TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

500C Chestnut Street Tower II

TVA BFNP TS 123

JUN 29 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Denton:

In the Matter of the	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

In accordance with the provisions of 10 CFR Part 50.59, we are enclosing 40 copies of a requested amendment to licenses DPR-33, DPR-52, and DPR-68 to change the technical specifications of Browns Ferry Nuclear Plant units 1, 2, and 3 (Enclosure 1). Also enclosed are justifications for the proposed changes (Enclosure 2). The enclosed changes consist of administrative changes, clarification changes, NRC requested changes, and changes to reflect new plant organization.

In accordance with the requirements of 10 CFR Part 170.22, we have determined these proposed amendments to be Class III for unit 1 and Class I for units 2 and 3. These classifications are based on the facts that the proposed amendment involves an issue which does not involve a significant hazard consideration for unit 1, and the proposed amendments for units 2 and 3 are duplicates of the unit 1 proposed amendment submitted by this letter. The remittance of $4800 ($4000 for unit 1, $800 for units 2 and 3) is being wired to the NRC, Attention: Licensing Fee Management Branch.

Very truly yours,

J E Gilleland

J. E. Gilleland
Assistant Manager of Power

Subscribed and sworn to before
me this 29 day of June 1979.

Lisa Bradbury
Notary Public

My Commission Expires 10/4/81

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Enclosures
cc: See page 2

7907060188

Mr. Harold R. Denton JUN 29 1979

cc (Enclosures):
Mr. Charles R. Christopher
Chairman, Limestone County Commission
P.O Box 188
Athens, Alabama 35611

Dr. Ira L. Myers
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36104

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ENCLOSURE 1

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PROPOSED CHANGES TO BROWNS FERRY

UNIT 1 TECHNICAL SPECIFICATIONS

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TABLE 3.1.A (Continued)

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

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

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

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

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

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

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

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

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

18. Operability is required when normal first-stage pressure is below 30% ($\leq$ 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 3.0 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.

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TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup not to exceed once per week
Inoperative	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup not to exceed once per week
APRM			
High Flux (15% scram)	C	Trip Output Relays (4)	Before Each Startup and Weekl When Required to be Operable
High Flux	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	(6)	(6)
High Reactor Pressure	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water Level (5)	A	Trip Channel and Alarm	Once/Month (1)
High Water Level in Scram Discharge Tank	A	Trip Channel and Alarm	Every 3 Months
Turbine Condenser Low Vacuum	A	Trip Channel and Alarm	Once/Month (1)
Main Steam Line High Radiation	B	Trip Channel and Alarm (4)	Once/Week

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

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

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

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

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

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

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TABLE 3.2.E

INSTRUMENTATION THAT MONITORS LEAKAGE INTO DRYWELL

System (2)	Setpoints	Action	Remarks
Equipment Drain		(1)	1. Used to determine identifiable reactor coolant leakage. 2. Considered part of sump system.
Flow Integrator	N/A		
Sump Fill Rate Timer	≥20.1 min.		
Sump Pump Out Rate Timer	≤13.4 min.		
Floor Drain		(1)	1. Used to determine unidentifiable reactor coolant leakage. 2. Considered part of sump system.
Flow Integrator	N/A		
Sump Fill Rate Timer	≥80.4 min.		
Sump Pump Out Rate Timer	≤8.9 min.		
Drywell Air Sampling	Gas and Particulate: 3 x Average Background	(3)	

NOTES:

(1) Whenever a system is required to be operable, there shall be one operable system either automatic or manual, or the action required in Section 3.6.C.2 shall be taken.

(2) An alternate system to determine the leakage flow is a manual system whereby the time between sump pump starts is monitored. The time interval will determine the leakage flow because the volume of the sump will be known.

(3) Upon receipt of alarm, immediate action will be taken to confirm the alarm and assess the possibility of increased leakage. Refer to Tech Spec Sections 3.6.C.2 and 3.6.C.3 for drywell air sampling system out of service.

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C. 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.

4.7.C Secondary Containment

D. 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.

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 except where specific written relief from ASME Section XI requirements has been granted by NRC pursuant to 10 CFR 50, Section 50.55a(g)(b)(i).

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3.7 CONTAINMENT SYSTEMS

G. Containment Atmosphere Dilution System (CAD)

1. The Containment Atmosphere Dilution (CAD) System shall be operable with:

   a. Two independent systems capable of supplying nitrogen to the drywell and torus.

   b. A minimum supply of 2500 gallons of liquid nitrogen per system.

2. The Containment Atmosphere Dilution (CAD) System shall be operable whenever the reactor mode switch is in the "RUN" position.

3. If one system is inoperable, the reactor may remain in operation for a period of 30 days provided all active components in the other system are operable.

4. If Specification 3.7.G.1 and 3.7.G.2, or 3.7.G.3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

5. Primary containment pressure shall be limited to a maximum of 30 psig during repressurization following a loss of coolant accident.

4.7 CONTAINMENT SYSTEMS

G. Containment Atmosphere Dilution System (CAD)

1. System Operability

   a. At least once per month, cycle each solenoid operated air/nitrogen valve with its hand switch (no containment isolation override) through at least one complete cycle of full travel and verify that each manual valve in the flow path is open.

   b. Verify that the CAD System contains a minimum supply of 2500 gals. of liquid nitrogen twice per week.

   c. At each cold shutdown verify that each solenoid operated air/nitrogen valve in the flow path can be opened with a containment isolation signal present (containment isolation override).

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POOR ORIGINAL

TABLE 3.7.A
PRIMARY CONTAINMENT ISOLATION VALVES

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
1	Main steamline isolation valves (FCV-1-14,26,37,&51;1-15, 27, 38, & 52)	4	4	$3 < T < 5$	0	GC
1	Main steamline drain isolation valves FCV-1-55 & 1-56	1	1	15	C	SC
1	Reactor Water sample line isolation valves	1	1	5	C	SC
2	RHRS shutdown cooling supply isolation valves FCV-74-48 & 47	1	1	40	C	SC
2	RHRS - LPCI to reactor FCV-74-53, 67		2	30	C	SC
2	Reactor vessel head spray isolation valves FCV-74-77, 78	1	1	30	C	SC
2	RHRS flush and drain vent to suppression chamber FCV-74-102, 103, 119, & 120		4	20	C	SC
2	Suppression Chamber Drain FCV-75-57, 58		2	15	C	SC
2	Drywell equipment drain discharge isolation valves FCV-77-15A, & 15B		2	15	0	GC
	Drywell floor drain discharge isolation valves FCV-77-2A & 2B		2	15	0	GC

