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

August 23, 1991

Mr. Dan A. Nauman
Senior Vice President, Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. Nauman:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. 77948, 77949, 77950) (TS 288)

The Commission has issued the enclosed Amendment Nos. 185 , 198 , and 157 to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3, respectively. These amendments are in response to your application dated October 20, 1990.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 185 to License No. DPR-33
2. Amendment No. 198 to License No. DPR-52
3. Amendment No. 157 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

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NAME	:MSanders	:DMoran:as/dw	:JWilliams	:TRoss for 8/22/91	Bmb
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AMENDMENT NO. 185 FOR BROWNS FERRY UNIT 1 - DOCKET NO. 50-259,
AMENDMENT NO. 198 FOR BROWNS FERRY UNIT 2 - DOCKET NO. 50-260, and
AMENDMENT NO. 157 FOR BROWNS FERRY UNIT 3 - DOCKET NO. 50-296
DATED: August 23, 1991

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 185
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 October 30, 1990 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-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. 185 , 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 its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 23, 1991

UNIT 1
EFFECTIVE PAGE LIST

REMOVE	INSERT
1.0-11	1.0-11
1.0-12	1.0-12*
3.2/4.2-12	3.2/4.2-12
3.2/4.2-13	3.2/4.2-13
3.2/4.2-63	3.2/4.2-63*
3.2/4.2-63a	3.2/4.2-63a
3.2/4.2-69	3.2/4.2-69
3.2/4.2-70	3.2/4.2-70*
3.5/4.5-30	3.5/4.5-30
3.5/4.5-31	3.5/4.5-31
3.7/4.7-1	3.7/4.7-1*
3.7/4.7-2	3.7/4.7-2

*Denotes overleaf page.

GG. Site Boundary - Shall be that line beyond which the land is not owned, leased, or otherwise controlled by TVA.

HH. Unrestricted Area - Any area at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for industrial, commercial, institutional, or recreational purposes.

II. Dose Equivalent I-131 - The DOSE EQUIVALENT I-131 shall be the concentration of I-131 (in µCi/gm) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factor used for this calculation shall be those listed in Table III of TID-14844 "Calculation of Distance Factors for Power and Test Reactor Sites".

JJ. Gaseous Waste Treatment System - The charcoal adsorber vessels installed on the discharge of the steam jet air ejector to provide delay to a unit's offgas activity prior to release.

KK. Members of the Public - Shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category shall not include non-employees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter restricted areas.

LL. Surveillance - Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual limiting conditions for operation unless otherwise stated in an individual Surveillance Requirements. Each Surveillance Requirement shall be performed within the specified time interval with,

(1) A maximum allowable extention not to exceed 25% of the surveillance interval, but

(2) The combined time entered for any 3 consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval

Performance of a Surveillance Requirement within the specified time interval shall constitute compliance and OPERABILITY requirements for a limiting condition for operation and associated action statements unless otherwise required by these specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

DEFINITIONS (Cont'd)

MM. Surveillance Requirements for ASME Section XI Pump and Valve Program - Surveillance Requirements for Inservice Testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

1. Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55(g)(6)(i).

2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these technical specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

3. The provisions of Specification 1.0.LL are applicable to the above required frequencies for performing inservice testing activities.

4. Performance of the above inservice testing activities shall be in addition to other specified surveillance requirements.

5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any technical specification.

6. The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, personnel, and sample expansion included in this generic letter.

NOTES FOR TABLE 3.2.A

1. Whenever the respective functions are required to be OPERABLE there shall be two OPERABLE or tripped trip systems for each function. If the first column cannot be met for one of the trip systems, that trip system or logic for that function shall be tripped (or the appropriate action listed below shall be taken). If the column cannot be met for all trip systems, the appropriate action listed below shall be taken.

    A. Initiate an orderly shutdown and have the reactor in Cold Shutdown in 24 hours.

    B. Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours.

    C. Isolate Reactor Water Cleanup System.

    D. Administratively control the affected system isolation valves in the closed position within one hour and then declare the affected system inoperable.

    E. Initiate primary containment isolation within 24 hours.

    F. The handling of spent fuel will be prohibited and all operations over spent fuels and open reactor wells shall be prohibited.

    G. Isolate the reactor building and start the standby gas treatment system.

    H. Immediately perform a logic system functional test on the logic in the other trip systems and daily thereafter not to exceed 7 days.

    I. Deleted

    J. Withdraw TIP.

    K. Manually isolate the affected lines. Refer to Section 4.2.E for the requirements of an inoperable system.

    L. If one SGTS train is inoperable take actions H or A and F. If two SGTS trains are inoperable take actions A and F.

2. Deleted

3. There are four sensors per steam line of which at least one sensor per trip system must be OPERABLE.

NOTES FOR TABLE 3.2.A (Cont'd)

4. Only required in RUN MODE (interlocked with Mode Switch).

5. Deleted

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

7. A train is considered a trip system.

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

9. Deleted

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

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

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

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

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

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

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

NOTES FOR TABLE 4.2.

(1) The CHANNEL CALIBRATION shall include the use of a known (traceable to the National Bureau of Standards Radiation Measurement System) radioactive source(s) positioned in a reproducible geometry with respect to the sensor or using standards that have obtained from suppliers that participate in measurement assurance activities with the National Bureau of Standards.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

a. Instrument indicates measured levels above the alarm/trip setpoint.

b. Instrument indicates an inoperative/downscale failure.

c. Instrument controls not set in operate mode (stack only).

