, the system with the inoperable channel shall be tripped. If the first column cannot be met for both trip systems, both trip systems shall be tripped.
2. W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (3293 MWt). A ratio of FRP/CMFLPD <1.0 is permitted at reduced power. See Specification 2.1 for APRM control rod block setpoint.
3. IRM downscale is bypassed when it is on its lowest range.
4. This function is bypassed when the count rate is $\geq$ 100 cps and IRM above range 2.
5. One instrument channel; i.e., one APRM or IRM or RBM, per trip system may be bypassed except only one of four SRM may be bypassed.
6. IRM channels A, E, C, G all in range 8 bypasses SRM channels A & C functions.

   IRM channels B, F, D, H all in range 8 bypasses SRM channels B & D functions.
7. The following operational restraints apply to the RBM only:
   a. Both RBM channels are bypassed when reactor power is $\leq$ 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. 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 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	

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	Sore 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

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^o)	Recirculation	564			X	
SS3-A(335^o)	Recirculation	564			X	
SS3-B(115^o)	Recirculation	564			X	
SS3-B(154^o)	Recirculation	564			X	
SS4-A	Recirculation	570			X	
SS4-B	Recirculation	570			X	
SS5-A(262^o)	Recirculation	581			X	
SS5-B(325^o)	Recirculation	581			X	
SS5-B(35^o)	Recirculation	581			X	
SS5-B(98^o)	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

3.9.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.

SURVEILLANCE REQUIREMENTS

4.9.A Auxiliary Electrical Equipment

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

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 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.

