DECEMBER 8 1978

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

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:

In response to your request for license amendments dated November 5, 1976, as supplemented by letter dated October 18, 1978, the Commission has issued the enclosed Amendments Nos. 46, 42, and 19 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 incorporate provisions into the facility Technical Specifications which establish limiting conditions for operation and surveillance requirements for drywell to suppression chamber differential pressure control and suppression pool water level.

These requirements provide assurance that facility operation will be in accordance with the assumptions utilized in your facility's plant-unique analysis which was performed in conjunction with the Mark I Containment Short Term Program evaluation.

The enclosed license amendments reflect those changes to your original request for license amendments which have been agreed to in discussions with your staff. These changes have been made to provide consistent requirements for all Mark I containment facilities.

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

7901030253

Sincerely,

ccp

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

CP-1

*SEE PREVIOUS YELLOW FOR CONCURRENCES

Enclosures and ccs:
See next page

OFFICE	ORB #3	ORB #3	DOR	OELD	ORB #3
SURNAME	*SSheppard:	RClark:mjf	BGrimes		TIppolito
DATE	8/25/78	12/01/78	12/6/78	/ /78	12/8/78

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

Distribution
Docket
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BJones (4) 12
BScharf (10)
JMcGough
DEisenhut
ACRS (16)
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DRoss
TBAbernathy
JRBuchanan
RDiggs
CGrimes

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

Gentlemen:

In response to your request for license amendments dated November 5, 1976, as supplemented by a phone call of August 8, 1978 between Ron Rogers (TVA) and John Booske (NRC/DOR), the Commission has issued the enclosed Amendments Nos. , , and 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 incorporate provisions into the facility Technical Specifications which establish limiting conditions for operation and surveillance requirements for drywell to suppression chamber differential pressure control and suppression pool water level.

These requirements provide assurance that facility operation will be in accordance with the assumptions utilized in your facility's plant-unique analysis which was performed in conjunction with the Mark I Containment Short Term Program evaluation.

The enclosed license amendments reflect those changes to your original request for license amendments which have been agreed to in discussions with your staff. These changes have been made to provide consistent requirements for all Mark I containment facilities.

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

Sincerely,

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

Enclosures and ccs:
See next page

OFFICE ➤	ORB #3	ORB #3	DOR	OELD	ORB #3
SURNAME ➤	SSheppard:mjf	RClark	CGrimes		TIppolito
DATE ➤	8/25/78	8/ /78	8/ /78	/ /78	/ /78

Enclosures:
1. Amendments Nos. 46, 42 and 19 to License Nos. DPR-33, DPR-52 and DPR-68
2. Safety Evaluation
3. Notice

cc w/enclosures:

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U. S. Environmental Protection Agency
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ATTN: EIS Coordinator
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Atlanta, Georgia 30308

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108 Parkway Towers
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Nashville, Tennessee 37219

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. 46
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated November 5, 1976, as supplemented October 18, 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. 46, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

7901030257

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: December 8, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 46

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

79/80
105/106
226/227
228/229
Add page 235a
266/267
268
269/270

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)
2	PdI-64-137 PdI-64-138	Drywell to Suppression Chamber Differential pressure	Indicator 0 to 2 psid	(1) (2) (3)

Amendment No. ~~38~~, 46

NOTES FOR TABLE .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.

(5) If the requirements of notes (1) and (2) cannot be met, and if one of the indications cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a Cold Condition within 24 hours.

TABLE 4.2.F
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

	Instrument Channel	Calibration Frequency	Instrument Check
1)	Reactor Water Level	Once/6 months	Each Shift
2)	Reactor Pressure	Once/6 months	Each Shift
3)	Drywell Pressure	Once/6 months	Each Shift
4)	Drywell Temperature	Once/6 months	Each Shift
5)	Suppression Chamber Air Temperature	Once/6 months	Each Shift
6)	Suppression Chamber Water Temperature	Once/6 months	Each Shift
7)	Suppression Chamber Water Level	Once/6 months	Each Shift
8)	Control Rod Position	NA	Each Shift
9)	Neutron Monitoring	(2)	Each Shift
10)	Drywell Pressure (PS-64-67)	Once/6 months	NA
11)	Drywell Pressure (PS-64-58B)	Once/6 months	NA
12)	Drywell Temperature (TR-64-52)	Once/6 months	NA
13)	Timer (IS-64-67)	Once/6 months	NA
14)	CAD Tank Level	Once/6 months	Once/day
15)	Containment Atmosphere Monitors	Once/6 months	Once/day
16)	Drywell to Suppression Chamber Differential Pressure	Once /6 months	Each Shift

