LICENSE AUTHORITY FILE COPY

DO NOT REMOVE

Posted Amdt. 108 to DPR-52

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

October 16, 1984

Docket Nos. 50-259/260/296

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

Dear Mr. Parris:

The Commission has issued the enclosed Amendment Nos. 114, 108 and 82 to Facility Operating License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3. These amendments are in response to your application dated April 30, 1982 (TVA BFNP TS 173) and supplemented by a letter dated June 10, 1982.

These amendments change the Appendices A and B Technical Specifications to (1) revise the reactor water cleanup system isolation instrumentation operability requirements (2) revise RHRSW pump operability requirements, (3) revise the suppression chamber water level datum for HPCI suction switchover (4) correct a typographical error, (5) delete surveillance requirements for RWCU system compartment temperature detectors (6) clarify residual heat removal pump operability and surveillance requirements (7) revise drywell-to-torus leak rate testing bases (8) revise the requirements on control rod drive maintenance when fuel is present around the rods (9) on Unit 2, revise the surveillance requirements for standby coolant supply pumps and (10) revise raw milk sampling requirements.

Your change requests to expand the definition of rated power (Technical Specifications Section 1.0.N) to include specific limits for perturbations of power above the rated power level when operating at rated steady state power, and revise the definition of "limiting condition for operation", were withdrawn by a telecon (R. Rogers/W. Long) on September 26, 1984.

A copy of the Safety Evaluation is also enclosed.

Sincerely,

Richard J. Clark

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

Enclosures:
1. Amendment No. 114 to License No. DPR-33
2. Amendment No. 108 to License No. DPR-52
3. Amendment No. 82 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

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

cc:

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

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

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

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

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

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

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

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

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

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

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

James A. Coffey
Site Director, BFNP
Tennessee Valley Authority
Post Office Box 2000
Decatur, Alabama 35602

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 108
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated April 30, 1982, as supplemented by letter dated June 10, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 108, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 16, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 108

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

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

56
61
64
66
71
88
97
110
147
153
164
272
302
303
304
310

2. Add new page 303A

Revise Appendix B as follows:

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

42

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

Amendment No. ~~82~~, 108

TABLE 3.2.A (Continued)

Minimum No. Instrument Channels Operable per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2(12)	Instrument Channel - Main Steam Line Tunnel High Temperature	≤ 200°F	B	1. Above trip setting initiates Main Steam Line Isolation
2(14)	Instrument Channel - Reactor Water Cleanup System Floor Drain High Temperature	160 - 180°F	C	1. Above trip setting initiates Isolation of Reactor Water Cleanup Line from Reactor and Reactor Water Return Line.
2	Instrument Channel - Reactor Water Cleanup System Space High Temperature	160 - 180°F	C	1. Same as above
1	Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	≤ 100 mr/hr or downscale	G	1. 1 upscale or 2 downscale will a. Initiate SGTS b. Isolate reactor zone and refueling floor. c. Close atmosphere control system.
1	Instrument Channel - Reactor Building Ventilation High Radiation Refuleing Zone	≤ 100 mr/hr or downscale	F	1. 1 upscale or 2 downscale will a. Initiate SGTS. b. Isolate refueling floor. c. Close atmosphere control system.
2 (7)(8)	Instrument Channel SGTS Flow - Train A Heaters	Charcoal Heaters ≤ 2000 cfm R. H. Heaters ≤ 2000 cfm	H and (A or F)	1. Below 2000 cfm, trip setting charcoal heaters will turn on. 2. Below 2000 cfm, trip setting R. H. heaters will shut off.
2 (7)(8)	Instrument Channel SGTS Flow - Train B Heaters	Charcoal Heaters ≤ 2000 cfm R.H. Heaters ≤ 2000 cfm	H and (A or F)	1. Below 2000 cfm, trip setting charcoal heaters will turn on. 2. Below 2000 cfm, trip setting R.H. heaters will shut off.
2 (7)(8)	Instrument Channel SGTS Flow - Train C Heaters	Charcoal Heaters ≤ 2000 cfm R.H. Heaters ≤ 2000 cfm	H and (A or F)	1. Below 2000 cfm, trip setting charcoal heaters will turn on. 2. Below 2000 cfm, trip setting R.H. heaters will shut off.

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

7. A train is considered a trip system.

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

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

10. Refer to Table 3.7.A and its notes for a listing of Isolation Valve Groups and their initiating signals.

11. A channel may be placed in an inoperable status for up to four hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

12. A channel contains four sensors, all of which must be operable for the channel to be operable.

Power operations permitted for up to 30 days with 15 of the 16 temperature switches operable.