(3) The channel calibration shall include the use of standard gas samples containing a nominal:

a. Zero volume percent hydrogen (compressed air) and,

b. One volume percent hydrogen, balance nitrogen.

(4) The channel functional test shall demonstrate that automatic isolation of this pathway and control room annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured level above the alarm/trip setpoint.

b. Instrument indicates an inoperative/downscale failure.

c. Instrument controls not set in operate mode.

The two channels are arranged in a coincidence logic such that 2 upscale, or 1 downscale and 1 upscale or 2 downscale will isolate the offgas line.

(5) The noble gas monitor shall have a LLD of 1E-5 (Xe 133 Equivalent).

(6) The noble gas monitor shall have a LLD of 1E-6 (Xe 133 Equivalent).

Table 4.2.L
Anticipated Transient Without Scram (ATWS) - Recirculation Pump Trip (RPT) Instrumentation Surveillance

Function	Functional Test	Channel Calibration	Instrument Check
Reactor Vessel Water Level Low (LS-3-58A1-D1)	M(27)	R(28)	N/A
Reactor Vessel Dome Pressure High (PIS-3-204A-D)	M(27)	R(28)	N/A

Trip setting of 100 mr/hr for the monitors in the refueling zone are based upon initiating normal ventilation isolation and SGTS operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the SGTS.

Flow integrators and sump fill rate and pump out rate timers are used to determine leakage in the drywell. A system whereby the time interval to fill a known volume will be utilized to provide a backup. An air sampling system is also provided to detect leakage inside the primary containment (See Table 3.2.E).

For each parameter monitored, as listed in Table 3.2.F, there are two channels of instrumentation except as noted. By comparing readings between the two channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

Instrumentation is provided for isolating the control room and initiating a pressurizing system that processes outside air before supplying it to the control room. An accident signal that isolates primary containment will also automatically isolate the control room and initiate the emergency pressurization system. In addition, there are radiation monitors in the normal ventilation system that will isolate the control room and initiate the emergency pressurization system. Activity required to cause automatic actuation is about one mRem/hr.

Because of the constant surveillance and control exercised by TVA over the Tennessee Valley, flood levels of large magnitudes can be predicted in advance of their actual occurrence. In all cases, full advantage will be taken of advance warning to take appropriate action whenever reservoir levels above normal pool are predicted. Therefore, during flood conditions, the plant will be permitted to operate until water begins to run across the top of the pumping station at elevation 565. Seismically qualified, redundant level switches each powered from a separate division of power are provided at the pumping station to give main control room indication of this condition. At that time an orderly shutdown of the plant will be initiated, although surges even to a depth of several feet over the pumping station deck will not cause the loss of the main condenser circulating water pumps.

The operability of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation dose to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

The operability of the seismic instrumentation ensures that sufficient capability is available to promptly determine the seismic response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for Browns Ferry Nuclear Plant and to determine whether the plant can continue to be operated safely. The instrumentation provided is consistent with specific portions of the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes."

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments will be calculated in accordance with guidance provided in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentration of potentially explosive gas mixtures in the offgas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with guidance provided in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20 Appendix B, Table II, Column 2. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

ATWS/RPT, Anticipated Transients without Scram/Recirculation Pump Trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an ATWS event. The response of the plant to this postulated event (ATWS/RPT) follows the BWR Owners Group Report by General Electric NEDE-31096-P-A and the accompanying NRC Staff Safety Evaluation Report.

ATWS/RPT utilizes the engineered safety feature (ESF) master/slave analog trip units (ATU) which consists of four level and four pressure channels total. The initiating logic consists of two independent trip systems each consisting of two reactor dome high pressure channels and two reactor vessel low level channels. A coincident trip of either two low levels or two high pressures in the same trip system causes initiation of ATWS/RPT. This signal from either trip system opens one of two EOC

3.5.E. High Pressure Coolant Injection System (HPCIS)

The HPCIS is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCIS continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or Core Spray system operation maintains core cooling. The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to pump 5000 gpm at reactor pressures between 1120 and 150 psig. The HPCIS is not required to be OPERABLE below 150 psig since this is well within the range of the low pressure cooling systems and below the pressure of any events for which HPCI is required to provide core cooling.

The minimum required NPSH for HPCI is 21 feet. There is adequate elevation head between the suppression pool and the HPCI pump, such that the required NPSH is available with a suppression pool temperature up to 140°F with no containment back pressure.

The HPCIS is not designed to operate at full capacity until reactor pressure exceeds 150 psig and the steam supply to the HPCI turbine is automatically isolated before reactor pressure decreases below 100 psig. The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the HPCI turbine for OPERABILITY testing yet, still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. The ADS provides additional backup to reduce pressure to the range where the CSS and RHRS will inject into the vessel if necessary. Considering the low reactor pressure, the redundancy and availability of CSS, RHRS, and ADS during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate HPCI OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate HPCI OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the HPCIS inoperable, a seven-day period to return the system to service is justified based on the availability of the ADS, CSS, RHRS (LPCI) and the RCICS. The availability of these redundant and diversified systems provides adequate assurance of core cooling while HPCIS is out of service.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the HPCIS will be OPERABLE when required.