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TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
3	Reactor water cleanup system supply isolation valves FCV-69-1, & 2	1	1	30	0	GC
4	HPCIS steamline isolation valves FCV-73-2 & 3	1	1	20	0	GC
5	RCICS steamline isolation valves FCV-71-2 & 3	1	1	15	0	GC
6	Drywell nitrogen purge inlet isolation valves (FCV-76-18)		1	10	C	SC
6	Suppression chamber nitrogen purge inlet isolation valves (FCV-76-19)		1	10	C	SC
6	Drywell Main Exhaust isolation valves (FCV-64-29 and 30)		2	90	C	SC
6	Suppression chamber main exhaust isolation valves (FCV-64-32 and 33)		2	90	C	SC
6	Drywell/Suppression Chamber purge inlet (FCV-64-17)		1	90	C	SC
6	Drywell Atmosphere purge inlet (FCV-64-18)		1	90	C	SC

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TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
6	Suppression Chamber purge inlet (FCV-64-19)		1	100	C	SC
6	Drywell/Suppression Chamber nitrogen purge inlet (FCV-76-17)		1	10	C	SC
6	Drywell Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-31)		1	10	C	SC
6	Suppression Chamber Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-34)		1	10	C	SC
7	RCIC Steamline Drain (FCV-71-6A, 6B)		2	5	0	GC
7	RCIC Condensate Pump Drain (FCV-71-7A, 7B)		2	5	C	SC
7	HPCI Hotwell pump discharge isolation valves (FCV-73-17A, 17B)		2	5	C	SC
7	HPCI steamline drain (FCV-73-6A, 6B)		2	5	0	GC
8	TIP Guide Tubes (5)		1 per guide tube	NA	C	GC

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 bypassed 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.

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POOR ORIGINAL

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

Browns Ferry unit *1* is located at Browns Ferry Nuclear Plant site on property owned by the United States and in custody of the TVA. The site shall consist of approximately 840 acres on the north shore of Wheeler Lake at Tennessee River Mile 294 in Limestone County, Alabama. The minimum distance from the outside of the secondary containment building to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 4,000 feet.

5.2 REACTOR

A. The reactor core may contain 764 fuel assemblies consisting of 7x7 assemblies having 49 fuel rods each, 8x8 assemblies having 63 fuel rods each, and 8x8 R assemblies having 62 fuel rods each. The number of each type in the core is given in the most recent reload amendment topical report.

B. The reactor core shall contain 185 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70 percent of theoretical density.

5.3 REACTOR VESSEL

The reactor vessel shall be as described in Table 4.2-2 of the FSAR. The applicable design codes shall be as described in Table 4.2-1 of the FSAR.

5.4 CONTAINMENT

A. The principal design parameters for the primary containment shall be as given in Table 5.2-1 of the FSAR. The applicable design codes shall be as described in Section 5.2 of the FSAR.

B. The secondary containment shall be as described in Section 5.3 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with the standards set forth in Section 5.2.3.4 of the FSAR.

5.5 FUEL STORAGE

A. The arrangement of fuel in the new-fuel storage facility shall be such that k_{eff}, for dry conditions, is less than 0.90 and flooded is less than 0.95 (Section 10.2 of FSAR).

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A summary of the more significant discussions and conclusions of the NSRB will be transmitted along with the final minutes.

7. Charter

A written charter delineating the establishment, composition, and mission of the NSRB and the dissemination of NSRB minutes and reports shall be maintained; this may be amended as required. The charter shall identify the responsibility and authority of the NSRB in conducting reviews, including responsibility to identify problems and to recommend solutions to the Manager of Power.

B. Plant Operations Review Committee (PORC)

1. Membership

The PORC shall consist of the plant superintendent, electrical maintenance supervisor, mechanical máintenance supervisor, instrument maintenance supervisor, health physics supervisor, operations supervisor, results supervisor, and QA staff supervisor. An assistant plant supervisor may serve as an alternate committee member when his supervisor is absent.

The plant superintendent will serve as chairman of the PORC. The assistant plant superintendent will serve as chairman in the absence of the plant superintendent.

2. Meeting Frequency

The PORC shall meet at regular monthly intervals and for special meetings as called by the chairman or as requested by individual members.

3. Quorum

Superintendent or assistant superintendent, plus five of the seven other members, or their alternate, will constitute a quorum. A member will be considered present if he is in telephone communication with the committee.

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POOR ORIGINAL

plant, the applicable codes required fatigue usage evaluation for the reactor pressure vessel only. The locations to be monitored shall be:

1. The feedwater nozzles

2. The shell at or near the waterline

3. The flange studs

b. Recording, Evaluating, and Reporting

(1) Transients that occur during plant operations will be reviewed and a cumulative fatigue usage factor determined.

(2) For transients which are more severe than the transients evaluated in the stress report, code fatigue usage calculations will be made and tabulated separately.

(3) In the **monthly** Operating Report, the fatigue usage factor determined for the transients defined in (1) and (2) above shall be added and a cumulative fatigue usage factor to date shall be listed. When the cumulative usage factor reaches a value of 1.0, an inservice inspection shall be included for the specific location at the next scheduled inspection (3-1/3-year interval) period and 3-1/3-year intervals thereafter, and a subsequent evaluation performed in accordance with the rules of ASME Section XI Code if any flaw indications are detected. The results of the evaluation shall be submitted in a Special Report (Section 6.7.3) for review by the Commission.

B. Except where covered by applicable regulations, items 1 through 8 above shall be retained for a period of at least 5 years and items 9 through 17 shall be retained for the life of the plant. A complete inventory of radioactive materials in possession shall be maintained current at all times.

(b). Annual Report

(1) A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, 3.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

(2) A report of facility changes, tests or experiments required pursuant to 10CFR50.59(b).

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate Regional Office, to be submitted no later than the tenth of each month following the calendar month covered by the report. A narrative summary of operating experience shall be submitted in the above schedule.

2. Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC, Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

B. Source Tests

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

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

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

a.	Secondary Containment Leak Rate Testing(5)	4.7.C	Within 90 days of completion of each test.
b.	Fatigue Usage Evaluation	6.6	monthly Operating Report
c.	Seismic Instrumentation Inoperability	3.2.J.3	Within 10 days after 30 days of inoperability

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FOOTNOTES

1. A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

2. The term "forced reduction in power" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the load on the unit be reduced for corrective action immediately or up to and including the very next weekend. Note that routine preventive maintenance, surveillance, and calibration activities requiring power reductions are not covered by this section.

3. The term "forced outage" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the unit be removed from service for corrective action immediately or up to and including the very next weekend.

4. This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

5. Each integrated leak rate test of the secondary containment shall be the subject of a summary technical report. This report should include data on the wind speed, wind direction, concurrent reactor building pressure, and emergency ventilation flow rate. The report shall also include analyses and interpretations of those data which demonstrate compliance with the specified leak rate limits.