In the event that normal ventilation is unavailable in the main steam line tunnel, the high temperature channels may be bypassed for a period of not to exceed four hours. During periods when normal ventilation is not available, such as during the performance of secondary containment leak rate tests, the control room indicators of the affected space temperatures shall be monitored for indications of small steam leaks. In the event of rapid increases in temperature (indicative of steam line break), the operator shall promptly close the main steam line isolation valves.

13. The nominal setpoints for alarm and reactor trip (1.5 and 3.0 times background, respectively) are established based on the normal background at full power. The allowable setpoints for alarm and reactor trip are 1.2-1.8 and 2.4-3.6 times background, respectively.

14. Requires two independent channels from each physical location, there are two locations.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys.(1)	Function	Trip Level Setting	Action	Remarks
1	Instrument Channel - Reactor Low Pressure (PS-68-93 & 94, SW #1)	100 psig ± 15	A	1. Below trip setting in conjunction with containment isolation signal and both suction valves open will close RHR (LPCI) admission valves.
2	Core Spray Auto Sequencing Timers (5)	$6 \le t \le 8$ secs.	B	1. With diesel power 2. One per motor
2	LPCI Auto Sequencing Timers (5)	$0 \le t \le 1$ sec.	B	1. With diesel power 2. One per motor
1	RHRSW A1, B3, C1, and D3 Timers	$13 \le t \le 15$ sec.	A	1. With diesel power 2. One per pump
2	Core Spray and LPCI Auto Sequencing Timers (6)	$0 \le t \le 1$ sec. $6 \le t \le 8$ sec. $12 \le t \le 16$ sec. $18 \le t \le 24$ sec.	B	1. With normal power 2. One per CSS motor 3. Two per RHR motor
1	RHRSW A1, B3, C1, and D3 Timers	$27 \le t \le 29$ sec.	A	1. With normal power 2. One per pump

Amendment No. 85, 108

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
1	Core Spray Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	ADS Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems and valves.
1	HPCI Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	RCIC Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1(2)	Instrument Channel - Condensate Header Low Level (LS-73-55A & B)	≥ Elev. 551'	A	1. Below trip setting will open HPCI suction valves to the suppression chamber.
1(2)	Instrument Channel - Suppression Chamber High Level	≤ 7" above instrument zero	A	1. Above trip setting will open HPCI suction valves to the suppression chamber.
2(2)	Instrument Channel - Reactor High Water Level	≤ 583" above vessel zero.	A	1. Above trip setting trips RCIC turbine.
1	Instrument Channel - RCIC Turbine Steam Line High Flow	≤ 450" H_2O (7)	A	1. Above trip setting isolates RCIC system and trips RCIC turbine.

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS System is required by section 3.5 to be operable, there shall be two operable trip systems except as noted. If a requirement of the first column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the first column reduced by more than one, action B shall be taken.

   Action:

   A. Repair in 24 hours. If the function is not operable in 24 hours, take action B.

   B. Declare the system or component inoperable.

   C. Immediately take action B until power is verified on the trip system.

   D. No action required, indicators are considered redundant.

2. In only one trip system.

3. Not considered in a trip system.

4. Requires one channel from each physical location (there are 4 locations) in the steam line space.

5. With diesel power, each RHRS pump is scheduled to start immediately and each CSS pump is sequenced to start about 7 sec later.

6. With normal power, one CSS and one RHRS pump is scheduled to start instantaneously, one CSS and one RHRS pump is sequenced to start after about 7 sec with similar pumps starting after about 14 sec and 21 sec, at which time the full complement of CSS and RHRS pumps would be operating.

7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of water corresponds to at least 150% above maximum steady state steam flow to assure that spurious isolation does not occur while ensuring the initiation of isolation following a postulated steam line break. Similarly, the HPCIS setting of 90 psi corresponds to at least 150% above maximum steady state flow while also ensuring the initiation of isolation following a postulated break.

8. Note 1 does not apply to this item.

9. The head tank is designed to assure that the discharge piping from the CS and RHR pumps are full. The pressure shall be maintained at or above the values listed in 3.5.H, which ensures water in the discharge piping and up to the head tank.

