June 22, 1978

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

Distribution
Docket
ORB #3
Local PDR
NRC PDR
GLear
TIppolito
SSheppard
RClark
VStello
BGrimes
Attorney, OELD
OI&E (5)
BJones (12)
BScharf (10)
JMcGough
DEisenhut
ACRS (16)
OPA (CMiles)
DRoss
TBAbernathy
JRBuchanan
RDiggs

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

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 38, 36 and 12 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. These amendments consist of changes to the Technical Specifications in response to your request of May 11, 1978 (TVA BFNP TS 108).

The amendments change the Technical Specifications to permit you to delete the requirements for the oxygen sensors as used in the containment atmosphere monitoring system. With your concurrence, we have modified your submittal to add additional surveillance requirements in Section 4.7.A.5.

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

Sincerely,

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

Enclosures:
1. Amendment No. 38 to DPR-33
2. Amendment No. 36 to DPR-52
3. Amendment No. 12 to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures: See page 2

*SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE	ORB #3	ORB #3	OELD	ORB #3		
SURNAME	SSheppard*	*RClark	*	TIppolito		
DATE	6/8/78	6/8/78	6/21/78	6/23/78		

NRC FORM 318 (9-76) NRCM 0240

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

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

Gentlemen:

The Commission has issued the enclosed Amendments Nos. , and to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. These amendments consist of changes to the Technical Specifications in response to your request of May 11, 1978 (TVA BFNP TS 108).

The amendments change the Technical Specifications to permit you to delete the requirements for the oxygen sensors as used in the containment atmosphere monitoring system. With your concurrence, we have modified your submittal to add additional surveillance requirements in Section 4.7.A.5.

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

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 38 to DPR-33
2. Amendment No. 36 to DPR-52
3. Amendment No. 12 to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures: See page 2

per changes made on SER copy

OFFICE	ORB #3	ORB #3	OELD	ORB #3		
SURNAME	SSheppard:mjf	RClark		GLear		
DATE	6/8/78	6/8/78	6/21/78	6/ /78		

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

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
License No. DPR-33

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

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

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 38, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 22, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 38

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

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

   79/80
   234/235
   248/249
   268/269
   270/271

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

TABLE 3.2.F

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	H_2M - 76 - 37 H_2M - 76 - 39	Drywell H_2 Concentration	0.1 - 20%	(1)
1	H_2M - 76 - 38	Suppression Chamber H_2 Concentration	0.1 - 20%	(1)

NOTES FOR TABLE 3.2.F

(1) From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

(2) From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made operable.

(3) If the requirements of notes (1) and (2) cannot be met, either the requirements of 3.5.H shall be complied with or an orderly shutdown shall be initiated and the reactor shall be in a Cold Condition within 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

3.7.A Primary Containment

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be operable at all times when primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be 0.5 psid.

b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate primary containment integrity.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be operable and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and c, below.

b. One drywell-suppression chamber vacuum breaker may be non-fully closed so long as it is determined to be not more than 3" open as indicated by the position lights.

4.7.A Primary Containment

the reactor shall be placed in cold shutdown and the above inspection shall be performed before the reactor is started up.

3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised and the associated instrumentation including setpoint shall be functionally tested for proper operation each three months.

b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. Each drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle every month.

b. When it is determined that two vacuum breakers are inoperable for opening at a time when operability is requir all other vacuum breaker

3.7.A Primary Containment

c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.

d. If specifications 3.7.A.4.a, .b, or .c cannot be met, the unit shall be placed in a cold shutdown condition in an orderly manner within 24 hours.

5. Oxygen Concentration

a. After completion of the fire-related startup retesting program, containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by weight and maintained in this condition. De-inerting may commence 24 hours prior to a shutdown.

c. If specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

4.7.A Primary Containment

valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

c. Once each operating cycle each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches.

d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.14 lb/sec of primary containment atmosphere with 1 psi differential.

5. Oxygen Concentration

a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

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 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.

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

H. Containment Atmosphere Monitoring (CAM System - H_2 Analyzer

1. Whenever the reactor is not in cold shutdown, two gas analyzer systems shall be operable for monitoring the drywell.

2. Whenever the reactor is not in cold shutdown, one gas analyzer system shall be operable for monitoring the torus.

3. If specification 3.7.H.1 cannot be met, but one system remains operable, the reactor may be operated for a period of 30 days. If both systems are inoperable, the reactor should be placed in shutdown condition within 24 hours.