TABLE 4.2.G

SURVEILLANCE REQUIREMENTS FOR CONTROL ROOM ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration	Instrument Check
Control Room Air Supply Duct Radiation Monitors	(1)	once/3 months	once/day (8)
Control Room Isolation Logic	once/6 months	N/A	N/A
Simulated automatic actuation of control room isolation and emergency pressurization system	once/operating cycle	N/A	N/A

These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Ten percent or ten snubbers whichever is less, represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. Those snubbers designated in Table 3.6.H as being in high radiation areas or especially difficult to remove need not be selected for functional tests provided operability was previously verified.

Snubbers of rated capacity greater than 50,000 lb. are exempt from the functional testing requirements because of the impracticability of testing such large units.

REFERENCES

1. Report, H. R. Erickson, Bergen Paterson to K. R. Goller, NRC, October 7, 1974, Subject: Hydraulic Shock Sway Arrestors

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits except as specified in 3.7.A.2.

a. Minimum water level = -7" (differential pressure control >0 psid)

-8" (0 psid differential pressure control)

b. Maximum water level = -1"

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

1. Pressure Suppression Chamber

a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

3.7 CONTAINMENT SYSTEMS

c. With the suppression pool water temperature > 95°F initiate pool cooling and restore the temperature to < 95°F within 24 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

d. With the suppression pool water temperature > 105°F during testing of ECCS or relief valves, stop all testing, initiate pool cooling and follow the action in specification 3.7.A.1.c above.

e. With the suppression pool water temperature > 120°F following reactor isolation, depressurize to < 200 psig at normal cooldown rates.

f. With the suppression pool water temperature > 110°F during startup or power operation the reactor shall be scrammed.

4.7 CONTAINMENT SYSTEMS

3.7.A Primary Containment

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F. and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

4.7.A Primary Containment

2. Integrated Leak Rate Testing

a. Integrated leak rate tests (ILRT's) shall be performed to verify primary containment integrity. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a, does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design pressure, P_p.

b. Integrated leak rate tests may be performed at P_p or at a test pressure, P_t of not less than 25 psig provided the resultant leakage rate, L_t, does not exceed a preestablished fraction of L_a determined as follows:

Prior to initial operation, integrated leak rate tests must be performed at P_t and P_p with the lower pressure test performed first to establish the allowable leak rates (in percent per 24 hours). The leakage rates thus measured shall be identified as L_{tm} and L_{pm} respectively. L_t shall not exceed $L_a \frac{L_{tm}}{L_{pm}}$ for values

of $\frac{L_{tm}}{L_{pm}} \leq 0.7$.

3.7 CONTAINMENT SYSTEMS

6. Drywell-Suppression Chamber Differential Pressure

   a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.3 psid except as specified in (1) and (2) below:

      (1) This differential shall be established within 24 hours of achieving operating temperature and pressure. The differential pressure may be reduced to less than 1.3 psid 24 hours prior to a scheduled shutdown.

      (2) This differential may be decreased to less than 1.3 psid for a maximum of four hours during required operability testing of the HPCI system, RCIC system and the drywell-pressure suppression chamber vacuum breakers.

   b. If the differential pressure of specification 3.7.A.6.a cannot be maintained and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

4.7 CONTAINMENT SYSTEMS

6. Drywell-Suppression Chamber Differential Pressure

   a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

X-107B	Spare (testable)
X-108A	Power
X-108B	CRD Rod Position Indic.
X-109	" " " "
X-110A	Power
X-110B	CRD Rod Position Indic.
X-230	Containment Air Monitoring System