3.5.F Reactor Core Isolation Cooling System (RCICS)

The RCICS functions to provide core cooling and makeup water to the reactor vessel during shutdown and isolation from the main heat sink and for certain pipe break accidents. The RCICS provides its design flow between 150 psig and 1120 psig reactor pressure. Below 150 psig, RCICS is not required to be OPERABLE since this pressure is substantially below that for any events in which RCICS is required to provide core cooling. RCICS will continue to operate below 150 psig at reduced flow until it automatically isolates at greater than or equal to 50 psig reactor steam pressure. 150 psig is also below the shutoff head of the CSS and RHRS, thus, considerable overlap exists with the cooling systems that provide core cooling at low reactor pressure. The minimum required NPSH for RCIC is 20 feet. There is adequate elevation head between the suppression pool and the RCIC pump, such that the required NPSH is available with a suppression pool temperature up to 140°F with no containment back pressure.

The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the RCIC turbine for OPERABILITY testing, yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. Considering the low reactor pressure and the availability of the low pressure coolant systems during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate RCIC OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate RCIC OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the RCICS inoperable, a seven-day period to return the system to service is justified based on the availability of the HPCIS to cool the core and upon consideration that the average risk associated with failure of the RCICS to cool the core when required is not increased.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the RCICS will be OPERABLE when required.

3.5.G Automatic Depressurization System (ADS)

This specification ensures the OPERABILITY of the ADS under all conditions for which the depressurization of the nuclear system is an essei. al response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low-pressure coolant injection (LPCI) and the core spray subsystems can operate to protect the fuel barrier. Note that this specification applies only to the automatic feature of the pressure relief system.

Specification 3.6.D specifies the requirements for the pressure relief function of the valves. It is possible for any number of the valves assigned to the ADS to be incapable of performing their ADS functions because of instrumentation failures, yet be fully capable of performing their pressure relief function.

LIMITING CONDITIONS FOR OPERATION

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.

   a. Minimum water level = -6.25" (differential pressure control >0 psid) -7.25" (0 psid differential pressure control)

   b. Maximum water level = -1"

SURVEILLANCE REQUIREMENTS

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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A. Primary Containment 3.7.A.1 (Cont'd) 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 CONDITION within the next 6 hours and in the COLD SHUTDOWN CONDITION 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 > 110°F during the STARTUP CONDITION, HOT STANDBY CONDITION (with all control rods not inserted), or REACTOR POWER OPERATION, the reactor shall be scrammed. f. With the suppression pool water temperature > 120°F following reactor isolation, depressurize to < 200 psig at normal cooldown rates.	

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 198
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 October 30, 1990 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 198 , 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 its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 23, 1991

UNIT 2
EFFECTIVE PAGE LIST

REMOVE	INSERT
1.0-11	1.0-11
1.0-12	1.0-12*
3.2/4.2-7	3.2/4.2-7*
3.2/4.2-8	3.2/4.2-8
3.2/4.2-12	3.2/4.2-12
3.2/4.2-13	3.2/4.2-13
3.2/4.2-31	3.2/4.2-31*
3.2/4.2-32	3.2/4.2-32
3.2/4.2-54	3.2/4.2-54
3.2/4.2-55	3.2/4.2-55*
3.2/4.2-63	3.2/4.2-63*
3.2/4.2-63a	3.2/4.2-63a
3.2/4.2-69	3.2/4.2-69*
3.2/4.2-70	3.2/4.2-70
3.5/4.5-28	3.5/4.5-28
3.5/4.5-29	3.5/4.5-29
3.6/4.6-32	3.6/4.6-32
3.6/4.6-33	3.6/4.6-33*

*Denotes overleaf page.

GG. Site Boundary - Shall be that line beyond which the land is not owned, leased, or otherwise controlled by TVA.

HH. Unrestricted Area - Any area at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for industrial, commercial, institutional, or recreational purposes.

II. Dose Equivalent I-131 - The DOSE EQUIVALENT I-131 shall be the concentration of I-131 (in µCi/gm) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factor used for this calculation shall be those listed in Table III of TID-14844 "Calculation of Distance Factors for Power and Test Reactor Sites".

JJ. Gaseous Waste Treatment System - The charcoal adsorber vessels installed on the discharge of the steam jet air ejector to provide delay to a unit's offgas activity prior to release.

KK. Members of the Public - Shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category shall not include non-employees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter restricted areas.

LL. Surveillance - Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual limiting conditions for operation unless otherwise stated in an individual Surveillance Requirements. Each Surveillance Requirement shall be performed within the specified time interval with,

(1) A maximum allowable extention not to exceed 25% of the surveillance interval, but

(2) The combined time entered for any 3 consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval

Performance of a Surveillance Requirement within the specified time interval shall constitute compliance and OPERABILITY requirements for a limiting condition for operation and associated action statements unless otherwise required by these specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

MM. Surveillance Requirements for ASME Section XI Pump and Valve Program - Surveillance Requirements for Inservice Testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

1. Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55(g)(6)(i).

2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these technical specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

3. The provisions of Specification 1.0.LL are applicable to the above required frequencies for performing inservice testing activities.

4. Performance of the above inservice testing activities shall be in addition to other specified surveillance requirements.

5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any technical specification.

6. The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, personnel, and sample expansion included in this generic letter.