4. If specification 3.7.H.2 cannot be met, the reactor may be operated for a period of 30 days.

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

H. Containment Atmosphere Monitoring (CAM) System - H_2 Analyzer

1. Once per month perform a channel calibration using standard gas samples containing a nominal:

   Three volume percent hydrogen, balance nitrogen

3.7.A & 4.7.A **Primary Containment**

The integrity of the primary containment and operation of the core standby cooling system in combination, limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep offsite doses well below 10 CFR 100 limits.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1,035 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 49 psig which is below the maximum of 62 psig. Maximum water volume of 135,000 ft^3 results in a downcomer submergence of 5'2-3/32" and the minimum volume of 123,000 ft^3 results in submergence approximately 12 inches less. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humbolt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

BASES

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operatibility. Under full power operation conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a peak long term water temperature of 170°F which is sufficient for complete condensation. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is not dependency on containment overpressure.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressuirzed in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Limiting suppression pool temperature to 105°F during RCIC, HPCI, or relief valve operation when decay heat and stored energy is removed from the primary system by discharging reactor steam directly to the suppression chamber assures adequate margin for controlled blowdown anytime during RCIC operation and assures margin for complete condensation of steam from the design basis loss-of-coolant accident.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

If a loss-of-coolant accident were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62 psig code permissible pressures even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212°F provides additional margin above that available at 330°F.

Inerting

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent or so) reaction of the zirconium and steam during a loss-of-coolant accident could lead to the liberation of hydrogen combined with an air atmosphere to result in a flammable concentration in the containment. If a sufficient amount of hydrogen is generated and oxygen is available in stoichiometric quantities the subsequent ignition of the hydrogen in rapid recombination rate could lead to failure of the containment to maintain a low leakage integrity. The 4.4% hydrogen concentration minimizes the possibility of hydrogen combustion following a loss-of-coolant accident.

BASES

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

To ensure that the hydrogen concentration is maintained less than 4% following an accident, liquid nitrogen is maintained on-site for containment atmosphere dilution. About 2260 gallons would be sufficient as a 7-day supply, and replenishment facilities can deliver liquid nitrogen to the site within one day; therefore, a requirement of 2500 gallons is conservative. Following a loss of coolant accident the Containment Air Monitoring (CAM) System continuously monitors the hydrogen concentration of the containment volume. Two independent systems (a system consists of one hydrogen sensing circuit) are installed in the drywell and one system is installed in the torus. Each sensor and associated circuit is periodically checked by a calibration gas to verify operation. Failure of a drywell system does not reduce the ability to monitor system atmosphere as a second independent and redundant system will still be operable.

In terms of separability, redundancy for a failure of the torus system is based upon at least one operable drywell system. The drywell hydrogen concentration can be used to limit the torus hydrogen concentration during post LOCA conditions. Post LOCA calculations show that the CAD system initiated within two hours at a flow rate of 100 scfm will limit the peak drywell and wetwell hydrogen concentration to 3.6% (at 4 hours) and 3.8% (at 32 hours), respectively. This is based upon purge initiation after 20 hours at a flow rate of 100 scfm to maintain containment pressure below 30 psig. Thus, peak torus hydrogen concentration can be controlled below 4.0 percent using either the direct torus hydrogen monitoring system or the drywell hydrogen monitoring system with appropriate conservatism ($\leq$ 3.8%), as a guide for CAD/Purge operations.

Vacuum Relief

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and reactor building so that the structural integrity of the containment is maintained. The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 2 psig; the external design pressure. One reactor building vacuum breaker may be out of service for repairs for a period of seven days. If repairs cannot be completed within seven days, the reactor coolant system is brought to a condition where vacuum relief is no longer required.