BASES

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 levels given in the specifications, containment pressure during the design basis accident is approximately 49 psig, which is below the maximum of 62 psig. The maximum water level indication of -1 inch corresponds to a downcomer submergence of 4 feet 7 inches and a water volume of 129,000 cubic feet with or without the drywell-suppression chamber differential pressure control. The minimum water level indication of -7 inches with differential pressure control and -8 inches without differential pressure control corresponds to a downcomer submergence of approximately 4 feet and a water volume of approximately 123,000 cubic feet. Maintaining the water level between these levels will assure that the torus water volume and downcomer submergence are within the aforementioned limits during normal plant operation. Alarms, adjusted for instrument error, will notify the operator when the limits of the torus water level are approached. 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 Humboldt 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.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed ("Torus Support System and Attached Piping Analysis for the Browns Ferry Nuclear Plant Units 1, 2, and 3," dated September 9, 1976 and supplemented October 12, 1976) which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure of 1.3 psid and a suppression chamber water level corresponding to a downcomer submergence range of 4.0 feet to 4.60 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

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. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 42
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 November 5, 1976, as supplemented October 15, 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. 42, 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: December 8, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 42

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

79/80
105/106
227/228
Add page 235a
267/268
269/270

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)
2	PdI-64-137 PdI-64-138	Drywell to Suppression Chamber Differential pressure	Indicator 0 to 2 psid	(1) (2) (3)

Amendment No. ~~36~~, 42

NOTES FOR TA : 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.

(5) If the requirements of notes (1) and (2) cannot be met, and if one of the indications cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown within 24 hours.

TABLE 4.2.F
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

	Instrument Channel	Calibration Frequency	Instrument Check
1)	Reactor Water Level	Once/6 months	Each Shift
2)	Reactor Pressure	Once/6 months	Each Shift
3)	Drywell Pressure	Once/6 months	Each Shift
4)	Drywell Temperature	Once/6 months	Each Shift
5)	Suppression Chamber Air Temperature	Once/6 months	Each Shift
6)	Suppression Chamber Water Temperature	Once/6 months	Each Shift
7)	Suppression Chamber Water Level	Once/6 months	Each Shift
8)	Control Rod Position	NA	Each Shift
9)	Neutron Monitoring	(2)	Each Shift
10)	Drywell Pressure (PS-64-67)	Once/6 months	NA
11)	Drywell Pressure (PS-64-58B)	Once/6 months	NA
12)	Drywell Temperature (TR-64-52)	Once/6 months	NA
13)	Timer (IS-64-67)	Once/6 months	NA
14)	CAD Tank Level	Once/6 months	Once/day
15)	Containment Atmosphere Monitors	Once/6 months	Once/day
16)	Drywell to Suppression Chamber Differential Pressure	Once /6 months	Each Shift

Amendment No. ~~38~~, 42

TABLE 4.2.G
SURVEILLANCE REQUIREMENTS FOR CONTROL ROOM ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration	Instrument Check
Control Room Air Supply Duct Radiation Monitors	(1)	once/3 months	once/day (8)
Control Room Isolation Logic	once/6 months	N/A	N/A
Simulated automatic actuation of control room isolation and emergency pressurization system	once/operating cycle	N/A	N/A

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits except as specified in 3.7.A.2.

   a. Minimum water level = -7" (differential pressure control >0 psid)

      -8" (0 psid differential pressure control)

   b. Maximum water level = -1"

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

1. Pressure Suppression Chamber

   a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

3.7 CONTAINMENT SYSTEMS

c. With the suppression pool water temperature > 95°F initiate pool cooling and restore the temperature to < 95°F within 24 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

d. With the suppression pool water temperature > 105°F during testing of ECCS or relief valves, stop all testing, initiate pool cooling and follow the action in specification 3.7.A.1.c above.

e. With the suppression pool water temperature > 120°F following reactor isolation, depressurize to < 200 psig at normal cooldown rates.

f. With the suppression pool water temperature > 110°F during startup or power operation the reactor shall be scrammed.