TABLE 4.2.A

SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency	Instrument Check
Group 6 Logic	once/operating cycle (18)	N/A	N/A
Group 8 (Initiating) Logic	Checked during channel functional test. No further test required.	N/A	N/A
Reactor Building Isolation (refueling floor) Logic	once/6 months (18)	(6)	N/A
Reactor Building Isolation (reactor zone) Logic	once/6 months (18)	(6)	N/A
SGTS Train A Logic	once/6 months (19)	N/A	N/A
SGTS Train B Logic	once/6 months (19)	N/A	N/A
SGTS Train C Logic	once/6 months (19)	N/A	N/A
Static Pressure Control (refueling floor) Logic	once/operating cycle (18)	(6)	N/A
Static Pressure Control (reactor zone) Logic	once/operating cycle (18)	(6)	N/A
Instrument Channel - Reactor Cleanup System Floor Drain High Temperature	(1)	once/operating cycle	N/A
Instrument Channel - Reactor Cleanup System Space High Temperature	(1)	once/operating cycle	N/A

TABLE 4.2.B (Continued)

Function	Functional Test	Calibration	Instrument Check
Instrument Channel Reactor Low Pressure (PS-68-93 & 94)	(1)	once/3 months	none
Core Spray Auto Sequencing Timers (Normal Power)	(4)	once/operating cycle	none
Core Spray Auto Sequencing Timers (Diesel Power)	(4)	once/operating cycle	none
LPCI Auto Sequencing Timers (Normal Power)	(4)	once/operating cycle	none
LPCI Auto Sequencing Timers (Diesel Power)	(4)	once/operating cycle	none
RHRSW A1, B3, C1, D3 Timers (Normal Power)	(4)	once/operating cycle	none
RHRSW A1, B3, C1, D3 Timers (Diesel Power)	(4)	once/operating cycle	none
ADS Timer	(4)	once/operating cycle	none

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

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

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

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

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

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

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

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

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

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

23. Deleted

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

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

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

27. The functional test frequency decreased to once/3 months to reduce challenges to relief valves per NUREG-0737, Item II.K.3.16.

LIMITING CONDITIONS FOR OPERATION

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the cold shutdown condition within 24 hours.

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and all diesel generators and all access paths of the RHRS (containment cooling mode) are operable.

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode)

SURVEILLANCE REQUIREMENTS

4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

4. No additional surveillance required.

5. When it is determined that one RHR pump (containment cooling mode) or associated heat exchanger is inoperable at a time when operability is required, the remaining RHR pumps (containment cooling mode), the associated heat exchangers and diesel generators, and all active components in the access paths of the RHRS (containment cooling mode) shall be demonstrated to be operable immediately and weekly thereafter until the inoperable RHR pump (containment cooling mode) and associated heat exchanger is returned to normal service.

6. When it is determined that two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable at a time when operability is required, the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all active components in the access paths of the RHRS (containment cooling

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.5.C (Continued)

4. Three of the D1, D2, B1, B2 RHRSW pumps assigned to the RHR heat exchanger supplying the standby coolant supply connection may be inoperable for a period not to exceed 30 days provided the operable pump is aligned to supply the RHR heat exchanger header and the associated diesel generator and essential control valves are operable.

5. The standby coolant supply capability may be inoperable for a period not to exceed ten days.

6. If specifications 3.5.C.2 through 3.5.C.5 are not met, an orderly shutdown shall be initiated and the unit placed in the cold shutdown condition within 24 hours.

7. There shall be at least 2 RHRSW pumps, associated with the selected RHR pumps, aligned for RHR heat exchanger service for each reactor vessel containing irradiated fuel.

4.5.C (Continued)

4. When it is determined that three of the RHRSW pumps supplying standby coolant are inoperable at a time when operability is required, the operable RHRSW pump and its associated diesel generator and the RHR heat exchanger header and associated essential control valves shall be demonstrated to be operable immediately and every 15 days thereafter.

3.5. BASES

Should the capability for ~~providing~~ flow through the cross-connect lines be lost, a ten day repair time is allowed before shutdown is required. This repair time is justified based on the very small probability for ever needing RHR pumps and heat exchangers to supply an adjacent unit.

REFERENCES

1. Residual Heat Removal System (BFNP FSAR subsection 4.8)
2. Core Standby Cooling Systems (BFNP FSAR Section 6)

3.5.C RHR Service Water System and Emergency Equipment Cooling Water System (EECWS)

There are two EECW headers (north and south) with four automatic starting RHRSW pumps on each header. All components requiring emergency cooling water are fed from both headers thus assuring continuity of operation if either header is operable. Each header alone can handle the flows to all components. Two RHRSW pumps can supply the full flow requirements of all essential EECW loads for any abnormal or postaccident situation.