When a drywell-suppression chamber vacuum breaker valve is exercised through an opening-closing cycle the position indicating lights in the control room are designed to function as specified below:

Initial and Final Condition	Check - On Green - On Red - Off	(Fully closed)
Opening Cycle	Check - Off Green - Off Red - On	(Cracked open) (> 80^{o} Open) (> 3^{o} Open)
Closing Cycle	Check - On Green - On Red - Off	(Fully Closed) (< 30^{o} Open) < 3^{o} Open)

The valve position indicating lights consist of one check light on the check light panel which confirms full closure, one green light next to the hand switch which confirms 80^{o} of full opening and one red light next to the hand switch which confirms "near closure" (within 3" of full closure). Each light is on a separate switch. If the check light circuit is operable when the valve is exercised by its air operator there exists a confirmation that the valve will fully close. If the red light circuit is operable, there exists a confirmation that the valve will at least "nearly close" (within 3^{o} of full closure). The green light circuit confirms the valve will fully open. If none of the lights change indication during the cycle, the air operator must be inoperable or the valve disc is stuck. For this case, a check light on and red light off confirms the disc is in a nearly closed position even if one of the indications is in error. Although the valve may be inoperable for full closure, it does not constitute a safety threat.

If the red light circuit alone is inoperable, the valve shall still be considered fully operable. If the green and red or the green light circuit along is inoperable the valve shall be sonsidered inoperable for

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 36
License No. DPR-52

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

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

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 36 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 22, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 36

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

Remove the following pages and replace with identically numbered pages:

79/80
235/236
249/250
269/270

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

TABLE 3.2.F

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	H_2M - 76 - 37 H_2M - 76 - 39	Drywell H_2 Concentration	0.1 - 20%	(1)
1	H_2M - 76 - 38	Suppression Chamber H_2 Concentration	0.1 - 20%	(1)

NOTES FOR TABLE 3 F

(1) From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

(2) From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made operable.

(3) If the requirements of notes (1) and (2) cannot be met, either the requirements of 3.5.H shall be complied with or an orderly shutdown shall be initiated and the reactor shall be in a Cold Condition within 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

3.7.A Primary Containment

c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.

d. If specifications 3.7.A.4.a, .b, or .c cannot be met, the unit shall be placed in a cold shutdown condition in an orderly manner within 24 hours.

5. Oxygen Concentration

a. After completion of the fire-related startup retesting program, containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by weight and maintained in this condition. De-inerting may commence 24 hours prior to a shutdown.

c. If specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

4.7.A Primary Containment

valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

c. Once each operating cycle each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches.

d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.14 lb/sec of primary containment atmosphere with 1 psi differential.

5. Oxygen Concentration

a. The primary containment oxygen concentration shall be measures and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

3.7 CONTAINMENT SYSTEMS

B. Standby Gas Treatment System

1. Except as specified in Specification 3.7.B.3 below, all three trains of the standby gas treatment system and the diesel generators required for operation of such trains shall be operable at all times when secondary containment integrity is required.

4.7 CONTAINMENT SYSTEMS

B. Standby Gas Treatment System

1. At least once per year, the following conditions shall be demonstrated.

   a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at a flow of 9000 cfm (± 10%).

   b. The inlet heaters on each circuit are capable of an output of at least 40 kW when tested in accordance with ANSI N510-1975.

   c. Air distribution is uniform within 20% across HEPA filters and charcoal adsorbers.

7 CONTAINMENT SYSTEMS

H. Containment Atmosphere Monitoring (CAM System - H_2 Analyzer

1. Whenever the reactor is not in cold shutdown, two gas analyzer systems shall be operable for monitoring the drywell.

2. Whenever the reactor is not in cold shutdown, one gas analyzer system (one hydrogen sensing circuit per system) shall be operable for monitoring the torus.

3. If specification 3.7.H.1 cannot be met, but one system remains operable, the reactor may be operated for a period of 30 days. If both systems are inoperable, the reactor should be placed in shutdown condition within 24 hours.

4. If specification 3.7.H.2 cannot be met, the reactor may be operated for a period of 30 days.

4.7 CONTAINMENT SYSTEMS

H. Containment Atmosphere Monitoring (CAM) System - H_2 Analyzer

1. Once per month perform a channel calibration using standard gas samples containing a nominal Three volume percent hydrogen balance nitrogen

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-74-57, 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

BASES

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operatibility. Under full power operation conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a peak long term water temperature of 170°F which is sufficient for complete condensation. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is not dependency on containment overpressure.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressuirzed in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Limiting suppression pool temperature to 105°F during RCIC, HPCI, or relief valve operation when decay heat and stored energy is removed from the primary system by discharging reactor steam directly to the suppression chamber assures adequate margin for controlled blowdown anytime during RCIC operation and assures margin for complete condensation of steam from the design basis loss-of-coolant accident.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

If a loss-of-coolant accident were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62 psig code permissible pressures even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212°F provides additional margin above that available at 330°F.