4.7 CONTAINMENT SYSTEMS

3.7 CONTAINMENT SYSTEMS

6. Drywell-Suppression Chamber Differential Pressure

   a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.3 psid except as specified in (1) and (2) below:

      (1) This differential shall be established within 24 hours of achieving operating temperature and pressure. The differential pressure may be reduced to less than 1.3 psid 24 hours prior to a scheduled shutdown.

      (2) This differential may be decreased to less than 1.3 psid for a maximum of four hours during required operability testing of the HPCI system, RCIC system and the drywell-pressure suppression chamber vacuum breakers.

   b. If the differential pressure of specification 3.7.A.6.a cannot be maintained and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

4.7 CONTAINMENT SYSTEMS

6. Drywell-Suppression Chamber Differential Pressure

   a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

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 levels given in the specification containment pressure during the design basis accident is approximately 49 psig, which is below the maximum of 62 psig. The maximum water level indication of -1 inch corresponds to a downcomer submergence of 4 feet 7 inches and a water volume of 129,000 cubic feet with or without the drywell-suppression chamber differential pressure control. The minimum water level indication of -7 inches with differential pressure control and -8 inches without differential pressure control corresponds to a downcomer submergence of approximately 4 feet and a water volume of approximately 123,000 cubic feet. Maintaining the water level between these levels will assure that the torus water volume and downcomer submergence are within the aforementioned limits during normal plant operation. Alarms, adjusted for instrument error, will notify the operator when the limits of the torus water level are approached. 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 Humboldt 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.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed ("Torus Support System and Attached Piping Analysis for the Browns Ferry Nuclear Plant Units 1, 2, and 3," dated September 9, 1976 and supplemented October 12, 1976) which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure of 1.3 psid and a suppression chamber water level corresponding to a downcomer submergence range of 4.0 feet to 4.60 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

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. 19
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated November 5, 1976, as supplemented October 18, 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. 19, 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: December 8, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 19

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Pages

81
82
83
102
231
Add page 246
246a
285
286

TABLE 3.2.F
SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	LI-3-46 A LI-3-46 B	Reactor Water Level	Indicator -107.5" to +107.5"	(1) (2) (3)
2	PI-3-54 PI-3-61	Reactor Pressure	Indicator 0-1200 psig	(1) (2) (3)
2	PR-64-50 PI-64-67	Drywell Pressure	Recorder 0-80 psia Indicator 0-80 psia	(1) (2) (3)
2	TI-64-52 TR-64-52	Drywell Temperature	Recorder, Indicator 0-400°F	(1) (2) (3)
1	TR-64-52	Suppression Chamber Air Temperature	Recorder 0-400°F	(1) (2) (3)
2	TI-64-55 TIS-64-55	Suppression Chamber Water Temperature	Indicator, 0-400°F	(1) (2) (3)
2	LI-64-54 A LI-64-66	Suppression Chamber Water Level	Indicator -25" to +25"	(1) (2) (3)
1	N/A	Control Rod Position	6V Indicating Lights)	(1) (2) (3) (4)
1	N/A	Neutron Monitoring	SRM, IRM, LPRM 0 to 100% power)	
1	PS-64-67	Drywell Pressure	Alarm at 35 psig)	(1) (2) (3) (4)
1	TR-64-52 and PS-64-58 B and IS-64-67	Drywell Temperature and Pressure and Timer	Alarm if temp. > 281°F and pressure > 2 psig after 30 minute delay)	
1	LI-84-2A	CAD Tank "A" Level	Indicator 0 to 100%	(1)
1	LI-84-13A	CAD Tank "B" Level	Indicator 0 to 100%	(1)

Amendment No. 19

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) (4)
2	PdI-64-137 PdI-64-138	Drywell to Suppression Chamber Differential Pressure	Indicator - 0 to 2 psid	(1) (2) (3)

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.

(5) If the requirements of notes (1) and (2) cannot be met, and if one of the indications cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a Cold Condition within 24 hours.