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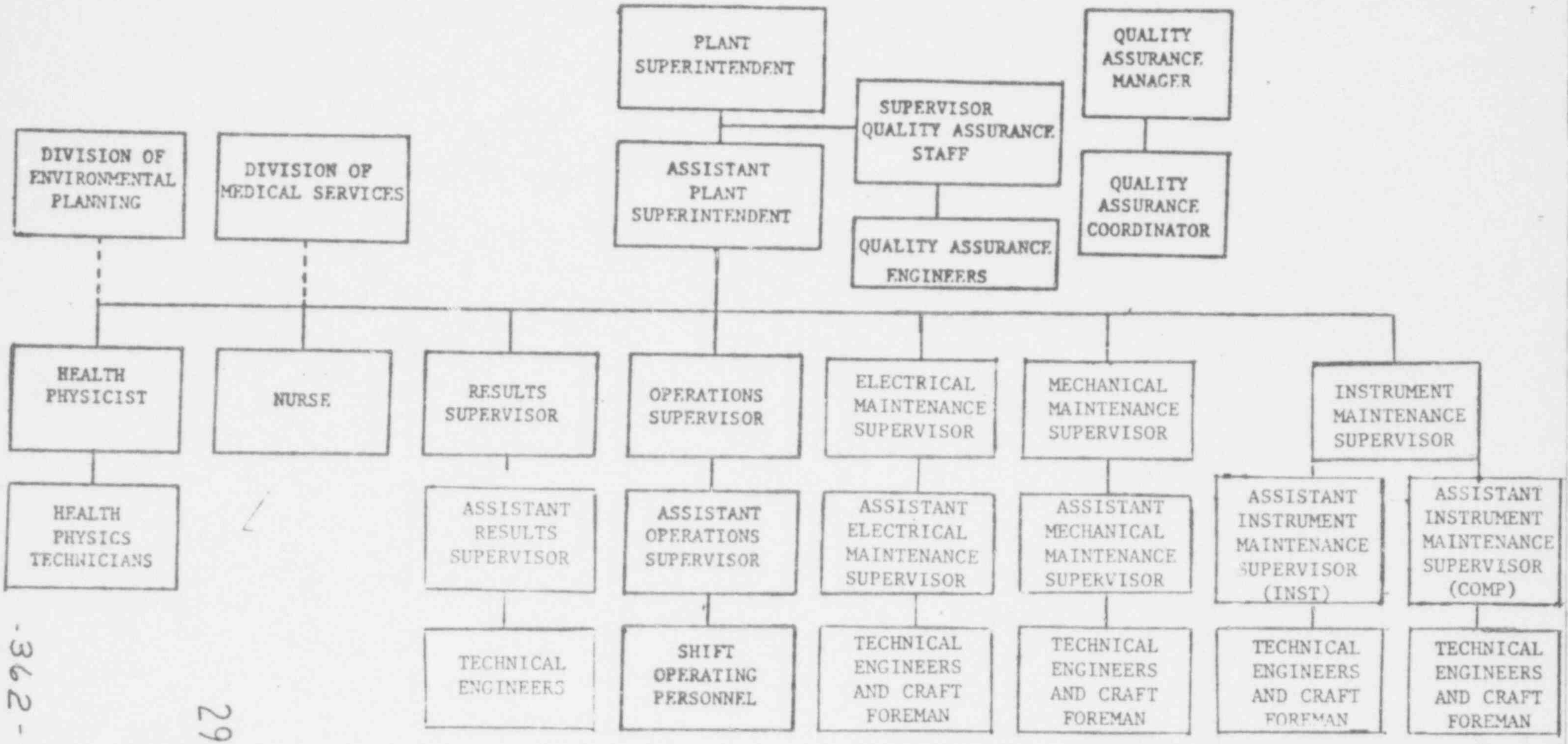

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FUNCTIONAL ORGANIZATION
FIGURE 6.1-2

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PROPOSED CHANGES TO BROWNS FERRY

UNIT 2 TECHNICAL SPECIFICATIONS

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TABLE 3.1.A (Continued)

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

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POOR ORIGINAL

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

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

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

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

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

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

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

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

18. Operability is required when normal first-stage pressure is below 30% ($\leq$ 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 3.0 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.

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TABLE 4.1.A

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup not to exceed once per week
Inoperative	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup not to exceed once per week
APRM			
High Flux (15% scram)	C	Trip Output Relays (4)	Before Each Startup and Weekl When Required to be Operable
High Flux	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	(6)	(6)
High Reactor Pressure	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water Level (5)	A	Trip Channel and Alarm	Once/Month (1)
High Water Level in Scram Discharge Tank	A	Trip Channel and Alarm	Every 3 Months
Turbine Condenser Low Vacuum	A	Trip Channel and Alarm	Once/Month (1)
Main Steam Line High Radiation	B	Trip Channel and Alarm (4)	Once/Week

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

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

298 082

TABLE 3.2.E
INSTRUMENTATION THAT MONITORS LEAKAGE INTO DRYWELL

System (2)	Setpoints		Action	Remarks
Equipment Drain			(1)	1. Used to determine identifiable reactor coolant leakage.
Flow Integrator	N/A			2. Considered part of sump system.
Sump Fill Rate Timer	$\geq$20.1 min.			
Sump Pump Out Rate Timer	$\leq$13.4 min.			
Floor Drain			(1)	1. Used to determine unidentifiable reactor coolant leakage.
Flow Integrator	N/A			2. Considered part of sump system.
Sump Fill Rate Timer	$\geq$80.4 min.			
Sump Pump Out Rate Timer	$\leq$8.9 min.			
Drywell Air Sampling	Gas and Particulate	3 x Average Background	(3)	

NOTES:

(1) Whenever a system is required to be operable, there shall be one operable system either automatic or manual, or the action required in Section 3.6.C.2 shall be taken.

(2) An alternate system to determine the leakage flow is a manual system whereby the time between sump pump starts is monitored. The time interval will determine the leakage flow because the volume of the sump will be known.

(3) Upon receipt of alarm, immediate action will be taken to confirm the alarm and assess the possibility of increased leakage. Refer to Tech Spec Sections 3.6.C.2 and 3.6.C.3 for drywell air sampling system out of service.

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 except where specific written relief from ASME Section XI requirements has been granted by NRC pursuant to 10 CFR 50, Section 50.55a(g)(b)(i).

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3.7 CONTAINMENT SYSTEMS

G. Containment Atmosphere Dilution System (CAD)

1. The Containment Atmosphere Dilution (CAD) System shall be operable with:

   a. Two independent systems capable of supplying nitrogen to the drywell and torus.

   b. A minimum supply of 2500 gallons of liquid nitrogen per system.

2. The Containment Atmosphere Dilution (CAD) System shall be operable whenever the reactor mode switch is in the "RUN" position.

3. If one system is inoperable, the reactor may remain in operation for a period of 30 days provided all active components in the other system are operable.

4. If Specification 3.7.G.1 and 3.7.G.2, or 3.7.G.3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

5. Primary containment pressure shall be limited to a maximum of 30 psig during repressurization following a loss of coolant accident.

4.7 CONTAINMENT SYSTEMS

G. Containment Atmosphere Dilution System (CAD)

1. System Operability

   a. At least once per month, cycle each solenoid operated air/nitrogen valve with its hand switch (no containment isolation override) through at least one complete cycle of full travel and verify that each manual valve in the flow path is open.

   b. Verify that the CAD System contains a minimum supply of 2500 gals. of liquid nitrogen twice per week.

   c. At each cold shutdown verify that each solenoid operated air/nitrogen valve in the flow path can be opened with a containment isolation signal present (containment isolation override).