Inerting

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent or so) reaction of the zirconium and steam during a loss-of-coolant accident could lead to the liberation of hydrogen combined with an air atmosphere to result in a flammable concentration in the containment. If a sufficient amount of hydrogen is generated and oxygen is available in stoichiometric quantities the subsequent ignition of the hydrogen in rapid recombination rate could lead to failure of the containment to maintain a low leakage integrity. The $<$ 4% hydrogen concentration minimizes the possibility of hydrogen combustion following a loss-of-coolant accident.

BASES

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

To ensure that the hydrogen concentration is maintained less than 4% following an accident, liquid nitrogen is maintained on-site for containment atmosphere dilution. About 2260 gallons would be sufficient as a 7-day supply, and replenishment facilities can deliver liquid nitrogen to the site within one day; therefore, a requirement of 2500 gallons is conservative. Following a loss of coolant accident the Containment Air Monitoring (CAM) System continuously monitors the hydrogen concentration of the containment volume. Two independent systems (a system consists of one hydrogen sensing circuit) are installed in the drywell and one system is installed in the torus. Each sensor and associated circuit is periodically checked by a calibration gas to verify operation. Failure of a drywell system does not reduce the ability to monitor system atmosphere as a second independent and redundant system will still be operable.

In terms of separability, redundancy for a failure of the torus system is based upon at least one operable drywell system. The drywell hydrogen concentration can be used to limit the torus hydrogen concentration during post LOCA conditions. Post LOCA calculations show that the CAD system initiated within two hours at a flow rate of 100 scfm will limit the peak drywell and wetwell hydrogen concentration to 3.6% (at 4 hours) and 3.8% at 32 hours), respectively. This is based upon purge initiation after 20 hours at a flow rate of 100 scfm to maintain containment pressure below 30 psig. Thus, peak torus hydrogen concentration can be controlled below 4.0 percent using either the direct torus hydrogen monitoring system or the drywell hydrogen monitoring system with appropriate conservatism ($\leq$3.8%), as a guide for CAD/Purge operations.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 12
License No. DPR-68

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

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

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 12 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 22, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 12

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

Remove the following pages and replace with identically numbered pages:

82
245
261
286A
287

Marginal lines indicate revised area.

TABLE 3.2.F
SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	H_2M - 76 - 37 H_2M - 76 - 39	Drywell H_2 Concentration	0.1 - 20%	(1)
1	H_2M - 76 - 38	Suppression Chamber H_2 Concentration	0.1 - 20%	(1)

3.7 CONTAINMENT SYSTEMS

d. If specifications 3.7.A.4.a, .b, or .c, cannot be met, the unit shall be placed in a cold shutdown condition in an orderly manner within 24 hours.

5. Oxygen Concentration

a. After completion of the 300-hour warranty run, containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by weight and maintained in this condition. De-inerting may commence 24 hours prior to a shutdown.

4.7 CONTAINMENT SYSTEMS

d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.14 lb/sec of primary containment atmosphere with 1 psi differential.

5. Oxygen Concentration

a. The primary containment oxygen concentration shall be measured and recorded daily. The exygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

3.7 CONTAINMENT SYSTEMS

H. Containment Atmosphere Monitoring (CAM System - H_2 Analyzer

1. Whenever the reactor is not in cold shutdown, two gas analyzer systems shall be operable for monitoring the drywell.

2. Whenever the reactor is not in cold shutdown, one gas analyzer system shall be operable for monitoring the torus.

3. If specification 3.7.H.1 cannot be met, but one system remains operable, the reactor may be operated for a period of 30 days. If both systems are inoperable, the reactor should be placed in shutdown condition within 24 hours.

4. If specification 3.7.H.2 cannot be met, the reactor may be operated for a period of 30 days.

4.7 CONTAINMENT SYSTEMS

H. Containment Atmosphere Monitoring (CAM) System - H_2 Analyzer

1. Once per month perform a channel calibration using standard gas samples containing a nominal three volume precent hydrogen, balance nitrogen.