TABLE 4.2.F
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

	Instrument Channel	Calibration Frequency	Instrument Check
1)	Reactor Water Level	Once/6 months	Each Shift
2)	Reactor Pressure	Once/6 months	Each Shift
3)	Drywell Pressure	Once/6 months	Each Shift
4)	Drywell Temperature	Once/6 months	Each Shift
5)	Suppression Chamber Air Temperature	Once/6 months	Each Shift
6)	Suppression Chamber Water Temperature	Once/6 months	Each Shift
7)	Suppression Chamber Water Level	Once/6 months	Each Shift
8)	Control Rod Position	NA	Each Shift
9)	Neutron Monitoring	(2)	Each Shift
10)	Drywell Pressure (PS-64-67)	Once/6 months	NA
11)	Drywell Pressure (PS-64-58B)	Once/6 months	NA
12)	Drywell Temperature (TR-64-52)	Once/6 months	NA
13)	Timer (IS-64-67)	Once/6 months	NA
14)	CAD Tank Level	Once/6 months	Once/day
15)	Containment Atmosphere Monitors	Once/6 months	Once/day
16)	Drywell to Suppression Chamber Differential Pressure	Once /6 months	Each Shift

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits except as specified in 3.7.A.2.

   a. Minimum water level = -7" (differential pressure control >0 psid)

      -8" (0 psid differential pressure control)

   b. Maximum water level = -1"

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

1. Pressure Suppression Chamber

   a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

3.7 CONTAINMENT SYSTEMS

c. If the specifications of 3.7.A.5.a through 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.

6. Drywell-Suppression Chamber Differential Pressure

a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.3 psid except as specified in (1) and (2) below:

(1) This differential shall be established within 24 hours of achieving operating temperature and pressure. The differential pressure may be reduced to less than 1.3 psid 24 hours prior to a scheduled shutdown.

(2) This differential may be decreased to less than 1.3 psid for a maximum of four hours during required operability testing of the HPCI system, RCIC system, and the drywell-pressure suppression chamber vacuum breakers.

4.7 CONTAINMENT SYSTEMS

6. Drywell-Suppression Chamber Differential Pressure

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

b. If the differential pressure of specification 3.7.A.6.a cannot be maintained and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

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 levels given in the specification, containment pressure during the design basis accident is approximately 49 psig, which is below the maximum of 62 psig. The maximum water level indication of -1 inch corresponds to a downcomer submergence of 4 feet 7 inches and a water volume of 129,000 cubic feet with or without the drywell-suppression chamber differential pressure control. The minimum water level indication of -7 inches with differential pressure control and -8 inches without differential pressure control corresponds to a downcomer submergence of approximately 4 feet and a water volume of approximately 123,000 cubic feet. Maintaining the water level between these levels will assure that the torus water volume and downcomer submergence are within the aforementioned limits during normal plant operation. Alarms, adjusted for instrument error, will notify the operator when the limits of the torus water level are approached. 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 Humboldt 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.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability. 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 no 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 depressurized in a timely manner to avoid the regine 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 pressure, 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.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed ("Torus Support System and Attached Piping Analysis for the Browns Ferry Nuclear Plant Units 1, 2, and 3," dated September 9, 1976 and supplemented October 12, 1976) which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure of 1.3 psid and a suppression chamber water level corresponding to a downcomer submergence range of 4.0 feet to 4.60 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 46 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 19 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

I. Introduction

In conjunction with the Short Term Program (STP) evaluation of Boiling Water Reactor facilities with the Mark I containment system, the Tennessee Valley Authority (TVA) submitted a Plant Unique Analysis (PUA) for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. This analysis was performed to confirm the structural and functional capability of the containment suppression chamber and attached piping, to withstand newly-identified suppression pool hydrodynamic loading conditions which had not been explicitly considered in the original design analysis for the plant. As part of the STP evaluation, specific loading conditions were developed for each Mark I facility, to account for the change in the magnitude of the loads due to plant-specific variations from the reference plant design for which the basic loading conditions were developed.

The results of the staff's review of the hydrodynamic load definition techniques and the Mark I containment plant unique analyses are described in the "Mark I Containment Short Term Program Safety Evaluation Report," NUREG-0408, December 1977. As discussed in this report, the NRC staff has concluded that each Mark I containment system would maintain its integrity and functional capability in the unlikely event of a design basis loss-of-coolant accident (LOCA) and, therefore, that licensed Mark I BWR facilities can continue to operate safely, without undue risk to the health and safety of the public, during an interim period of approximately two years, while a methodical, comprehensive Long Term Program is conducted.