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable Per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level(6) (LIS-3-203 A-D)	$\geq$ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure (PS-68-93 and -94)	100 $\pm$ 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D)	$\geq$ 398" above vessel zero	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PIS-64-56A-D)	$\leq$ 2.5 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS

TABLE 3.2.A (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable Per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - High Radiation Main Steam Line Tunnel (6)	3 times normal rated full power background (13)	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Low Pressure Main Steam Line (PIS-1-72, 76, 82, 86)	≥ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel - High Flow Main Steam Line (PdIS-1-13A-D, 25A-D, 36A-D, 50A-D)	≤ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation
2(12)	Instrument Channel - Main Steam Line Tunnel High Temperature	≤ 200°F	B	1. Above trip setting initiates Main Steam Line Isolation.
1	Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	≤ 100 mr/hr or downscale	G	1. 1 upscale or 2 downscale will a. Initiate SGTS b. Isolate reactor zone and refueling floor. c. Close atmosphere control system.

NOTES FOR TABLE 3.2.

1. Whenever the respective functions are required to be OPERABLE there shall be two OPERABLE or tripped trip systems for each function. If the first column cannot be met for one of the trip systems, that trip system or logic for that function shall be tripped (or the appropriate action listed below shall be taken). If the column cannot be met for all trip systems, the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in Cold Shutdown in 24 hours.

   B. Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Administratively control the affected system isolation valves in the closed position within one hour and then declare the affected system inoperable.

   E. Initiate primary containment isolation within 24 hours.

   F. The handling of spent fuel will be prohibited and all operations over spent fuels and open reactor wells shall be prohibited.

   G. Isolate the reactor building and start the standby gas treatment system.

   H. Immediately perform a logic system functional test on the logic in the other trip systems and daily thereafter not to exceed 7 days.

   I. Deleted

   J. Withdraw TIP.

   K. Manually isolate the affected lines. Refer to Section 4.2.E for the requirements of an inoperable system.

   L. If one SGTS train is inoperable take actions H or A and F. If two SGTS trains are inoperable take actions A and F.

2. Deleted

3. There are four sensors per steam line of which at least one sensor per trip system must be OPERABLE.

4. Only required in RUN MODE (interlocked with Mode Switch).

5. Deleted

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

7. A train is considered a trip system.

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

9. Deleted

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

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

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

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

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

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

TABLE 3.2.F

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	LI-3-58A LI-3-58B	Reactor Water Level	Indicator - 155" to +60"	(1) (2) (3)
2	PI-3-74A PI-3-74B	Reactor Pressure	Indicator 0-1200 psig	(1) (2) (3)
2	XR-64-50 PI-64-67B TI-64-52AB	Drywell Pressure	Recorder 0-80 psia Indicator 0-80 psia	(1) (2) (3)
2	XR-64-50	Drywell Temperature	Recorder, Indicator 0-400°F	(1) (2) (3)
1	XR-64-52	Suppression Chamber Air Temperature	Recorder 0-400°F	(1) (2) (3)
1	N/A	Control Rod Position	6V Indicating Lights)	
1	N/A	Neutron Monitoring	SRM, IRM, LPRM 0 to 100% power)	(1) (2) (3) (4)
1	PS-64-67B	Drywell Pressure	Alarm at 35 psig)	
1	TS-64-52A & PIS-64-58A & IS-64-67A	Drywell Temperature and Pressure and Timer	Alarm if temp. > 281°F and pressure >2.5 psig after 30 minute delay)	(1) (2) (3) (4)
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)

TABLE 3.2.F (cont'd)

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	H_2M - 76 - 94 H_2M - 76 - 104	Drywell and Torus Hydrogen 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)
1/Valve		Relief Valve Tailpipe Thermocouple Temperature or Acoustic Monitor on Relief Valve Tailpipe		(5)
1	RR-90-272CD RR-90-273CD	High Range Primary Containment Radiation Recorders	Recorder 1-10^7 R/Hr	(7)(8)
2	LI-64-159A XR-64-159	Suppression Chamber Water Level-Wide Range	Indicator, Recorder 0-240"	(1) (2) (3)
2	PI-64-160A XR-64-159	Drywell Pressure Wide Range	Indicator, Recorder) 0-300 psig)	(1) (2) (3)
2	TI-64-161 TR-64-161 TI-64-162 TR-64-162	Suppression Pool Bulk Temperature	Indicator, Recorder)) 30° - 230° F))	(1) (2) (3) (4) (6)
1	RM-90-306 RR-90-360	Wide Range Gaseous Effluent Radiation Monitor and recorder	Monitor and recorder (Noble Gas 10^{-7} - 10^{+5} μCi/cc)	(7)(8)(9)

TABLE 4.2.F

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
1) Reactor Water Level (LI-3-58A&B)	Once/6 months	Each Shift
2) Reactor Pressure (PI-3-74A&B)	Once/6 months	Each Shift
3) Drywell Pressure (PI-64-67B) and XR-64-50	Once/6 months	Each Shift
4) Drywell Temperature (TI-64-52AB) and XR-64-50	Once/6 months	Each Shift
5) Suppression Chamber Air Temperature (XR-64-52)	Once/6 months	Each Shift
8) Control Rod Position	N/A	Each Shift
9) Neutron Monitoring	(2)	Each Shift
10) Drywell Pressure (PS-64-67B)	Once/6 months	N/A
11) Drywell Pressure (PIS-64-58A)	Once/6 months	N/A
12) Drywell Temperature (TS-64-52A)	Once/6 months	N/A
13) Timer (IS-64-67A)	Once/6 months	N/A
14) CAD Tank Level	Once/6 months	Once/day
15) Containment Atmosphere Monitors	Once/6 months	Once/day