Inerting

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent or so) reaction of the zirconium and steam during a loss-of-coolant accident could lead to the liberation of hydrogen combined with an air atmosphere to result in a flammable concentration in the containment. If a sufficient amount of hydrogen is generated and oxygen is available in stoichiometric quantities, the subsequent ignition of the hydrogen in rapid recombination rate could lead to failure of the containment to maintain low leakage integrity. The <4% hydrogen concentration minimizes the possibility of hydrogen combustion following a loss-of-coolant accident.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

To ensure that the hydrogen concentration is maintained less than 4% following an accident, liquid nitrogen is maintained on-site for containment atmosphere dilution. About 2260 gallons would be sufficient as a 7-day supply, and replenishment facilities can deliver liquid nitrogen to the site within one day; therefore, a requirement of 2500 gallons is conservative.

Following a loss-of-coolant accident the Containment Air Monitoring (CAM) System continuously monitors the hydrogen concentration of the containment volume. Two independent systems (a system consists of one hydrogen sensing circuit) are installed in the drywell and one system is installed in the torus. Each sensor and associated circuit is periodically checked by a calibration gas to verify operation.

Failure of a drywell system does not reduce the ability to monitor system atmosphere as a second independent and redundant system will still be operable.

In terms of separability, redundancy for a failure of the torus system is based upon at least one operable drywell system. The drywell hydrogen concentration can be used to limit the torus hydrogen concentration during post LOCA conditions. Post LOCA calculations show that the CAD system within two hours at a flow rate of 100 scfm will limit the peak drywell and wetwell hydrogen concentration to 3.9% (at 3 hours) and 3.9% (at 32 hours), respectively. This is based upon purge initiation after 20 hours at a flow rate of 100 scfm to maintain containment pressure below 30 psig. Thus, peak torus hydrogen concentration can be controlled below 4.0 percent using either the direct torus hydrogen monitoring system or the drywell hydrogen monitoring system with appropriate conservatism (≤ 3.9%), as a guide for CAD/Purge operations.

Vacuum Relief

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and reactor building so that the structural integrity of the containment is maintained. The vacuum relief system from the pressure suppression chamber to reactor buiding consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 2 psig; the external design pressure. One reactor building vacuum breaker may be out of service for repairs for a period of seven days. If repairs cannot be completed within seven days, the reactor coolant system is brought to a condition where vacuum relief is no longer required.

When a drywell-suppression chamber vacuum breaker valve is exercised through an opening-closing cycle the position indicating lights in the control room are designed to function as specified below:

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 36 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 12 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

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

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

1.0 Introduction

By letter dated May 11, 1978, the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. The proposed amendments and revised Technical Specifications would delete the surveillance requirements for the containment oxygen monitoring instrumentation on the basis that this instrumentation is not required for post-accident monitoring.

The Technical Specifications for the Browns Ferry Plant currently require that specific instrumentation be used to monitor the oxygen concentration in the containment during normal plant operation, and identify the surveillance requirements for that instrumentation. The primary containment is operated with an oxygen-deficient (i.e., inerted) atmosphere as part of those measures for combustible gas control following a postulated loss-of-coolant accident (LOCA). Based on its reassessment of the design of the combustible gas control system, the licensee has requested that the requirements for specific oxygen monitoring instrumentation be deleted from the plant's Technical Specifications. Those Technical Specification requirements which limit the maximum oxygen concentration in the containment during normal operation would be retained (i.e., the containment will continue to be inerted during normal operation).

The licensee has proposed these changes to allow the use of alternate measurement techniques to establish the containment oxygen concentration. Past experience has shown that the existing oxygen monitoring subsystems have not demonstrated sufficiently reliable performance.

2.0 Discussion

Following a postulated LOCA, hydrogen is generated from a reaction between the Zircaloy fuel cladding and the primary coolant (metal-water reaction), and both hydrogen and oxygen are generated as a result of radiolysis of the primary coolant and the water in the suppression pool. When a sufficient quantity of hydrogen has been generated in an oxygenated atmosphere, a combustible mixture of gasses is formed. In order to protect the containment structure and engineered safety feature systems from the potential consequences of combustion, a Combustible Gas Control System (CGCS) is incorporated into the plant design. (reference 1). For Browns Ferry, the CGCS consists of inerting the containment atmosphere with nitrogen during normal operation and diluting the containment atmosphere with nitrogen following a postulated LOCA to maintain the hydrogen concentration below the flammability limit.