There are four RHR heat exchanger headers (A, B, C, & D) with one RHR heat exchanger from each unit on each header. There are two RHRSW pumps on each header; one normally assigned to each header (A2, B2, C2, or D2) and one on alternate assignment (A1, B1, C1, or D1). One RHR heat exchanger header can adequately deliver the flow supplied by both RHRSW pumps to any two of the three RHRSW heat exchangers on the header. One RHRSW pump can supply the full flow requirement of one RHR heat exchanger. Two RHR heat exchangers can more than adequately handle the cooling requirements of one unit in any abnormal or postaccident situation.

The RHR Service Water Systems was designed as a shared system for three units. The specification, as written, is conservative when consideration is given to particular pumps being out of service and to possible valving arrangements. If unusual operating conditions arise such that more pumps are out of service than allowed by this specification, a special case request may be made to the NRC to allow continued operation if the actual system cooling requirements can be assured.

Should three of the four RHRSW pumps normally or alternately assigned to the RHR heat exchanger headers supplying the standby coolant supply connection become inoperable, capability for long-term fluid makeup to the unit reactor and for cooling of the unit containment remains operable. Because of the availability of makeup and cooling capability which is demonstrated to be operable immediately and with specified subsequent surveillance, a 30-day repair period is justified. Unit 2 may be supplied standby coolant from either of four pumps--B1, B2, D1, or D2. Should the capability to provide standby coolant supply be lost, a 10-day repair time is justified based on the low probability for ever needing the standby coolant supply.

opening. If the check and green or check light circuit alone is inoperable, the valve shall be considered inoperable for full closure. If the red and check light circuits are inoperable the valve shall be considered inoperable and open greater than 3°. For a light circuit to be considered operable the light must go on and off in proper sequence during the opening-closing cycle. If none of the lights change indication during the cycle, the valve shall be considered inoperable and open unless the check light stays on and the red light stays off in which case the valve shall be considered inoperable for opening.

The twelve drywell vacuum breaker valves which connect the suppression chamber and drywell are sized on the basis of the Bodega pressure suppression system tests. Ten operable to open vacuum breaker valves (18-inch) selected on this test basis and confirmed by the green lights are adequate to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling operations to a value which is within suppression system design values.

The containment design has been examined to determine that a leakage equivalent to one drywell vacuum breaker opened to no more than a nominal 3° as confirmed by the red light is acceptable.

On this basis an indefinite allowable repair time for an inoperable red light circuit on any valve or an inoperable check and green or check light circuit alone or a malfunction of the operator or disc (if nearly closed) on one valve, or an inoperable green and red or green light circuit alone on two valves is justified.

During each operating cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least 1 psi with respect to the suppression chamber pressure and held constant. The 2 psig set point will not be exceeded. The subsequent suppression chamber pressure transient (if any) will be monitored with a sensitive pressure gauge. If the drywell pressure cannot be increased by 1 psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event the leakage source will be identified and eliminated before power operation is resumed.

With a differential pressure of greater than 1 psig, the rate of change of the suppression chamber pressure must not exceed 0.38 inches of water per minute as measured over a 10 minute period, which corresponds to about 0.14 lb/sec of containment air. In the event the rate of change exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

The water in the suppression chamber is used for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

3.10 CORE ALTERATIONS

Applicability

Applies to the fuel handling and core reactivity limitations.

Objective

To ensure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.

Specification

A. Refueling Interlocks

1. The reactor mode switch shall be locked in the "Refuel" position during core alterations and the refueling interlocks shall be operable except as specified in 3.10.A.6 and 3.10.A.7 below.

2. Fuel shall not be loaded into the reactor core unless all control rods are fully inserted.

4.10 CORE ALTERATIONS

Applicability

Applies to the periodic testing of those interlocks and instrumentation used during refueling and core alterations.

Objective

To verify the operability of instrumentation and interlocks used in refueling and core alterations.

Specification

A. Refueling Interlocks

1. Prior to any fuel handling with the head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall be tested at weekly intervals thereafter until no longer required. They shall also be tested following any repair work associated with the interlocks.

2. No additional surveillance required.

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. Maintenance may be performed on a single control rod or control rod drive without removing the fuel in the control cell if the following conditions are met:

   a. The requirements of specification 3.10.A.1 are met, and

   b. All control rods diagonally and face adjacent to the maintenance rod are fully inserted and have their directional control valves electrically disarmed.