Subsequent to the submittal of the PUA, the licensee was requested to submit proposed Technical Specifications which assure that the allowable range of these two parameters during facility operation would be in accordance with the values utilized in the PUA.

The licensee has been operating this facility with differential pressure control to enhance the safety margins of the containment structure since early 1976. This evaluation provides a more detailed basis for establishing the allowable range of drywell-wetwell differential pressure and torus water level, in order to quantify containment safety margins. This amendment incorporates these parameters into the Technical Specifications with the associated limiting conditions for operation and surveillance requirements.

By letter dated November 5, 1976, as supplemented by letter dated October 18, 1978, the licensee proposed changes to the facility Technical Specifications to incorporate limiting conditions for operation and surveillance requirements for differential pressure control and torus water level. Our evaluation of these proposed changes follows.

II. Evaluation

The licensee has proposed certain Technical Specification requirements for the purpose of assuring that the normal plant operating conditions are within the envelope of conditions considered in their PUA. These Technical Specification changes establish (1) limiting condition for operation (LCOs) for drywell to torus differential pressure and torus water level, and (2) associated surveillance requirements. All other initial conditions utilized in the PUA are either presently included in the Technical Specifications or are configurational conditions which have been confirmed by the licensee and will not change during normal operation.

Differential pressure between the drywell and the suppression chamber will result in leakage of the drywell atmosphere to the lower pressure regions of the reactor building and to the torus airspace. This leakage from the drywell will cause a slow decay in the differential pressure. Therefore, surveillance requirements for the differential pressure have been included in the Technical Specifications. Surveillance frequency of once per operating shift for the differential pressure was selected on the basis of previous operating experience.

The torus water level is not expected to vary significantly during normal operation, unless certain systems connected to the suppression pool are activated. The torus water level would normally be monitored whenever such systems are in use. Therefore, we find that inclusion of periodic torus water level surveillance requirements in the Technical Specifications is not required.

We have reviewed the differential pressure and torus water level monitoring instrumentation systems proposed by the licensee with regard to the number of available channels and the instrumentation accuracy. This type of instrumentation is typically calibrated at six-month intervals. To assure proper operation during such intervals, two monitoring channels for both differential pressure and torus water level have been provided, such that a comparison of the readings will indicate when one of the channels is inoperative or drifting. The errors in the drywell-torus differential pressure instrumentation are sufficiently small relative to the magnitude of the measurement (i.e., a maximum differential pressure measurement error of 0.1 psid in a measurement of 1.0 to 2.0 psid) that they may be neglected, based on the expected load variation with differential pressure and torus water level. Alarms are used for the torus water level indication which have been adjusted for the relative errors in the instrumentation.

There are certain periods during normal plant operation when the differential pressure control cannot be maintained. Therefore, provisions have been included in the Technical Specifications to relax the differential pressure-control requirements during specified periods. The justification for relaxing the differential pressure control during these specific periods and the basis for selecting the duration of the periods are discussed in detail below.

A. Startup and Shutdown

During plant startup and shutdown, the drywell atmosphere undergoes significant barometric changes due to the variation in heat loads from the primary and auxiliary systems. In addition, it is during these periods that the drywell is being either inerted with nitrogen gas or deinerted. In order to keep the periods during which the differential pressure control is not fully effective as short as is reasonable, we have limited the relaxation of the differential pressure control requirements for the startup and shutdown periods to 24 hours following startup and 24 hours prior to a shutdown. This time period was selected on a basis similar to that for the inerting requirements, already existing in the Technical Specifications. The postulated design basis accident for the containment assumes that the primary system is at operating pressure and temperature. During the startup and shutdown transients, the primary system is at operating pressure and temperature for only a part of the transient, during which the differential pressure is being established. These time periods have been shown by previous operating experience to be adequate with respect to the startup and shutdown transients, and at the same time sufficiently small in comparison to the duration of the average power run. Since the principal accident event to which differential pressure control is important to assure containment integrity (i.e., with a factor of safety of two) is a large break LOCA, we have considered whether there is a significantly greater probability of a large break LOCA during the startup and shutdown transients. We have concluded that there is not. Further, the operation of the plant systems is monitored more closely than normal during these periods and a finite magnitude of differential pressure will be available during the majority of these periods to mitigate the potential consequences of an accident.