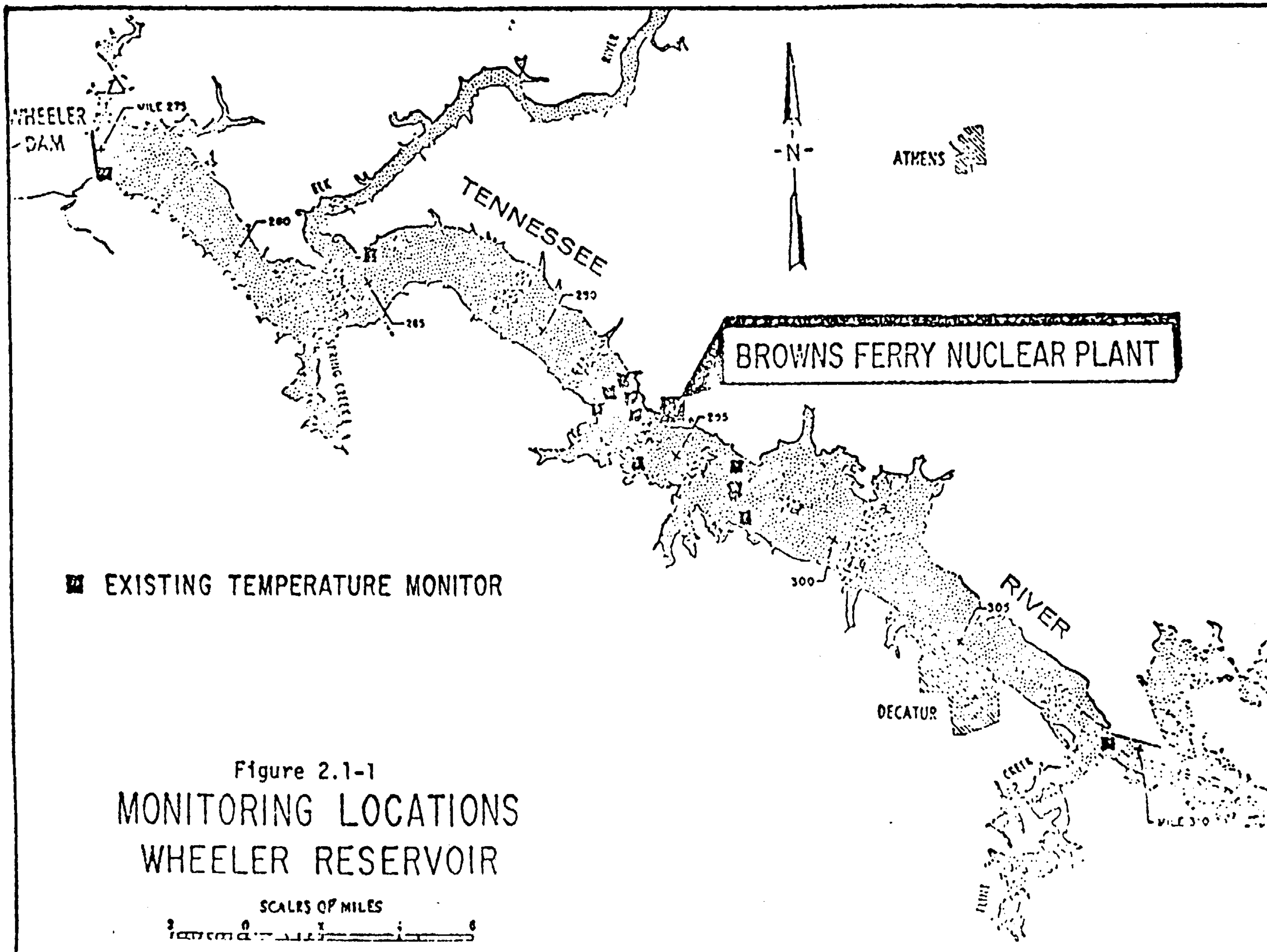
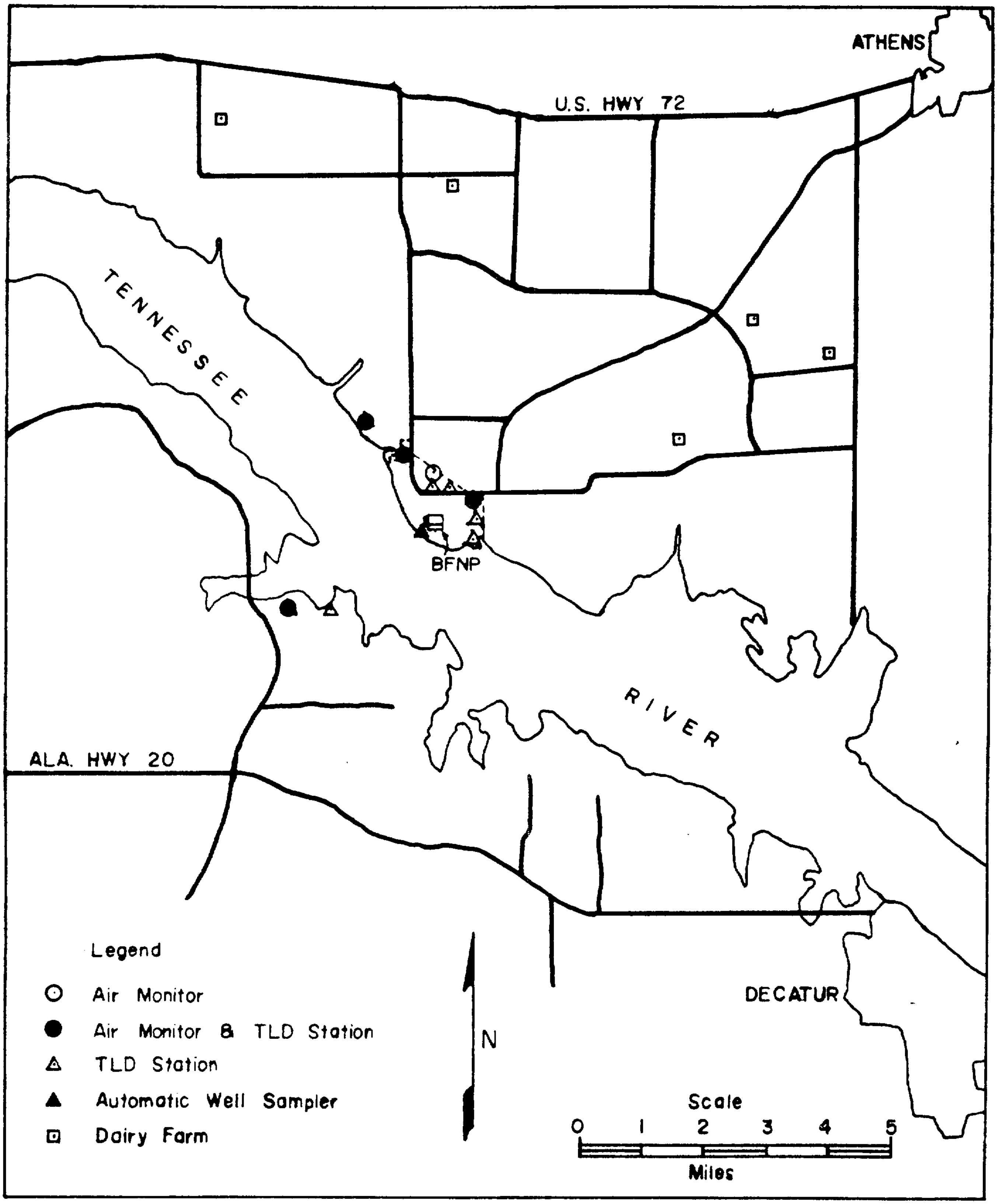