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

Instrument Channel	Calibration Frequency	Instrument Check
16) Drywell to Suppression Chamber Differential Pressure	Once/6 months	Each Shift
17) Relief Valve Tailpipe Thermocouple Temperature	N/A	Once/month (24)
18) Acoustic Monitor on Relief Valve Tailpipe	Once/cycle (25)	Once/month (26)
19) High Range Primary Containment Radiation Monitors (RR-90-272CD) (RR-90-273CD)	Once/18 Months (30)	Once/month
20) Suppression Chamber Water Level-Wide Range (LI-64-159A) (XR-64-159)	Once/18 Months	Once/shift
21) Drywell Pressure - Wide Range (PI-64-160A) (XR-64-159)	Once/18 Months	Once/shift
22) Suppression Pool Bulk Temperature (TI-64-161) (TR-64-161) (TI-64-162) (TR-64-162)	Once/18 Months	Once/shift
23) Wide Range Gaseous Effluent Radiation Monitor and recorder (RM-90-306 and RR-90-360)	Once/18 Months	Once/shift

NOTES FOR TABLE 4.2.K

(1) The CHANNEL CALIBRATION shall include the use of a known (traceable to the National Bureau of Standards Radiation Measurement System) radioactive source(s) positioned in a reproducible geometry with respect to the sensor or using standards that have obtained from suppliers that participate in measurement assurance activities with the National Bureau of Standards.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

a. Instrument indicates measured levels above the alarm/trip setpoint.

b. Instrument indicates an inoperative/downscale failure.

c. Instrument controls not set in operate mode (stack only).

(3) The channel calibration shall include the use of standard gas samples containing a nominal:

a. Zero volume percent hydrogen (compressed air) and,

b. One volume percent hydrogen, balance nitrogen.

(4) The channel functional test shall demonstrate that automatic isolation of this pathway and control room annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured level above the alarm/trip setpoint.

b. Instrument indicates an inoperative/downscale failure.

c. Instrument controls not set in operate mode.

The two channels are arranged in a coincidence logic such that 2 upscale, or 1 downscale and 1 upscale or 2 downscale will isolate the offgas line.

(5) The noble gas monitor shall have a LLD of 1E-5 (Xe 133 Equivalent).

(6) The noble gas monitor shall have a LLD of 1E-6 (Xe 133 Equivalent).

Table 4.2.L
Anticipated Transient Without Scram (ATWS) -
Recirculation Pump Trip (RPT) Instrumentation Surveillance

Function	Functional Test	Channel Calibration	Instrument Check
Reactor Vessel Water Level Low (LS-3-58A1-D1)	M(27)	R(28)	N/A
Reactor Vessel Dome Pressure High (PIS-3-204A-D)	M(27)	R(28)	N/A

adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two post treatment off-gas radiation monitors are provided and, when their trip point is reached, cause an isolation of the off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip or both have a downscale trip.

Both instruments are required for trip but the instruments are set so that the instantaneous stack release rate limit given in Specification 3.8 is not exceeded.

Four radiation monitors are provided for each unit which initiate Primary Containment Isolation (Group 6 isolation valves) Reactor Building Isolation and operation of the Standby Gas Treatment System. These instrument channels monitor the radiation in the reactor zone ventilation exhaust ducts and in the refueling zone.

Trip setting of 100 mr/hr for the monitors in the refueling zone are based upon initiating normal ventilation isolation and SGTS operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the SGTS.

Flow integrators and sump fill rate and pump out rate timers are used to determine leakage in the drywell. A system whereby the time interval to fill a known volume will be utilized to provide a backup. An air sampling system is also provided to detect leakage inside the primary containment (See Table 3.2.E).

For each parameter monitored, as listed in Table 3.2.F, there are two channels of instrumentation except as noted. By comparing readings between the two channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

Instrumentation is provided for isolating the control room and initiating a pressurizing system that processes outside air before supplying it to the control room. An accident signal that isolates primary containment will also automatically isolate the control room and initiate the emergency pressurization system. In addition, there are radiation monitors in the normal ventilation system that will isolate the control room and initiate the emergency pressurization system. Activity required to cause automatic actuation is about one mRem/hr.

Because of the constant surveillance and control exercised by TVA over the Tennessee Valley, flood levels of large magnitudes can be predicted in advance of their actual occurrence. In all cases, full advantage will be taken of advance warning to take appropriate action whenever reservoir levels above normal pool are predicted. Therefore, during flood conditions, the plant will be permitted to operate until water begins to run across the top of the pumping station at elevation 565. Seismically qualified, redundant level switches each powered from a separate division of power are provided at the pumping station to give main control room indication of this condition. At that time an orderly shutdown of the plant will be initiated, although surges even to a depth of several feet over the pumping station deck will not cause the loss of the main condenser circulating water pumps.

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation dose to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the seismic response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for Browns Ferry Nuclear Plant and to determine whether the plant can continue to be operated safely. The instrumentation provided is consistent with specific portions of the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes."

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments will be calculated in accordance with guidance provided in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentration of potentially explosive gas mixtures in the off-gas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with guidance provided in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20 Appendix B, Table II, Column 2. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3.5.E. High Pressure Coolant Injection System (HPCIS)

The HPCIS is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCIS continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or Core Spray system operation maintains core cooling. The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to pump 5000 gpm at reactor pressures between 1120 and 150 psig. The HPCIS is not required to be OPERABLE below 150 psig since this is well within the range of the low pressure cooling systems and below the pressure of any events for which HPCI is required to provide core cooling.