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TABLE 3.7.A
PRIMARY CONTAINMENT ISOLATION VALVES

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
1	Main steamline isolation valves (FCV-1-14,26,37,&51;1-15, 27, 38, & 52)	4	4	3 < T < 5	0	GC
1	Main steamline drain isolation valves FCV-1-55 & 1-56	1	1	15	C	SC
1	Reactor Water sample line isolation valves	1	1	5	C	SC
2	RHRS shutdown cooling supply isolation valves FCV-74-48 & 47	1	1	40	C	SC
2	RHRS - LPCI to reactor FCV-74-53, 67		2	30	C	SC
2	Reactor vessel head spray isolation valves FCV-74-77, 78	1	1	30	C	SC
2	RHRS flush and drain vent to suppression chamber FCV-74-102, 103, 119, & 120		4	20	C	SC
2	Suppression Chamber Drain FCV-75-57, 58		2	15	C	SC
2	Drywell equipment drain discharge isolation valves FCV-77-15A, & 15B		2	15	0	GC
2	Drywell floor drain discharge isolation valves FCV-77-2A & 2B		2	15	0	GC

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
3	Reactor water cleanup system supply isolation valves FCV-69-1, & 2	1	1	30	0	GC
4	HPCIS steamline isolation valves FCV-73-2 & 3	1	1	20	0	GC
5	RCICS steamline isolation valves FCV-71-2 & 3	1	1	15	0	GC
6	Drywell nitrogen purge inlet isolation valves (FCV-76-18)		1	10	C	SC
6	Suppression chamber nitrogen purge inlet isolation valves (FCV-76-19)		1	10	C	SC
6	Drywell Main Exhaust isolation valves (FCV-64-29 and 30)		2	90	C	SC
6	Suppression chamber main exhaust isolation valves (FCV-64-32 and 33)		2	90	C	SC
6	Drywell/Suppression Chamber purge inlet (FCV-64-17)		1	90	C	SC
6	Drywell Atmosphere purge inlet (FCV-64-18)		1	90	C	SC

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TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
6	Suppression Chamber purge inlet (FCV-64-19)		1	100	C	SC
6	Drywell/Suppression Chamber nitrogen purge inlet (FCV-76-17)		1	10	C	SC
6	Drywell Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-31)		1	10	C	SC
6	Suppression Chamber Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-34)		1	10	C	SC
7	RCIC Steamline Drain (FCV-71-6A, 6B)		2	5	0	GC
7	RCIC Condensate Pump Drain (FCV-71-7A, 7B)		2	5	C	SC
7	HPCI Hotwell pump discharge isolation valves (FCV-73-17A, 17B)		2	5	C	SC
7	HPCI steamline drain (FCV-73-6A,6B)		2	5	0	GC
8	TIP Guide Tubes (5)		1 per guide tube	NA	C	GC

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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 bypassed 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.

298 089

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

Browns Ferry unit 2 is located at Browns Ferry Nuclear Plant site on property owned by the United States and in custody of the TVA. The site shall consist of approximately 840 acres on the north shore of Wheeler Lake at Tennessee River Mile 294 in Limestone County, Alabama. The minimum distance from the outside of the secondary containment building to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 4,000 feet.

5.2 REACTOR

A. The reactor core may contain 764 fuel assemblies consisting of 7x7 assemblies having 49 fuel rods each, 8x8 assemblies having 63 fuel rods each, and 8x8 R assemblies having 62 fuel rods each. The number of each type in the core is given in the most recent reload amendment topical report.

B. The reactor core shall contain 185 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70 percent of theoretical density.

5.3 REACTOR VESSEL

The reactor vessel shall be as described in Table 4.2-2 of the FSAR. The applicable design codes shall be as described in Table 4.2-1 of the FSAR.

5.4 CONTAINMENT

A. The principal design parameters for the primary containment shall be as given in Table 5.2-1 of the FSAR. The applicable design codes shall be as described in Section 5.2 of the FSAR.

B. The secondary containment shall be as described in Section 5.3 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with the standards set forth in Section 5.2.3.4 of the FSAR.

5.5 FUEL STORAGE

A. The arrangement of fuel in the new-fuel storage facility shall be such that k_{eff}, for dry conditions, is less than 0.90 and flooded is less than 0.95 (Section 10.2 of FSAR).

298 090

A summary of the more significant discussions and conclusions of the NSRB will be transmitted along with the final minutes.

7. Charter

A written charter delineating the establishment, composition, and mission of the NSRB and the dissemination of NSRB minutes and reports shall be maintained; this may be amended as required. The charter shall identify the responsibility and authority of the NSRB in conducting reviews, including responsibility to identify problems and to recommend solutions to the Manager of Power.

B. Plant Operations Review Committee (PORC)

1. Membership

The PORC shall consist of the plant superintendent, electrical maintenance supervisor, mechanical maintenance supervisor, instrument maintenance supervisor, health physics supervisor, operations supervisor, results supervisor, and QA staff supervisor. An assistant plant supervisor may serve as an alternate committee member when his supervisor is absent.

The plant superintendent will serve as chairman of the PORC. The assistant plant superintendent will serve as chairman in the absence of the plant superintendent.

2. Meeting Frequency

The PORC shall meet at regular monthly intervals and for special meetings as called by the chairman or as requested by individual members.

3. Quorum

Superintendent or assistant superintendent, plus five of the seven other members, or their alternate, will constitute a quorum. A member will be considered present if he is in telephone communication with the committee.

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plant, the applicable codes required fatigue usage evaluation for the reactor pressure vessel only. The locations to be monitored shall be:

1. The feedwater nozzles

2. The shell at or near the waterline

3. The flange studs

b. Recording, Evaluating, and Reporting

(1) Transients that occur during plant operations will be reviewed and a cumulative fatigue usage factor determined.

(2) For transients which are more severe than the transients evaluated in the stress report, code fatigue usage calculations will be made and tabulated separately.

(3) In the **monthly** Operating Report, the fatigue usage factor determined for the transients defined in (1) and (2) above shall be added and a cumulative fatigue usage factor to date shall be listed. When the cumulative usage factor reaches a value of 1.0, an inservice inspection shall be included for the specific location at the next scheduled inspection (3-1/3-year interval) period and 3-1/3-year intervals thereafter, and a subsequent evaluation performed in accordance with the rules of ASME Section XI Code if any flaw indications are detected. The results of the evaluation shall be submitted in a Special Report (Section 6.7.3) for review by the Commission.

B. Except where covered by applicable regulations, items 1 through 8 above shall be retained for a period of at least 5 years and items 9 through 17 shall be retained for the life of the plant. A complete inventory of radioactive materials in possession shall be maintained current at all times.

298 092

(b). Annual Report

(1) A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

(2) A report of facility changes, tests or experiments required pursuant to 10CFR50.59(b).

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate Regional Office, to be submitted no later than the tenth of each month following the calendar month covered by the report. A narrative summary of operating experience shall be submitted in the above schedule.

2. Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

298 093

B. Source Tests

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

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

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

a.	Secondary Containment Leak Rate Testing (5)	4.7.C	Within 90 days of completion of each test.
b.	Fatigue Usage Evaluation	6.6	monthly Operating Report
c.	Seismic Instrumentation Inoperability	3.2.J.3	Within 10 days after 30 days of inoperability

FOOTNOTES

1. A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

2. The term "forced reduction in power" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the load on the unit be reduced for corrective action immediately or up to and including the very next weekend. Note that routine preventive maintenance, surveillance, and calibration activities requiring power reductions are not covered by this section.

3. The term "forced outage" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the unit be removed from service for corrective action immediately or up to and including the very next weekend.

4. This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

5. Each integrated leak rate test of the secondary containment shall be the subject of a summary technical report. This report should include data on the wind speed, wind direction, concurrent reactor building pressure, and emergency ventilation flow rate. The report shall also include analyses and interpretations of those data which demonstrate compliance with the specified leak rate limits.

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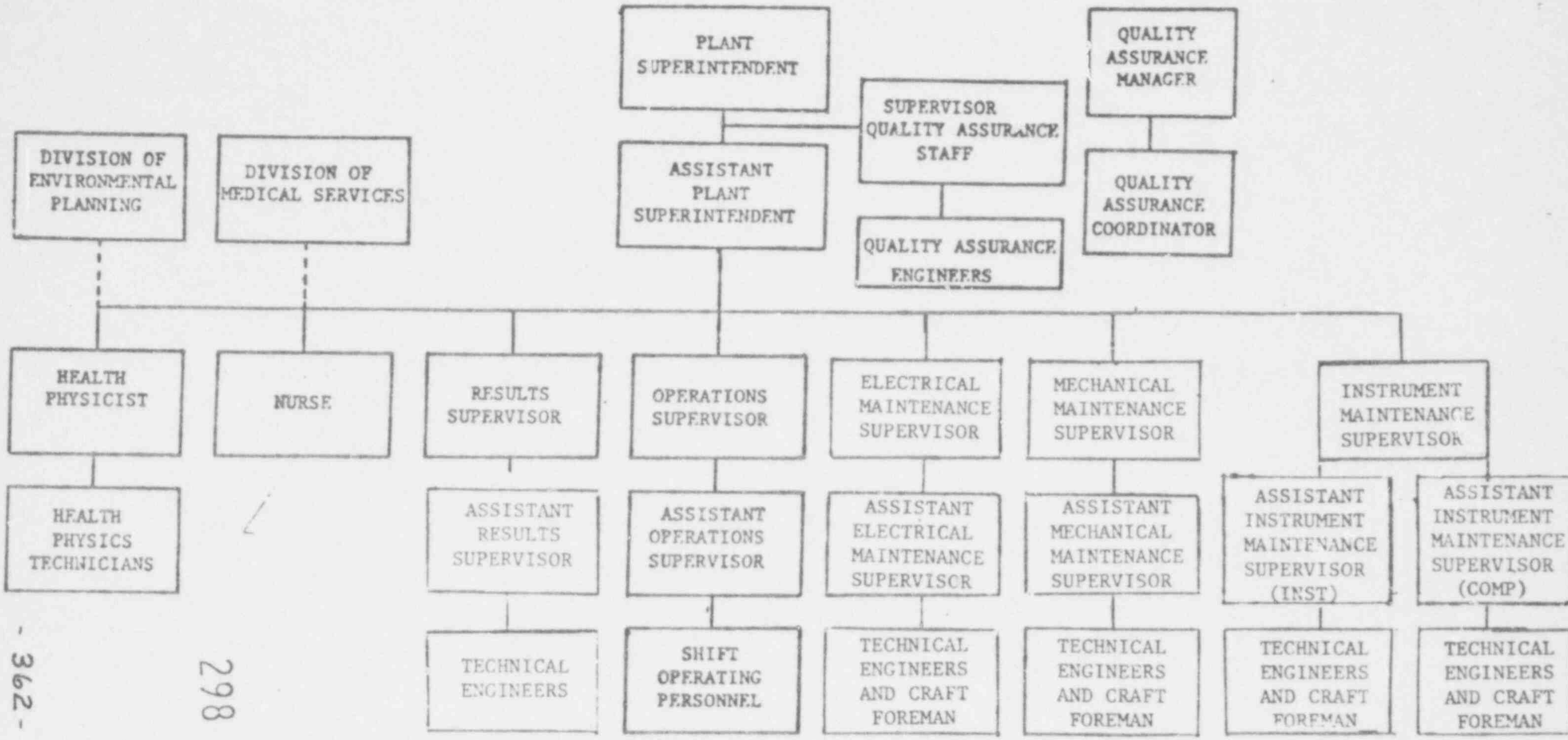

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BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FUNCTIONAL ORGANIZATION
FIGURE 6.1-2

PROPOSED CHANGES TO BROWNS FERRY

UNIT 3 TECHNICAL SPECIFICATIONS

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

Min. No. of Operable Inst. Channels Per Trip System (1)	Trip Function	Trip Level Setting	Shut-down	Modes in Which Function Must Be Operable: Refuel (7)	Startup/Hot Standby	[illegible]	Action (1)
4	Turbine Stop Valve Closure	≤ 10% Valve Closure		X(4)	X(4)	X(4)	[illegible].A or 1.D
2	Turbine Control Valve - Loss of Control 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.[illegible] or 1.C
2	Main Steam Line High Radiation (14)	≤ 6X Normal Full Power Background (20)		X(9)	X(9)	X(9)	1.A or 1.C

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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 ystem (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 3.0 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.

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TABLE 4.1.A

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM High Flux	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup not to exceed once per week
Inoperative	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup not to exceed once per week
APRM High Flux (15% scram)	C	Trip Output Relays (4)	Before Each Startup and Weekly When Required to be Operable
High Flux	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	(6)	(6)
High Reactor Pressure	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water Level (5)	A	Trip Channel and Alarm	Once/Month (1)
High Water Level in Scram Discharge Tank	A	Trip Channel and Alarm	Every 3 Months
Turbine Condenser Low Vacuum	A	Trip Channel and Alarm	Once/Month (1)

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

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

5. 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.

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TABLE 3.2.E
INSTRUMENTATION THAT MONITORS LEAKAGE INTO DRYWELL

System (2)	Setpoints		Action	Remarks
Equipment Drain			(1)	1. Used to determine identifiable reactor coolant leakage.
Flow Integrator	N/A			2. Considered part of sump system.
Sump Fill Rate Timer	≥ 20.1 min.			
Sump Pump Out Rate Timer	≤ 13.4 min.			
Floor Drain			(1)	1. Used to determine unidentifiable reactor coolant leakage.
Flow Integrator	N/A			2. Considered part of sump system.
Sump Fill Rate Timer	≥ 80.4 min.			
Sump Pump Out Rate Timer	≤ 8.9 min.			
Drywell Air Sampling	Gas and Particulate	3 x Average Background	(3)	

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NOTES:

(1) Whenever a system is required to be operable, there shall be one operable system either automatic or manual, or the action required in Section 3.6.C.2 shall be taken.