Figure 2.1-1
MONITORING LOCATIONS
WHEELER RESERVOIR

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: Shut-down	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	$\leq$ 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	$\leq$ 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	$\geq$ 3 Indicated on Scale		(11)	(11)	X(12)	1.A or 1.B
2	High Reactor Pressure	$\leq$ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14)	$\leq$ 2 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14)	$\geq$ 538" above vessel zero		X	X	X	1.A
2	High Water Level in Scram Discharge Tank	$\leq$ 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			Action(1)
			Refuel(7)	Startup/Hot Standby	Run	
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.

LIMITING CONDITIONS FOR OPERATION

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.

SURVEILLANCE REQUIREMENTS

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.

LIMITING CONDITIONS FOR OPERATION

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.

SURVEILLANCE REQUIREMENTS

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.

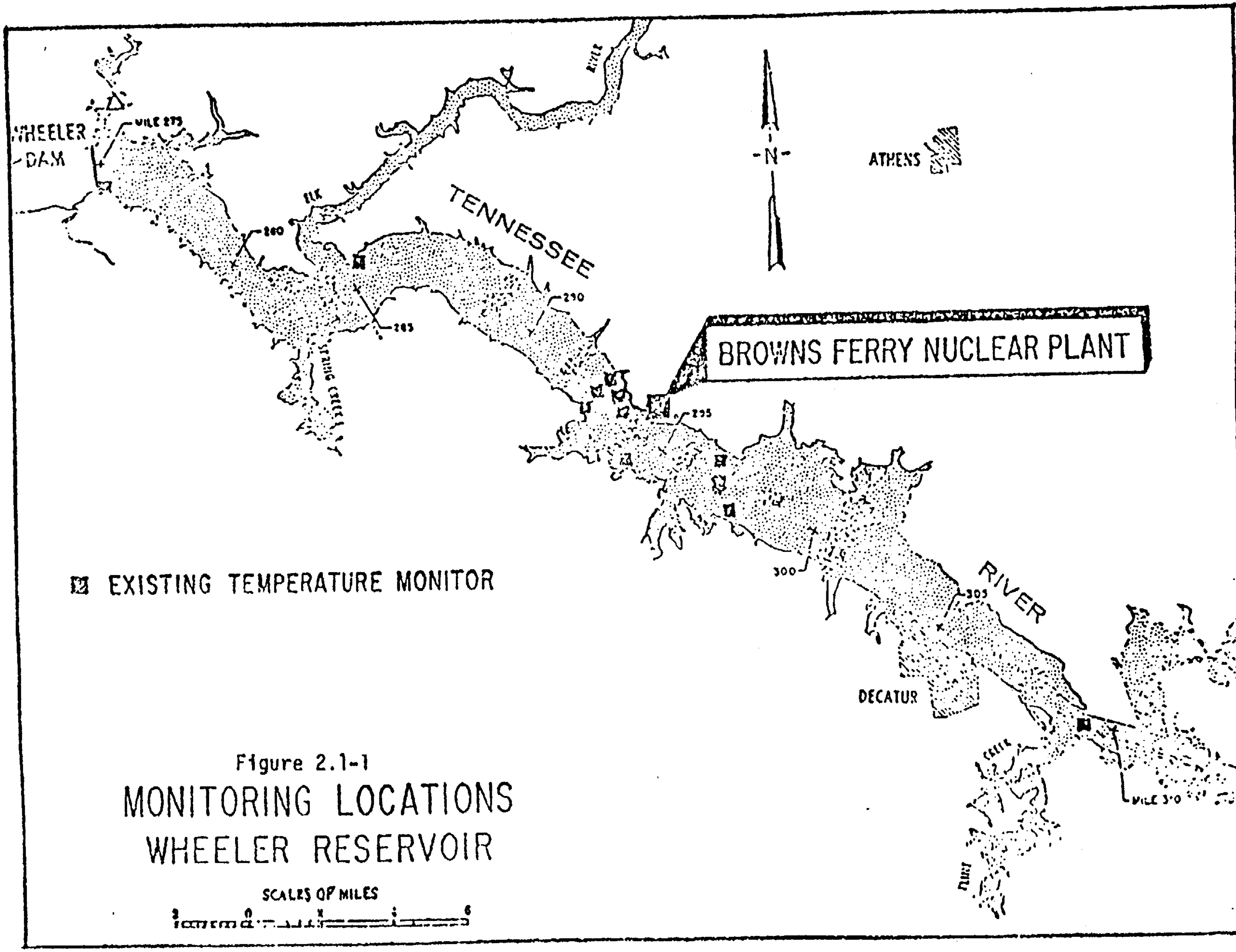
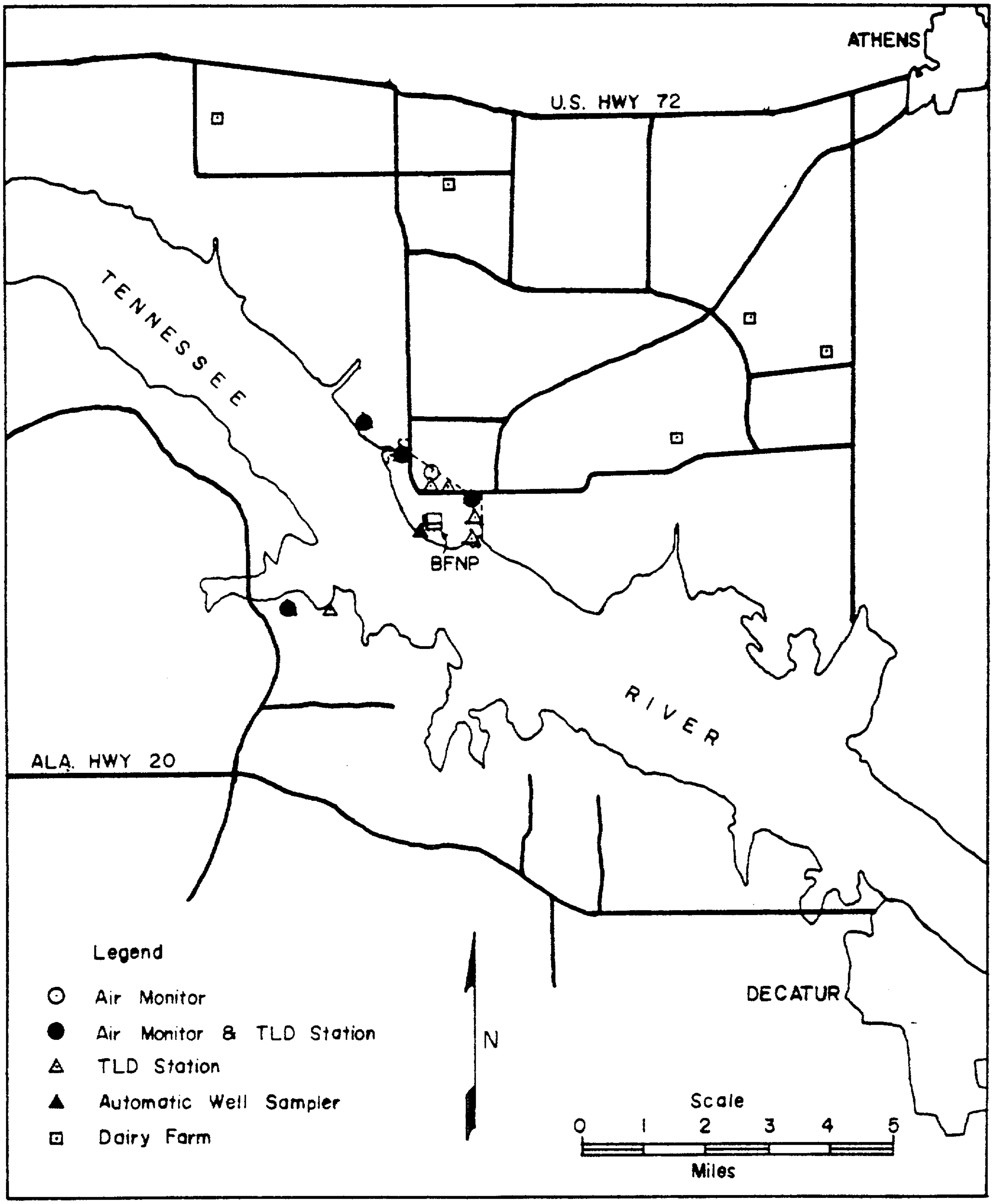