The minimum required NPSH for HPCI is 21 feet. There is adequate elevation head between the suppression pool and the HPCI pump, such that the required NPSH is available with a suppression pool temperature up to 140°F with no containment back pressure.

The HPCIS is not designed to operate at full capacity until reactor pressure exceeds 150 psig and the steam supply to the HPCI turbine is automatically isolated before reactor pressure decreases below 100 psig. The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the HPCI turbine for OPERABILITY testing yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. The ADS provides additional backup to reduce pressure to the range where the CSS and RHRS will inject into the vessel if necessary. Considering the low reactor pressure, the redundancy and availability of CSS, RHRS, and ADS during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate HPCI OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate HPCI OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the HPCIS inoperable, a seven-day period to return the system to service is justified based on the availability of the ADS, CSS, RHRS (LPCI) and the RCICS. The availability of these redundant and diversified systems provides adequate assurance of core cooling while HPCIS is out of service.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the HPCIS will be OPERABLE when required.

3.5.F Reactor Core Isolation Cooling System (RCICS)

The RCICS functions to provide core cooling and makeup water to the reactor vessel during shutdown and isolation from the main heat sink and for certain pipe break accidents. The RCICS provides its design flow between 150 psig and 1120 psig reactor pressure. Below 150 psig, RCICS is not required to be OPERABLE since this pressure is substantially below that for any events in which RCICS is required to provide core cooling. RCICS will continue to operate below 150 psig at reduced flow until it automatically isolates at greater than or equal to 50 psig reactor steam pressure. 150 psig is also below the shutoff head of the CSS and RHRS, thus, considerable overlap exists with the cooling systems that provide core cooling at low reactor pressure. The minimum required NPSH for RCIC is 20 feet. There is adequate elevation head between the suppression pool and the RCIC pump, such that the required NPSH is available with a suppression pool temperature up to 140°F with no containment back pressure.

The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the RCIC turbine for OPERABILITY testing, yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. Considering the low reactor pressure and the availability of the low pressure coolant systems during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate RCIC OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate RCIC OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the RCICS inoperable, a seven-day period to return the system to service is justified based on the availability of the HPCIS to cool the core and upon consideration that the average risk associated with failure of the RCICS to cool the core when required is not increased.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the RCICS will be OPERABLE when required.

3.5.G Automatic Depressurization System (ADS)

The ADS consists of six of the thirteen relief valves. It is designed to provide depressurization of the reactor coolant system during a small break loss of coolant accident (LOCA) if HPCI fails or is unable to maintain the required water level in the reactor vessel. ADS operation reduces the reactor vessel pressure to within the operating pressure range of the low pressure emergency core cooling systems (core spray and LPCI) so that they can operate to protect the fuel barrier. Specification 3.5.G applies only to the automatic feature of the pressure relief system.

Specification 3.6.D specifies the requirements for the pressure relief function of the valves. It is possible for any number of the valves assigned to the ADS to be incapable of performing their ADS functions because of instrumentation failures, yet be fully capable of performing their pressure relief function.

The emergency core cooling system LOCA analyses for small line breaks assumed that four of the six ADS valves were operable. By requiring six

3.6.E/4.6.E (Cont'd)

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10 percent or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the unit shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump diffuser body; however, the converse is not true. The lack of any substantial stress in the jet pump diffuser body makes failure impossible without an initial nozzle-riser system failure.

3.6.F/4.6.F Recirculation Pump Operation

Operation without forced recirculation is permitted for up to 12 hours when the reactor is not in the RUN mode. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring at least one recirculation pump to be operable while in the RUN mode (i.e., requiring a manual scram if both recirculation pumps are tripped) provides protection against the potential occurrence of core thermal-hydraulic instabilities at low flow conditions.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50% of its rated speed provides assurance when going from one-to-two pump operation that excessive vibration of the jet pump risers will not occur.

3.6.G/4.6.G Structural Integrity

The requirements for the reactor coolant systems inservice inspection program have been identified by evaluating the need for a sampling examination of areas of high stress and highest probability of failure in the system and the need to meet as closely as possible the requirements of Section XI, of the ASME Boiler and Pressure Vessel Code.

The program reflects the built-in limitations of access to the reactor coolant systems.

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in Section 4.6.G.4 to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgment from actual plant observation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in three additional examinations above the requirements of Section XI of ASME Code.

An augmented inservice surveillance program is required to determine whether any stress corrosion has occurred in any stainless steel piping, stainless components, and highly-stressed alloy steel such as hanger springs, as a result of environmental conditions associated with the March 22, 1975 fire.

REFERENCES

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)
2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code
3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)
4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)
5. Mechanical Maintenance Instruction 46 (Mechanical Equipment, Concrete, and Structural Steel Cleaning Procedure for Residue From Plant Fire - Units 1 and 2)
6. Mechanical Maintenance Instruction 53 (Evaluation of Corrosion Damage of Piping Components Which Were Exposed to Residue From March 22, 1975 Fire)
7. Plant Safety Analysis (BFNP FSAR Subsection 4.12)

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 157
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 October 30, 1990 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 def- se and security or to the health and safety of the public; and

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-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. 157 , 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 its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 23, 1991

UNIT 3
EFFECTIVE PAGE LIST

REMOVE	INSERT
1.0-11	1.0-11
1.0-12	1.0-12*
3.2/4.2-12	3.2/4.2-12
3.2/4.2-13	3.2/4.2-13
3.2/4.2-62	3.2/4.2-62*
3.2/4.2-62a	3.2/4.2-62a
3.2/4.2-68	3.2/4.2-68
3.2/4.2-69	3.2/4.2-69*
3.5/4.5-31	3.5/4.5-31
3.5/4.5-32	3.5/4.5-32

*Denotes overleaf page.