4.10.A Refueling Interlocks

3. No additional surveillance required.

4. No additional surveillance required.

5. Prior to performing control rod or control rod drive maintenance on a control cell without removing fuel assemblies the surveillance requirements of specification 4.10.A.1 shall be performed and all rods face adjacent and diagonally adjacent to the maintenance rod shall be electrically disarmed per specification 3.10.A.5.b.

LIMITING CONDITIONS FOR OPERATION

3.10.A.6

A maximum of two non-adjacent control rods may simultaneously be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance without removing the fuel from the cells 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 refueling interlocks shall be operable.

SURVEILLANCE REQUIREMENTS

4.10.A.6

Prior to performing control rod or control rod drive maintenance on two control cells simultaneously without removing the fuel from the cells, two SRO's shall verify that the requirements of specification 3.10.A.6 are satisfied.

LIMITING CONDITIONS FOR OPERATION

3.10.A Refueling Interlocks

6. (Continued)

b. All directional control valves for remaining control rods shall be disarmed electrically except as specified in 3.10.A.7 and sufficient margin to criticality shall be demonstrated.

c. The two maintenance cells 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.B.

7. 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

SURVEILLANCE REQUIREMENTS

4.10.A Refueling Interlocks

7. With the mode selection switch in the refuel or shutdown mode, no more than one control rod may be withdrawn without first removing fuel from the cell except as specified in 4.10.A.6. Any number of rods may be withdrawn once verified by two licensed operators that the fuel has been removed from each cell.

rods and the refueling platform provide redundant methods of preventing inadvertent criticality even after procedural violations. The interlocks on hoists provide yet another method of avoiding inadvertent criticality.

Fuel handling is normally conducted with the fuel grapple hoist. The total load on this hoist when the interlock is required consists of the weight of the fuel grapple and the fuel assembly. This total is approximately 1,500 lbs, in comparison to the load-trip setting of 1,000 lbs. Provisions have also been made to allow fuel handling with either of the three auxiliary hoists and still maintain the refueling interlocks. The 400-lb load-trip setting on these hoists is adequate to trip the interlock when one of the more than 600-lb fuel bundles is being handled.

During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time without removing fuel from the cells. 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 requirement that an adequate shutdown margin be demonstrated and that all remaining control rods have there directional control valves electrically disarmed ensures that inadvertent criticality cannot occur during this maintenance. The adequacy of the shutdown margin is verified by demonstrating that at least 0.38% Δk shutdown margin is available. Disarming the directional control valves does not inhibit control rod scram capability.