Figure 2.1-1
MONITORING LOCATIONS
WHEELER RESERVOIR

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 downscal 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).

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	

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
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 Systems

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 mouths, 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.

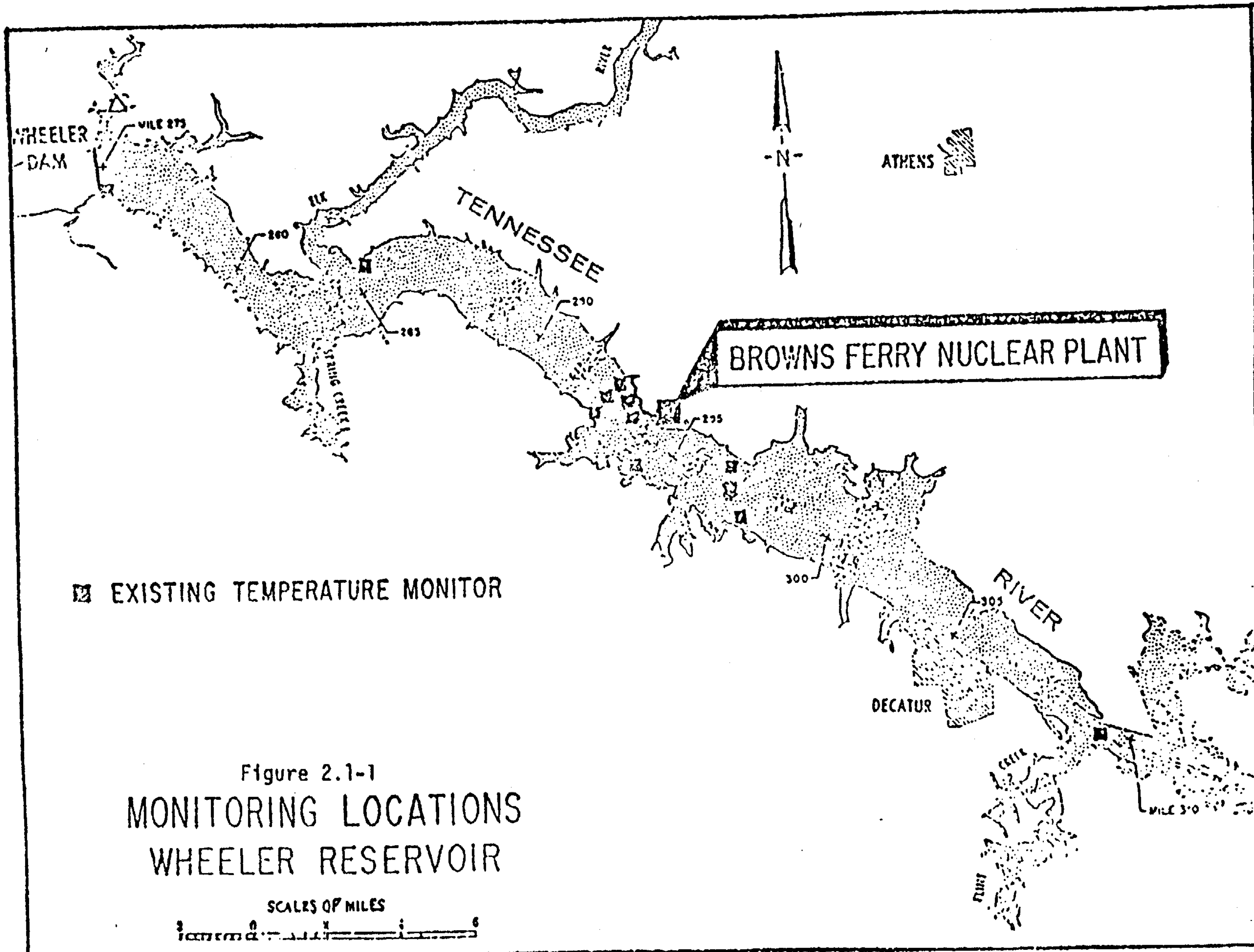
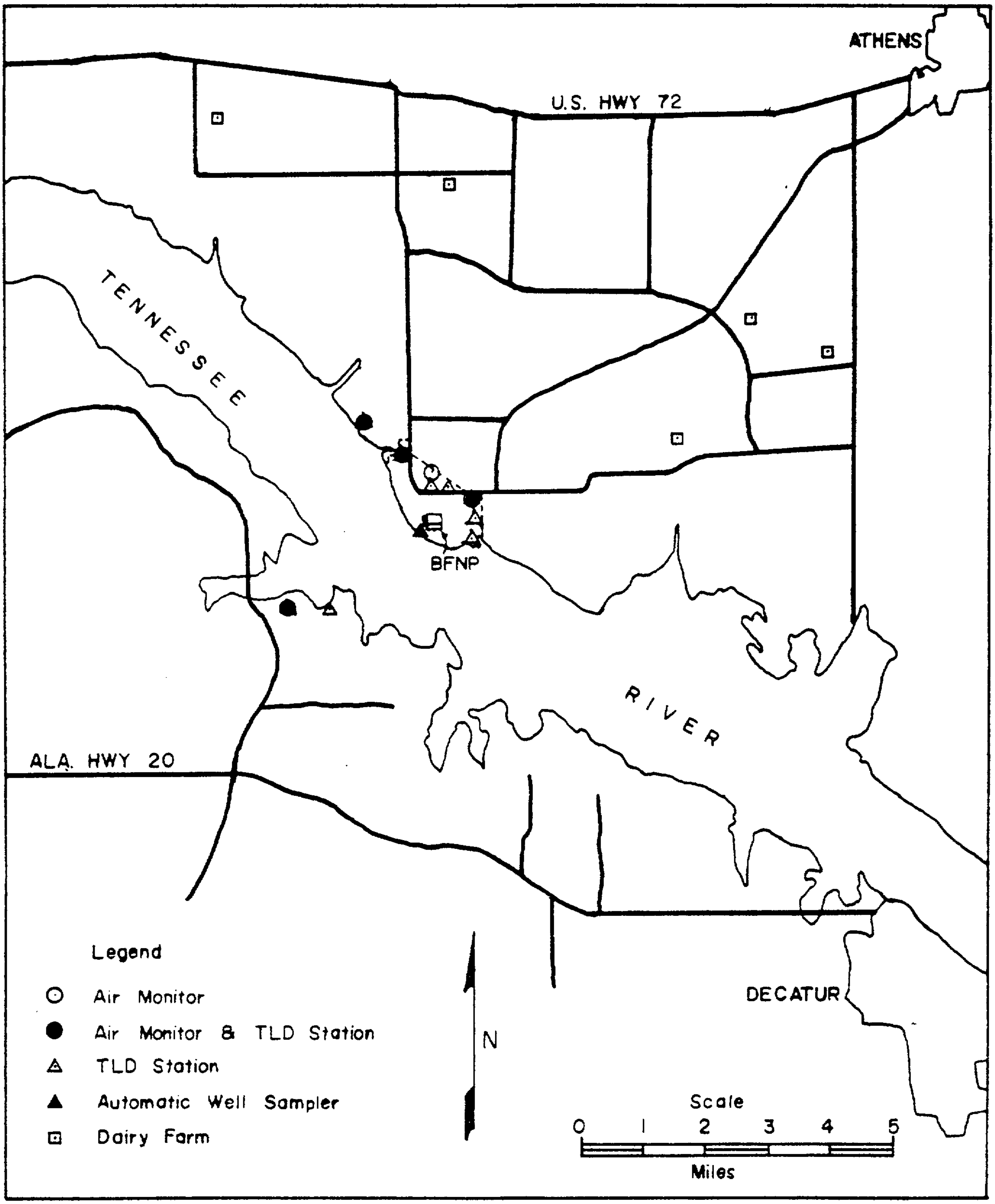


Figure 2.1-1
MONITORING LOCATIONS
WHEELER RESERVOIR

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

Tom 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	Modes in Which Function Must Be Operable: Shut-down	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
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 $\geq$ 100 cps and IRM above range 2.