(2) An alternate system to determine the leakage flow is a manual system whereby the time between sump pump starts is monitored. The time interval will determine the leakage flow because the valume of the sump will be known.

(3) Upon receipt of alarm, immediate action will be taken to confirm the alarm and assess the possibility of increased leakage. Refer to Tech Spec Sections 3.6.C.2 and 3.6.C.3 for drywell air sampling system out of service.

3.7 CONTAINMENT SYSTEMS

D. 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 CONTAINMENT SYSTEMS

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 except where specific written relief from ASME Section XI has been granted by NRC pursuant to 10 CFR 50, Section 50.55a(g)(b)(i).

(2) With the reactor power less than 75% trip main steam isolation valves individually and verify closure time.

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3.7 CONTAINMENT SYSTEMS

G. Containment Atmosphere Dilution System (CAD)

1. The Containment Atmosphere Dilution (CAD) System shall be operable with:

   a. Two independent systems capable of supplying nitrogen to the drywell and torus.

   b. A minimum supply of 2500 gallons of liquid nitrogen per system.

2. The Containment Atmosphere Dilution (CAD) System shall be operable whenever the reactor mode switch is in the "RUN" position.

3. If one system is inoperable, the reactor may remain in operation for a period of 30 days provided all active components in the other system are operable.

4. If Specification 3.7.G.1 and 3.7.G.2, or 3.7.G.3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

4.7 CONTAINMENT SYSTEMS

G. Containment Atmosphere Dilution System (CAD)

1. System Operability

   a. At least once per month, cycle each solenoid operated air/nitrogen valve with its hand switch (no containment isolation override) through at least one complete cycle of full travel and verify that each manual valve in the flow path is open.

   b. Verify that the CAD System contains a minimum supply of 2500 gals. of liquid nitrogen twice per week.

   c. At each cold shutdown verify that each solenoid operated air/nitrogen valve in the flow path can be opened with a containment isolation signal present (containment isolation override).

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TABLE 3.7.A
PRIMARY CONTAINMENT ISOLATION VALVES

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
3	Reactor water cleanup system supply isolation valves (FCV-69-1 & 2)	1	1	30	0	GC
4	HPCIS steamline isolation valves (FCV-73-2 & 3)	1	1	20	0	-
5	RCICS steamline isolation valves (FCV-71-2 & 3)	1	1	15	0	GC
6	Drywell nitrogen purge inlet isolation valves (FCV-76-18)		1	10	C	SC
6	Suppression chamber nitrogen purge inlet isolation valves (FCV-76-19)		1	10	C	SC
6	Drywell Main Exhaust isolation valves (FCV-64-29 and 30)		2	90	C	SC
6	Suppression chamber main exhaust isolation valves (FCV-64-32 and 33)		2	90	C	SC
6	Drywell/Suppression Chamber purge inlet (FCV-64-17)		1	90	C	SC
6	Drywell Atmosphere purge inlet (FCV-64-18)		1	90	C	SC

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TABLE 3.7.A
PRIMARY CONTAINMENT ISOLATION VALVES

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
6	Suppression Chamber purge inlet (FCV-64-19)		1	100	C	SC
6	Drywell/Suppression Chamber nitrogen purge inlet (FCV-76-17)		1	10	C	SC
6	Drywell Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-31)		1	10	C	SC
6	Suppression Chamber Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-34)		1	10	C	SC
7	RCIC Steamline Drain (FCV-71-6A & 6B)		2	5	0	GC
7	RCIC Condensate Pump Drain (FCV-71-7A & 7B)		2	5	C	SC
7	HPCI Hotwell pump discharge isolation valves (FCV-73-17A & 17B)		2	5	C	SC
7	HPCI steamline drain (FCV-73-6A & 6B)		2	5	0	GC
8	TIP Guide Tubes (5)		1 per guide tube	NA	C	GC

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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 bypassed 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.

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5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

Browns Ferry units 1, 2, and 3 are located at Browns Ferry Nuclear Plant site on property owned by the United States and in custody of the TVA. The site shall consist of approximately 840 acres on the north shore of Wheeler Lake at Tennessee River Mile 294 in Limestone County, Alabama. The minimum distance from the outside of the secondary containment building to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 4,000 feet.

5.2 REACTOR

A. The reactor core may contain 764 fuel assemblies consisting of 8x8 assemblies having 63 fuel rods each, and 8x8 R assemblies having 62 fuel rods each. The number of each type in the core is given in the most recent reload amendment topical report.

B. The reactor core shall contain 185 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C).compacted to approximately 70 percent of theoretical density.

5.3 REACTOR VESSEL

The reactor vessel shall be as described in Table 4.2-2 of the FSAR. The applicable design codes shall be as described in Table 4.2-1 of the FSAR.

5.4 CONTAINMENT

A. The principal design parameters for the primary containment shall be given in Table 5.2-1 of the FSAR. The applicable design codes shall be as described in Section 5.2 of the FSAR.

B. The secondary containment shall be as described in Section 5.3 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with the standards set forth in Section 5.2.3.4 of the FSAR.

5.5 FUEL STORAGE

A. The arrangement of the fuel in the new-fuel storage facilitity shall be such that k_{eff}, for dry conditions,

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A summary of the more significant discussions and conclusions of the NSRB will be transmitted along with the final minutes.

7. Charter

A written charter delineating the establishment, composition, and mission of the NSRB and the dissemination of NSRB minutes and reports shall be maintained; this may be amended as required. The charter shall identify the responsibility and authority of the NSRB in conducting reviews, including responsibility to identify problems and to recommend solutions to the Manager of Power.

B. Plant Operations Review Committee (PORC)

1. Membership

The PORC shall consist of the plant superintendent, electrical maintenance supervisor, mechanical maintenance supervisor, instrument maintenance supervisor, health physics supervisor, operations supervisor, results supervisor, and QA staff supervisor. An assistant plant supervisor may serve as an alternate committee member when his supervisor is absent.

The plant superintendent will serve as chairman of the PORC. The assistant plant superintendent will serve as chairman in the absence of the plant superintendent.

2. Meeting Frequency

The PORC shall meet at regular monthly intervals and for special meetings as called by the chairman or as requested by individual members.

3. Quorum

Superintendent or assistant superintendent, plus five of the seven other members, or their alternate, will constitute a quorum. A member will be considered present if he is in telephone communication with the committee.

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plant, the applicable codes required fatigue usage evaluation for the reactor pressure vessel only. The locations to be monitored shall be:

1. The feedwater nozzles

2. The shell at or near the waterline

3. The flange studs

b. Recording, Evaluating, and Reporting

(1) Transients that occur during plant operations will be reviewed and a cumulative fatigue usage factor determined.

(2) For transients which are more severe than the transients evaluated in the stress report, code fatigue usage calculations will be made and tabulated separately.