GG. Site Boundary - Shall be that line beyond which the land is not owned, leased, or otherwise controlled by TVA.

HH. Unrestricted Area - Any area at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for industrial, commercial, institutional, or recreational purposes.

II. Dose Equivalent I-131 - The DOSE EQUIVALENT I-131 shall be the concentration of I-131 (in μCi/gm) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factor used for this calculation shall be those listed in Table III of TID-14844 "Calculation of Distance Factors for Power and Test Reactor Sites".

JJ. Gaseous Waste Treatment System - The charcoal adsorber vessels installed on the discharge of the steam jet air ejector to provide delay to a unit's offgas activity prior to release.

KK. Members of the Public - Shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category shall not include non-employees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter restricted areas.

LL. Surveillance - Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual limiting conditions for operation unless otherwise stated in an individual Surveillance Requirements. Each Surveillance Requirement shall be performed within the specified time interval with,

(1) A maximum allowable extention not to exceed 25% of the surveillance interval, but

(2) The combined time entered for any 3 consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval

Performance of a Surveillance Requirement within the specified time interval shall constitute compliance and OPERABILITY requirements for a limiting condition for operation and associated action statements unless otherwise required by these specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

MM. Surveillance Requirements for ASME Section XI Pump and Valve Program - Surveillance Requirements for Inservice Testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

1. Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these technical specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

3. The provisions of Specification 1.0.LL are applicable to the above required frequencies for performing inservice testing activities.

4. Performance of the above inservice testing activities shall be in addition to other specified surveillance requirements.

5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any technical specification.

6. The inservice inspection program for ˍ ›ing identifed in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, personnel, and sample expansion included in this generic letter.

NOTES FOR TABLE 3.2.

1. Whenever the respective functions are required to be OPERABLE, there shall be two OPERABLE or tripped trip systems for each function. If the first column cannot be met for one of the trip systems, that trip system or logic for that function shall be tripped (or the appropriate action listed below shall be taken). If the column cannot be met for all trip systems, the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in COLD SHUTDOWN CONDITION in 24 hours.

   B. Initiate an orderly load reduction and have main steam lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Administratively control the affected system isolation valves in the closed position within one hour and then declare the affected system inoperable.

   E. Initiate primary containment isolation within 24 hours.

   F. The handling of spent fuel will be prohibited and all operations over spent fuels and open reactor wells shall be prohibited.

   G. Isolate the reactor building and start the standby gas treatment system.

   H. Immediately perform a logic system functional test on the logic in the other trip systems and daily thereafter not to exceed 7 days.

   I. DELETED

   J. Withdraw TIP.

   K. Manually isolate the affected lines. Refer to Section 4.2.E for the requirements of an inoperable system.

   L. If one SGTS train is inoperable take action H or actions A and F. If two SGTS trains are inoperable take actions A and F.

2. Deleted

3. There are four sensors per steam line of which at least one sensor per trip system must be OPERABLE.

4. Only required in RUN MODE (interlocked with Mode Switch).

5. Deleted

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

7. A train is considered a trip system.

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

9. DELETED

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

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

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

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

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

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

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

NOTES FOR TABLE 4.2.K

(1) The CHANNEL CALIBRATION shall include the use of a known (traceable to the National Bureau of Standards Radiation Measurement System) radioactive source(s) positioned in a reproducible geometry with respect to the sensor or using standards that have obtained from suppliers that participate in measurement assurance activities with the National Bureau of Standards.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

a. Instrument indicates measured levels above the alarm/trip setpoint.

b. Instrument indicates an inoperable/downscale failure.

c. Instrument controls not set in operate mode (stack only).

(3) The channel calibration shall include the use of standard gas samples containing a nominal:

a. Zero volume percent hydrogen (compressed air) and,

b. One volume percent hydrogen, balance nitrogen.

(4) The channel functional test shall demonstrate that automatic isolation of this pathway and control room annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the alarm/trip setpoint.

b. Instrument indicates an inoperative/downscale failure.

c. Instrument controls not set in operate mode.

The two channels are arranged in a coincidence logic such that 2 upscale, or 1 downscale and 1 upscale or 2 downscale will isolate the offgas line.

(5) The noble gas monitor shall have a LLD of 1E-5 (Xe 133 Equivalent).

(6) The noble gas monitor shall have a LLD of 1E-6 (Xe 133 Equivalent).

Table 4.2.L
Anticipated Transient Without Scram (ATWS) -
Recirculation Pump Trip (RPT) Instrumentation Surveillance

Function	Functional Test	Channel Calibration	Instrument Check
Reactor Vessel Water Level Low (LS-3-58A1-D1)	M(28)	R(29)	N/A
Reactor Vessel Dome Pressure High (PIS-3-204A-D)	M(28)	R(29)	N/A

Trip setting of 100 mr/hr for the monitors in the refueling zone are based upon initiating normal ventilation isolation and SGTS operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the SGTS.

Flow integrators and sump fill rate and pump out rate timers are used to determine leakage in the drywell. A system whereby the time interval to fill a known volume will be utilized to provide a backup. An air sampling system is also provided to detect leakage inside the primary containment (See Table 3.2.E).