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

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

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

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

   a. Both RBM channels are bypassed when reactor power is $\leq$ 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. Scram Insertion Times

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.

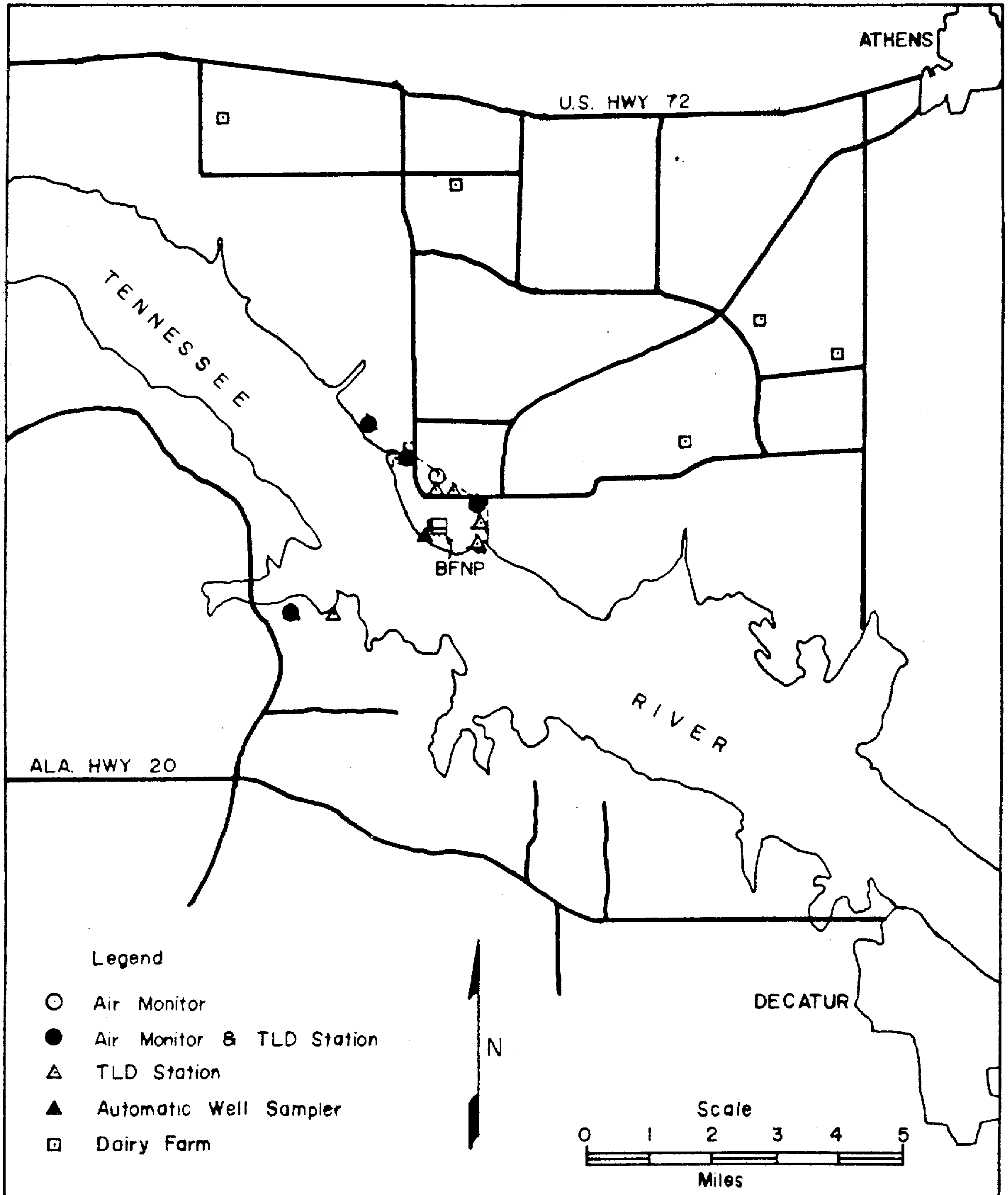
4.10 CORE ALTERATIONS

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 simultaneously 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 the 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