(3) In the monthly Operating Report, the fatigue usage factor determined for the transients defined in (1) and (2) above shall be added and a cumulative fatigue usage factor to date shall be listed. When the cumulative usage factor reaches a value of 1.0, an inservice inspection shall be included for the specific location at the next scheduled inspection (3-1/3-year interval) period and 3-1/3-year intervals thereafter, and a subsequent evaluation performed in accordance with the rules of ASME Section XI Code if any flaw indications are detected. The results of the evaluation shall be submitted in a Special Report (Section 6.7.3) for review by the Commission.

B. Except where covered by applicable regulations, items 1 through 8 above shall be retained for a period of at least 5 years and items 9 through 17 shall be retained for the life of the plant. A complete inventory of radioactive materials in possession shall be maintained current at all times.

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(b) Annual Report

(1) A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

(2) A report of facility changes, tests or experiments required pursuant to 10CFR50.59(b).

c. Monthly Operating Report: Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate Regional Office, to be submitted no later than the tenth of each month following the calendar month covered by the report. A narrative summary of operating experience shall be submitted in the above schedule.

2. Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

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B. Source Tests

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

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

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

a. Secondary Containment Leak Rate Testing(5)	4.7.C	Within 90 days of completion of each test.
b. Fatigue Usage Evaluation	6.6	monthly Operating Report

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U-3

6.0 ADMINISTRATIVE CONTROLS

FOOTNOTES

1. A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

2. The term "forced reduction in power" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the load on the unit be reduced for corrective action immediately or up to and including the very next weekend. Note that routine perventive maintenance, surveillance, and calibration activities requiring power reductions are not covered by this section.

3. The term "forced outage" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the unit be removed from service for corrective action immediately or up to and including the very next weekend.

4. This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

5. Each integrated leak rate test of the secondary containment shall be the subject of a summary technical report. This report should include data on the wind speed, wind direction, concurrent reactor building pressure, and emergency ventilation flow rate. The report shall also include analyses and interpretations of those data which demonstrate compliance with the specified leak rate limits.

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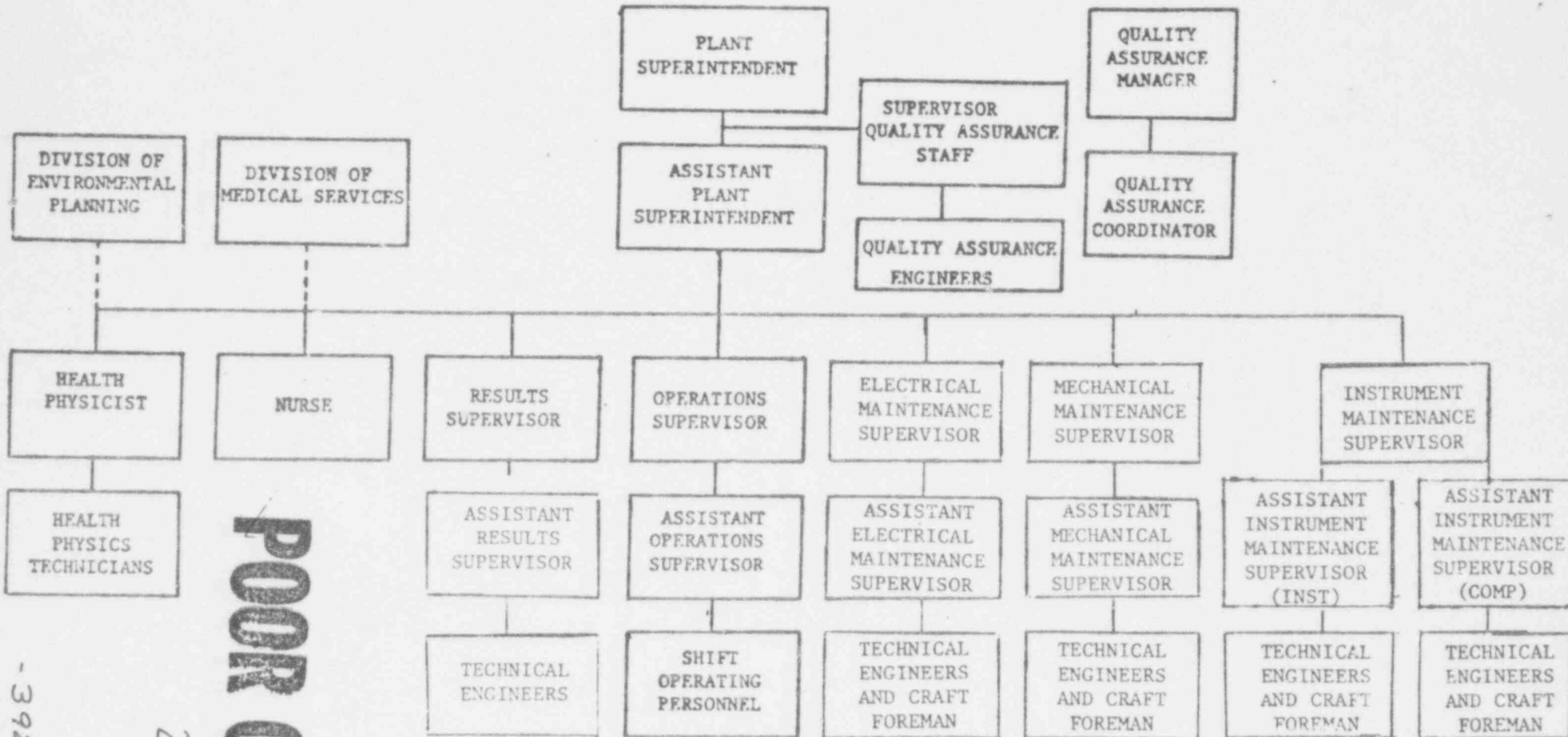

BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

FUNCTIONAL ORGANIZATION
FIGURE 6.1-2

298 114

ENCLOSURE 2

298 115

REASONS AND JUSTIFICATIONS
FOR PROPOSED CHANGES
TO BROWNS FERRY
NUCLEAR PLANT TECHNICAL
SPECIFICATIONS FOR UNITS
1, 2, AND 3

UNIT 1

Page 34 (Table 3.1.A) and page 36 (Note 20) ---- It is proposed to change the Main Steam Line Radiation Monitor (MSLRM) setpoints from 1.5X and 3X background to higher values, name, 3X and 6X background. The recommended setpoint in GEK 779, Vol. IV is 3X background for alarm and 6X background for isolation trip. The accuracy of the MSLRM is ± 25 percent for the 100- 1000 MR/HR range with a ± 3 percent per week drift (Reference GEK 32426A). The current setpoint of 1.5X add 3X background is very hard to maintain without exceeding technical specifications when the instrument drifts down or having high alarms when the instrument drifts up (see Attachment 1). The MSLRM is installed to detect and respond to increases in main steam line radiation that might indicate gross fuel cladding ruptures (NFDO-10174 Oct. 1977). Neither precision nor accuracy is required to measure gross failures as indicated by GE specifications. Therefore, raising the setpoints will not degrade instrument response to gross failure, but will allow instrument accuracy to be taken into effect when setpoints are calculated.