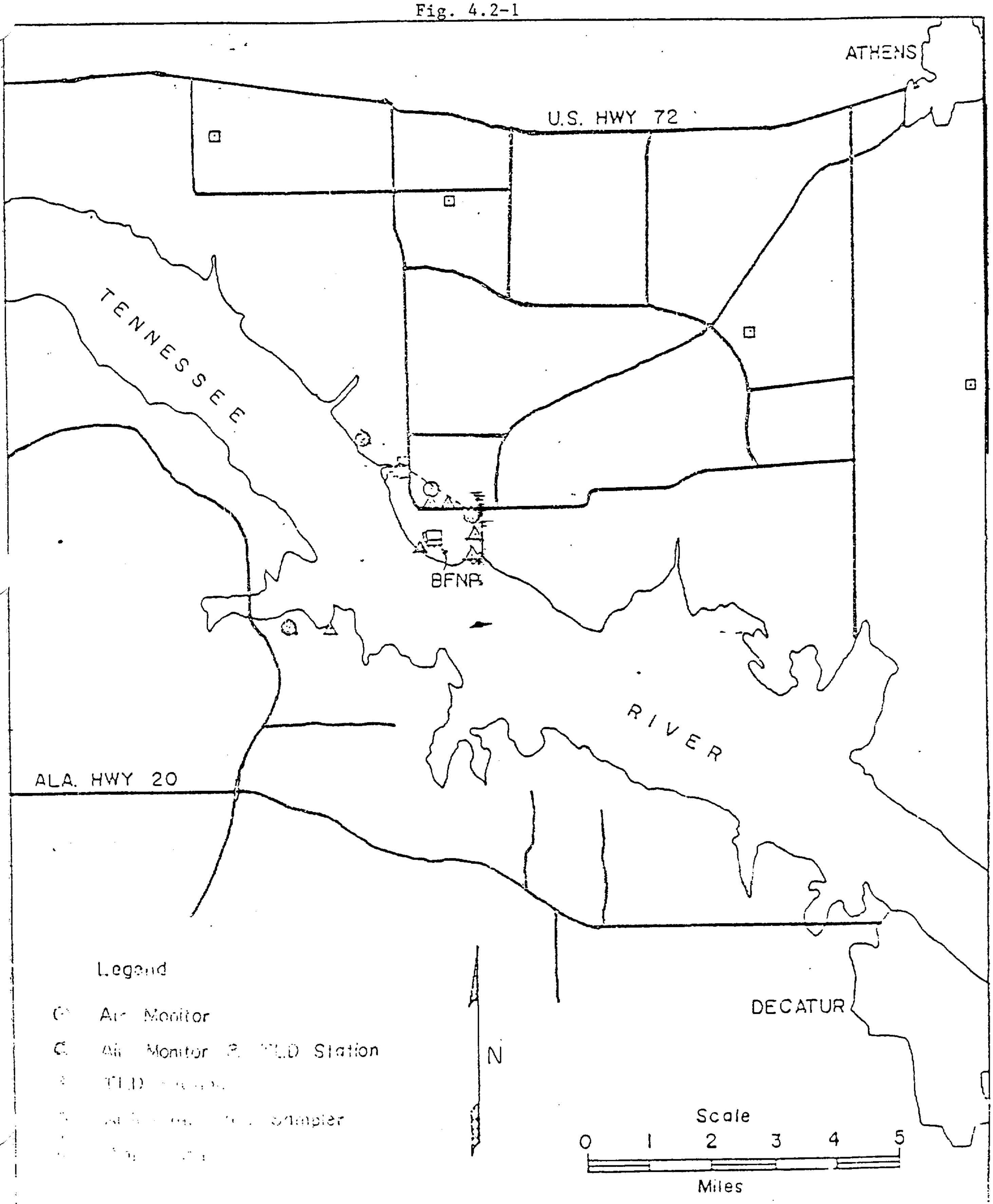
Specification 3.10.A.7 allows unloading of a significant portion of the reactor core. This operation is performed with the mode switch in the "refuel" position to provide the refueling interlocks normally available during refueling operations. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed ensures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod provides primary reactivity control for the fuel assemblies in the cell associated with that control rod.

Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core. The requirements for SRM operability during these core alterations assure sufficient core monitoring.

LOCAL MONITORING STATIONS

BROWNS FERRY NUCLEAR PLANT

Fig. 4.2-1



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 114 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 82 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3

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

1.0 Introduction

By letter dated April 30, 1982 (TVA BFNP TS 173) and supplemented by letter dated June 10, 1982, the Tennessee Valley Authority (the licensee or TVA) requested amendments to Facility Operating License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3. The proposed amendments would revise the Technical Specifications appended to the above Facility Operating Licenses to:
1) revise the operability requirements for the reactor water cleanup (RWCU) system isolation instrumentation, 2) revise the operability requirements for the residual heat removal service water (RHRSW) pumps, 3) revise the suppression chamber water level datum for HPCI suction switchover, 4) delete surveillance requirements for RWCU system compartment temperature detectors, 5) clarify operability and surveillance requirements on the residual heat removal (RHR) pumps, 6) revise the bases for drywall-to-torus leak rate testing, 7) correct a typographical error, 8) revise the requirements on control rod drive maintenance when fuel is present around the rods, 9) for Unit 2 only, revise the surveillance requirements on standby coolant supply pump operability, and 10) revise raw milk sampling requirements.

2.0 Evaluation

2.1 RWCU System High Temperature Sensor (Units 1,2,3)

Technical Specification Table 3.2.A requires that the reactor water cleanup system floor drain high temperature instrumentation be operable with two channels per trip system. A proposed change would expand this requirement to specify that each trip system requires two independent channels from each of the two floor drain locations. (This change is consistent with the as-built facility and is not the result of a modification). This change provides clarification only and is acceptable.

2.2 RHRSW Pump Timers (Units 1,2,3)

The licensee has requested changes to the residual heat removal service water (RHRSW) pump timer assignements to each Unit. Pump timers A1, B3, C1, and D3 would be assigned to Units 1 and 2 replacing timers A1, B3, C1, and D1. Timers A3, B1, C3, and D1 would be assigned to Unit 3, replacing A1, B3, C1, and D3. The revised assignments are consistent with FSAR figure 10.9.3 control logic requirements and are acceptable.