B. Testing and Maintenance

During normal operation, there are a number of tests which are required to be conducted to demonstrate the continued functional performance of engineered safety features. The testing of certain systems will require, or result in, a reduction in the drywell-torus differential pressure. The operability testing of the drywell-torus vacuum breakers requires the removal of the differential pressure to permit the vacuum breakers to open. For the testing of high-energy systems (e.g. high pressure coolant injection pumps) during normal operation, the discharge flow is routed to the suppression pool. This energy deposition will raise the temperature of the suppression pool, resulting in an increase in torus pressure and a reduction in the differential pressure.

Functional performance testing of engineered safety features is necessary to assure proper maintenance of these systems throughout the life of the plant. Some of these tests (i.e., pump operability and drywell-wetwell vacuum breakers) may require or result in a reduction in the differential pressure. We estimate that not more than four tests will be required each month which will result in a reduction in differential pressure. In order to keep the periods during which the differential pressure control is not fully effective as short as is reasonable, we have permitted a relaxation of differential pressure control in order to conduct the tests, limited to a period of up to four hours. Again, we have carefully considered whether the probability of a large LOCA is significantly greater during these testing periods than that during normal operation. We conclude that it is not. Moreover, only the test of the drywell-wetwell vacuum breakers requires complete removal of the differential pressure.

Provisions have also been included in the Technical Specifications for performing maintenance activities on the differential pressure control system and for resolving operational difficulties which may result in an inadvertent reduction in the differential pressure for a short period of time. In certain circumstances, corrective action can be taken without having to attain a cold shutdown condition. To avoid repeated and unnecessary partial cooldown cycles, a restoration period has been incorporated into the action requirements of the LCO for differential pressure control; i.e., in the event that the differential pressure cannot be restored in six hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours. The six hour restoration period was selected on the basis that it represents an adequate minimum period of time during which any short-term malfunctions could be corrected, coupled with the minimum period of time required to conduct a controlled shutdown. The allowable time to conduct a controlled shutdown has been minimized, because the containment transient response is more a function of the primary system pressure than the reactor power level. On this basis, we find the proposed restoration period and action requirement acceptable.

We conclude that the limits imposed on the periods of time during which operation is permitted without the differential pressure control fully effective provides adequate assurance of overall containment integrity, and the periods of time differential pressure control is completely removed are acceptably small.

Environmental Consideration

We have determined that the 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 the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Section 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.

Conclusion

The proposed Technical Specifications will provide the necessary assurance that the plant's operating conditions remain within the envelope of the conditions assumed in the Plant Unique Analysis (PUA) performed in conjunction with the Mark I Containment Short Term Program. The PUA supplements the facility's Final Safety Analysis Report (FSAR) in that it demonstrates the plant's capability to withstand the suppression pool hydrodynamic loads which were not explicitly considered in the FSAR. We therefore conclude that the proposed changes to the Technical Specifications are acceptable.

We further conclude, based on the considerations discussed above, 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: December 8, 1978

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 46, 42, and 19 to Facility Operating License Nos. DPR-33, DPR-52 and DPR-68 issued to Tennessee Valley Authority (the licensee), which revised the 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 revised the Technical Specifications to incorporate rquirements for establishing and maintaining the drywell to suppression chamber differential pressure and suppression chamber water level, to maintain the margins of safety established in the NRC staff's "Mark I Containment Short Term Program Safety Evaluation," NUREG-0408. Operation in accordance with the conditions specified in NUREG-0408 has been previously authorized in 43 FR 13117 dated March 29, 1978.

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 Section 51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated November 5, 1976, as supplemented by letter dated October 18, 1978, (2) Amendment No. 46 to License No. DPR-33, Amendment No. 42 to License DPR-52, and Amendment No. 19 to License 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 8th day of December 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch No. 3
Division of Operating Reactors