Page 37 ---------------- It is proposed to change the IRM Functional Test Minimum Frequency for the high flux and inoperative modes to read: "Once per week during refueling and before each startup not to exceed once per week." This change is proposed in order to provide agreement with the functional test requirements of Table 4.2.C.

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Page 74 - - - - - - - - - - - - - - - It is proposed to change Note 4 to read as follows:

"SRM's A and C downscale function is bypassed when IRM's A, C, E, and G are above range 2. SRM detector not in startup position is bypassed when the count rate is $\geq$ 100 CPS or the above condition is satisfied." This change will make the specification for instrumentation that initiate rod blocks agree with the as installed equipment.

Page 77 - - - - - - - - - - - - - - - It is proposed to add the following statement to Note 3 of Table 3.2.E "Refer to Technical Specification Sections 3.6.C.2 and 3.6.C.3 for drywell air sampling system out of service." This addition should clarify the requirements of Section 3.2.E.

Page 242 - - - - - - - - - - - - - - - It is proposed to change specification 4.7.D.1.b(1) to read: ". . . .closed and reopened except where specific written relief from ASME Section XI requirements has been granted by NRC pursuant to 10 CFR 50, Section 50.55a(g)(b)(1)." The NRC has identified certain valves which are not to be cycled except at cold shutdowns. HPCI and RCIC steam supply valves are examples of these valves.

Page 248 - - - - - - - - - - - - - - - It is prop d to change Section 4.7.G.1 to read as shown. This proposal reflects discussions held with NRC on June 1, 1979. The proposed change prevents the possibility of overriding a containment isolation signal, except during testing while in the cold shutdown condition.

Page 250 - - - - - - - - - - - - - - - It is proposed to change "FCV 74-57, 58" to "FCV 75-57,58" - the present valve numbers as quoted are incorrect.

Page 251 - - - - - - - - - - - - - - - It is proposed to delete valve FCV 69-12 from Table 3.7.A. This valve is not a containment isolation valve. Isolation is provided by check valves 69-579 and 3-572.

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Page 252 - - - - - - - - - - - - - - - It is proposed to change "FCV 75-57, 58" to "FCV 73-6A, 6B" and to change the normal operation of FCV 71-7A, 7B from "O" to "C" and change Action on Initiating signal from "GC" to "SC." The present valve numbers as quoted are incorrect. The normal valve position of FCV 71-7A, 7B is closed.

Page 304 - - - - - - - - - - - - - - - It is proposed to delete the word "withdrawn" in tech spec 4.10.A.3 and insert "bypassed." This would remove an otherwise ambiguous statement.

Page 330 - - - - - - - - - - - - - - - It is proposed to revise paragraph 5.2.A to delete the number of assemblies of each type of fuel in the core and add the following: "The number of each type in the core is given in the most recent reload amendment topical report."

This proposed change will serve to reduce the number of cycle dependent page changes to future reload technical specifications. We consider this proposed change administrative in nature.

Page 335 - - - - - - - - - - - - - - - It is proposed to change the membership and quorum of the Plant Operations Review Committee (PORC) to reflect new plant organization.

Pages 348, 351, and 356 - - - - - - - - It is proposed to change paragraph 6.6.A.17.b(3) and 6.7.3.C.1.b from a frequency of "annual operating report" to "monthly operating report" to further comply with NRC's letter from A. Schwencer to G. Williams dated September 16, 1977. It is also proposed to add paragraph 6.7.1.b(2) in order to report cumulative fatigue usage factors for the reactor vessels monthly in the monthly operating report.

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Page 357 - - - - - - - - - - - - - - - - It is proposed to delete the words "inside and outside temperatures during the test" from footnote 5 of Section 6.0. These temperatures are not pertinent to the results of the integrated leak rate test of the secondary containment.

Page 362 - - - - - - - - - - - - - - - - It is proposed to revise Figure 6.1-2 to reflect new plant organization.

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UNIT 2

Page 34 (Table 3.1.A) and page 36 (Note 20) - - - - - - Same as for pages 34 and 36 for

Page 37 - - - - - - - - - - - - - - - - Same as page 37 for unit 1 above

Page 74 - - - - - - - - - - - - - - - - Same as page 74 for unit 1 above

Page 77 - - - - - - - - - - - - - - - - Same as page 77 for unit 1 above

Page 242 - - - - - - - - - - - - - - - Same as page 242 for unit 1 above

Page 248 - - - - - - - - - - - - - - - Same as page 248 for unit 1 above

Page 250 - - - - - - - - - - - - - - - Same as page 250 for unit 1 above

Page 251 - - - - - - - - - - - - - - - Same as page 251 for unit 1 above

Page 252 - - - - - - - - - - - - - - - Same as page 252 for unit 1 above

Page 304 - - - - - - - - - - - - - - - Same as page 304 for unit 1 above

Page 330 - - - - - - - - - - - - - - - Same as page 330 for unit 1 above

Page 335 - - - - - - - - - - - - - - - Same as page 335 for unit 1 above

Pages 348, 351, and 356 - - - - - - - Same as pages 348, 351, and 356 for unit 1 above

Page 357 - - - - - - - - - - - - - - - Same as page 357 for unit 1 above

Page 362 - - - - - - - - - - - - - - - Same as page 362 for unit 1 above

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UNIT 3

Page 33 (Table 3.1.A) and page 35 (Note 20) - - - - -	Same as for pages 34 and 36 for unit 1 above
Page 36 - - - - - - - - - - - - - - - -	Same as page 37 for unit 1 above
Page 77 - - - - - - - - - - - - - - - -	Same as page 74 for unit 1 above
Page 80 - - - - - - - - - - - - - - - -	Same as page 77 for unit 1 above
Page 248 - - - - - - - - - - - - - - - -	Same as page 248 for unit 1 above
Page 254 - - - - - - - - - - - - - - - -	Same as page 242 for unit 1 above
Page 263 - - - - - - - - - - - - - - - -	Same as page 251 for unit 1 above
Page 264 - - - - - - - - - - - - - - - -	It is proposed to change the normal position of FCV 71-7A, 7B from "O" to "C" and change Action on Initiating Signal from "GC" to "SC". The normal position of FCV 71-7A, 7B is closed.
Page 335 - - - - - - - - - - - - - - - -	Same as page 304 for unit 1 above
Page 360 - - - - - - - - - - - - - - - -	Same as page 330 for unit 1 above
Page 365 - - - - - - - - - - - - - - - -	Same as page 335 for unit 1 above
Page 378, 381, and 386 - - - - - - - -	Same as pages 348, 351, and 356 for unit 1 above
Page 387 - - - - - - - - - - - - - - - -	Same as page 357 for unit 1 above
Page 392 - - - - - - - - - - - - - - - -	Same as page 362 for unit 1 above

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ATTACHMENT 1

POOR ORIGINAL