2.3 HPCI Suction Switchover Setpoint (Units 1,2,3)

Table 3.2.B of the Technical Specifications specifies the high pressure coolant injection (HPCI) suction trip level setting as "7" above normal water level". A change proposed by the licensee would revise the trip level setting to "7" above instrument zero". This would provide a fixed datum, making the Trip level independent of future changes to the normal water level. This change in itself would not result in a change to the setpoint and is therefore acceptable.

2.4 Editorial Correction (Units 1,2,3)

Note 9 to Table 3.2.B of the Technical Specifications refers to paragraph "3.5.I" for a list of pressures to be maintained by the head tank. The licensee has requested that Note 9 be changed to reference "3.5.H"; this change would correct an editorial error and is therefore acceptable.

2.5 RWCU Instrumentation Surveillance (Units 1,2,3)

The licensee has requested changes to Technical Specification Table 4.2.A to delete surveillance requirements for the reactor water cleanup space temperature detectors (RTDs). The RWCU space temperature RTD channels are not part of the primary containment isolation system and have no related safety limit or limiting condition for operation. (Separate channels using temperature switches are provided for RWCU floor drain/space high temperature isolation). Based on consistency with 10 CFR 50.36(c), and NUREG-0123 BWR Standard Technical Specifications this change is acceptable.

2.6 RHR System Operability (Units 1,2,3)

The RHR system for each unit consists of four loops (A,B,C, & D). Each loop consists of a pump, heat exchanger, piping path, and associated diesel generator. If two RHR pumps or associated heat exchangers are inoperable, Specification 3.5.B.6 permits operation for seven days if the remaining two loops are operable. Specification 3.5.B.6 does not specifically include a requirement for operability of the diesel generators associated with the remaining trains. However, such a requirement is intended as evidenced by the associated surveillance specification which includes the diesel generators in the surveillance tests. A change proposed by TVA would revise 3.5.B.6 to specifically include a requirement for operability of the associated diesel generators serving the operable redundant RHR trains. This change would revise the limiting condition for operation, to be consistent with its associated surveillance requirement and with Specification 1.0.E (definition of "operability"). This change is therefore acceptable.

2.7 Drywell to Suppression Chamber Leakage (Units 1,2,3)

Technical Specification 4.7.A.4.d requires a periodic test to determine if drywell to suppression chamber leakage is within a limit of 0.14 pounds per second of air when the pressure differential is 1 psi. The bases for 4.7.A.4.d states that 0.14 pps of air corresponds to a 0.25 inches of water per minute rate-of-change of suppression chamber pressure. Based on a suppression chamber air volume of 119,000 cubic feet, the correct value for the rate-of-change is 0.38 inches of water per minute. TVA has proposed that the bases for 4.7.A.4.d be revised accordingly. This change would correct the error and is acceptable.

2.8 Standby Coolant Supply (Unit 2)

Standby coolant supply connection and RHR crossties are provided to maintain a long-term reactor core and primary containment cooling capability independent of primary containment integrity or operability of the Residual Heat Removal System associated with a given unit. The standby coolant supply connection and RHR crossties provide added long-term redundancy to the other emergency core and containment cooling systems, and are designed to accommodate certain situations which could jeopardize the functioning of these systems. By proper valve alignment, the network created by the standby coolant supply connection and RHR crossties permits the D2 (or D1) RHR service water pump and header to supply raw water directly to the reactor core of Units 1 or 2 as the reactor pressure approaches 50 psig. The service water pump and header can also be valved to supply raw water to the drywell or suppression chamber of either unit. In a similar fashion, the B2 (or B1) RHR service water pump and header can supply raw water to the reactor core of Units 2 or 3 or into the respective suppression chambers.

Technical Specification 4.5.C.4 for each Browns Ferry unit specifies:

> "When it is determined that one of the RHRSW pumps supplying standby coolant is inoperable at a time when operability is required, the

operable RHRSW pump on the same header and its associated diesel generator and the RHR heat exchanger header and associated essential control valves shall be demonstrated to be operable immediately and every 15 days thereafter."

Because Unit 2 has four pumps available, (by virtue of crossties to both the B&D headers) whereas Units 1 and 3 only have two, this requirement sometimes requires unnecessary diesel testing when Unit 2 is operating and Unit 1 or 3 is in an outage. (The Technical Specifications for Unit 2 require additional testing upon loss of one of four pumps, whereas the Technical Specifications for Units 1 and 3 require additional testing upon loss of one of two pumps). The licensee has requested a change to the Unit 2 Technical Specification to require testing of the operable RHRSW pump heat exchanger, control valves, and associated diesel generator when it is determined that three pumps are inoperable. This will provide compatibility with Units 1 and 3 in that for each unit additional testing will be required when only one pump is available for supplying standby coolant. This change is therefore acceptable based on consistency with Units 1 and 3.