The operation of the CGCS is now to be based on the measured hydrogen concentration in the containment following a postulated accident. The operation of the CGCS was previously based on the monitored oxygen concentration. Control of either hydrogen or oxygen will assure that a combustible mixture of gases will not be formed. The initially inerted containment atmosphere assures that the hydrogen released from any metal-water reaction will not exceed a combustible concentration, and it provides a longer period of time to the point at which the nitrogen dilution system is manually initiated.

Two inter-related systems, Containment Atmosphere Monitoring (CAM) and Containment Atmosphere Dilution (CAD) systems, are provided in the Browns Ferry Units 1-3 plant to monitor the concentration of oxygen and hydrogen and thereby prevent the creation of a combustible gas mixture in the primary containment following a LOCA. In order to ensure that a combustible mixture is not created, either the hydrogen concentration must remain below 4% by volume or the oxygen concentration must remain below 5% by volume. As discussed above, the only significant sources of hydrogen and oxygen buildup in the containment following a LOCA are hydrogen evolution from metal-water reactions and the radiolysis of water. Since the buildup of hydrogen following a LOCA will occur more rapidly than the oxygen buildup due to the early occurrence of a metal-water reaction, control based only on hydrogen monitoring is adequate if the H_2 concentration is demonstrated by analysis to be less than 4%. The hydrogen concentration is controlled and the containment pressure is maintained below 30 psig by operation of the CAD system and subsequent purge. The offsite dose following the purgung must be less than or equal to the guidelines contained in 10 CFR 100.

3.0 Evaluation

As discussed above, the licensee submitted an evaluation to demonstrate that the hydrogen sensors could be used exclusively to monitor containment atmosphere for a combustible mixture following a postulated LOCA and that the present requirement for both hydrogen and oxygen sensors was unnecessary and redundant, provided other acceptable means of assuring that the oxygen concentration during normal operation can be demonstrated to be less than 4%. As additional support, the licensee also submitted calculations to demonstrate that the potential hydrogen concentration in containment would be below a combustible mixture following a postulated LOCA even if the containment was not inerted (i.e., if the atmosphere were not diluted with nitrogen as currently required by the Technical Specifications to reduce the oxygen concentration to less than 4%). The licensee did not request to delete the requirements for inerting and this subject is not covered by this safety evaluation. In addition, the licensee has not proposed to change the requirement in Section 4.7.A.5 of the Technical Specifications that the primary containment oxygen concentration be measured and recorded daily.

We find that the current requirements for surveillance testing of the oxygen monitoring instrumentation are not necessary. Therefore, the Technical Specifications have been modified to delete the specific requirements for oxygen instrumentation and to include the more general requirements for surveillance of and calibration by an appropriate method which the licensee would choose to establish the oxygen concentration during normal operation. The staff has proposed, and the licensee has agreed, to modify the surveillance requirements in Section 4.7.A.5 of the Technical Specifications to specify that the uncertainty associated with the method be predetermined and factored into the measurement of the oxygen concentration. The measurement will be taken daily as previously discussed. These requirements assure that the CGCS for the Browns Ferry Plant will be able to maintain the hydrogen concentration below a combustible level in the unlikely event of a LOCA. Accordingly, we find the changes to the Technical Specifications are acceptable.

4.0 Environmental Considerations

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

5.0 Conclusion

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

Dated: June 22, 1978

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 38 to Facility Operating License No. DPR-33, Amendment No. 36 to Facility Operating License No. DPR-52 and Amendment No. 12 to Facility Operating License No. DPR-68 issued to Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3, located in Limestone County, Alabama, The amendments are effective as of the date of issuance.

The amendments change the Technical Specifications to delete the requirements for the oxygen sensors used in the containment atmosphere monitoring system and augment the surveillance requirements associated with the daily oxygen analyses of primary containment.

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

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

For further details with respect to this action, see (1) the application for amendments dated May 11, 1978, (2) Amendment No. 38 to License No. DPR-33, Amendment No. 36 to License No. DPR-52, and Amendment No. 12 to License No. DPR-68, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Athens Public Library. South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 22nd day of June 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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