2.9 Control Rod Drive Maintenance (Units 1,2,3)

Tennessee Valley Authority has requested changes to the Technical Specifications that would alter the requirements for performing maintenance on control rods without removing the fuel assemblies surrounding them. The staff has reviewed the proposed changes and prepared the following evaluation.

The current Technical Specification 3.10.A permits maintenance on a control rod without removal of fuel from around the rod if analysis has demonstrated that the core will be subcritical by at least 0.38 percent delta k/k with that rod and the strongest additional rod completely withdrawn. Alternatively, if all other control rods are fully inserted and have their directional control valves electrically disarmed, it is not necessary to assume the second rod to be withdrawn. For two rods to be withdrawn similar conditions prevail except that the margin to subcriticality is not specified. Any number of rods may be withdrawn if the fuel surrounding each rod is first removed from the core. In each circumstance described above the mode switch must be locked in the refuel position and be operable. Withdrawn rods may be bypassed in order to satisfy the "one-rod-out" interlock. No fuel may be loaded into the core unless all rods are fully inserted.

Two factors motivate the request for a change in the Technical Specification. Except for specific times in the cycle (beginning of cycle, e.g.) the identity of the strongest rod and the shutdown margin are not known. Extensive calculation are required to obtain this knowledge. In addition, the requirement for disarming all remaining directional control valves during single rod maintenance (the most often encountered situation) requires a time consuming procedure which results in added personnel exposure and wear on the directional control valve electrical connectors.

The revised Technical Specification deletes the requirement to obtain the "strongest-rod-out" shutdown margin and replaces it with a requirement to demonstrate margin to criticality for the situation in which maintenance is to be performed on two rods. For performing maintenance on a single control rod only the immediately surrounding rods are required to have their directional control valves disarmed. Surveillance on the refueling interlocks must also be performed prior to withdrawing the rod for maintenance. For maintenance on two non-adjacent control rods without removing fuel from the cells all other control rods must have their directional control valves disarmed electrically. The two maintenance cells must be separated by more than two control cells in any direction. As before, any number of control rods may be removed for maintenance if the fuel is removed from each cell prior to the removal of the rod.

Withdrawal of a single control rod from the core will not result in criticality since sufficient shutdown margin is required to preclude its occurrence. The single rod withdrawn interlock prevents the withdrawal of a second rod. Performance of surveillance on the interlock prior to withdrawing the rod assures that the interlock is operable. Additional assurance against withdrawal of a high worth adjacent rod is provided by electrically disarming three directional control valves on these rods. We conclude that sufficient control exists to preclude inadvertent criticality when maintenance is being performed on a single rod.

When maintenance is being done on two rods without fuel removal additional precautions are required. Prior to withdrawal of the second rod a determination is made that criticality will not occur when the rod is withdrawn. Low worth of the second rod is assured by the requirement that it be separated from the first by more than two control cells in each direction. Additional assurance that a third rod cannot be withdrawn is provided by the requirement that all remaining rods have their directional control valves electrically disarmed. An additional requirement that the source range monitor be operable provide assurance that the subcriticality of the core may be monitored at all times. We conclude that sufficient controls exist to preclude inadvertent criticality when maintenance is being performed on two rods simultaneously.

Based on our review, which is described above, we conclude that the proposed revisions to Technical Specification 3.10.A of Browns Ferry Nuclear Plant, Units 1, 2, and 3 are acceptable.

2.10 Raw Milk Sampling (Units 1,2,3)

Environmental Technical Specifications (Appendix B) Paragraph 4.2.3.b requires that "milk shall be collected monthly when animals are off pasture, from at least four farms in the vicinity of the plant and analyzed as indicated in Table 4.2-1 and figure 4.2-1." Figure 4.2-1 depicts five dairy farms.

The licensee has requested a change to the figure deleting two of the dairy farms and adding a new one, stating that the two deleted sampling points no longer have milk producing animals. This change is acceptable based on the need to have four locations with milk producing animals.

3.0 Environmental Considerations

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

4.0 Conclusion

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

Principal Contributor: W. Brooks, W. Long, R. Clark

Dated: October 16, 1984