For each parameter monitored, as listed in Table 3.2.F, there are two channels of instrumentation except as noted. By comparing readings between the two channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

Instrumentation is provided for isolating the control room and initiating a pressurizing system that processes outside air before supplying it to the control room. An accident signal that isolates primary containment will also automatically isolate the control room and initiate the emergency pressurization system. In addition, there are radiation monitors in the normal ventilation system that will isolate the control room and initiate the emergency pressurization system. Activity required to cause automatic actuation is about one mRem/hr.

Because of the constant surveillance and control exercised by TVA over the Tennessee Valley, flood levels of large magnitudes can be predicted in advance of their actual occurrence. In all cases, full advantage will be taken of advance warning to take appropriate action whenever reservoir levels above normal pool are predicted. Therefore, during flood conditions, the plant will be permitted to operate until water begins to run across the top of the pumping station at elevation 565. Seismically qualified, redundant level switches each powered from a separate division of power are provided at the pumping station to give main control room indication of this condition. At that time an orderly shutdown of the plant will be initiated, although surges even to a depth of several feet over the pumping station deck will not cause the loss of the main condenser circulating water pumps.

The operability of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation dose to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

The operability of the seismic instrumentation ensures that sufficient capability is available to promptly determine the seismic response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for Browns Ferry Nuclear Plant and to determine whether the plant can continue to be operated safely. The instrumentation provided is consistent with specific portions of the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes."

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments will be calculated in accordance with guidance provided in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentration of potentially explosive gas mixtures in the offgas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with guidance provided in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20 Appendix B, Table II, Column 2. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

ATWS/RPT, Anticipated Transients without Scram/Recirculation Pump Trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an ATWS event. The response of the plant to this postulated event (ATWS/RPT) follows the BWR Owners Group Report by General Electric NEDE-31096-P-A and the accompanying NRC Staff Safety Evaluation Report.

ATWS/RPT utilizes the engineered safety feature (ESF) master/slave analog trip units (ATU) which consists of four level and four pressure channels total. The initiating logic consists of two independent trip systems each consisting of two reactor dome high pressure channels and two reactor vessel low level channels. A coincident trip of either two low levels or two high pressures in the same trip system causes initiation of ATWS/RPT. This signal from either trip system opens one of two EOC

3.5.E. High Pressure Coolant Injection System (HPCIS)

The HPCIS is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCIS continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or Core Spray system operation maintains core cooling. The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to pump 5000 gpm at reactor pressures between 1120 and 150 psig. The HPCIS is not required to be OPERABLE below 150 psig since this is well within the range of the low pressure cooling systems and below the pressure of any events for which HPCI is required to provide core cooling.

The minimum required NPSH for HPCI is 21 feet. There is adequate elevation head between the suppression pool and the HPCI pump, such that the required NPSH is available with a suppression pool temperature up to 140°F with no containment back pressure.

The HPCIS is not designed to operate at full capacity until reactor pressure exceeds 150 psig and the steam supply to the HPCI turbine is automatically isolated before reactor pressure decreases below 100 psig. The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the HPCI turbine for OPERABILITY testing, yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. The ADS provides additional backup to reduce pressure to the range where the CSS and RHRS will inject into the vessel if necessary. Considering the low reactor pressure, the redundancy and availability of CSS, RHRS, and ADS during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate HPCI OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate HPCI OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the HPCIS inoperable, a seven-day period to return the system to service is justified based on the availability of the ADS, CSS, RHRS (LPCI) and the RCICS. The availability of these redundant and diversified systems provides adequate assurance of core cooling while HPCIS is out of service.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the HPCIS will be OPERABLE when required.

3.5 BASES (Cont'd)

3.5.F Reactor Core Isolation Cooling System (RCICS)

The RCICS functions to provide core cooling and makeup water to the reactor vessel during shutdown and isolation from the main heat sink and for certain pipe break accidents. The RCICS provides its design flow between 150 psig and 1120 psig reactor pressure. Below 150 psig, RCICS is not required to be OPERABLE since this pressure is substantially below that for any events in which RCICS is required to provide core cooling. RCICS will continue to operate below 150 psig at reduced flow until it automatically isolates at greater than or equal to 50 psig reactor steam pressure. 150 psig is also below the shutoff head of the CSS and RHRS, thus, considerable overlap exists with the cooling systems that provide core cooling at low reactor pressure. The minimum required NPSH for RCIC is 20 feet. There is adequate elevation head between the suppression pool and the RCIC pump, such that the required NPSH is available with a suppression pool temperature up to 140°F with no containment back pressure.

The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the RCIC turbine for OPERABILITY testing, yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. Considering the low reactor pressure and the availability of the low pressure coolant systems during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate RCIC OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate RCIC OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the RCICS inoperable, a seven-day period to return the system to service is justified based on the availability of the HPCIS to cool the core and upon consideration that the average risk associated with failure of the RCICS to cool the core when required is not increased.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the RCICS will be OPERABLE when required.

3.5.G Automatic Depressurization System (ADS)

This specification ensures the OPERABILITY of the ADS under all conditions for which the depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low-pressure coolant injection (LPCI) and the core spray subsystems can operate to protect the fuel barrier. Note that this specification applies only to the automatic feature of the pressure relief system.

Specification 3.6.D specifies the requirements for the pressure relief function of the valves. It is possible for any number of the valves assigned to the ADS to be incapable of performing their ADS functions because of instrumentation failures, yet be fully capable of performing their pressure relief function.

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

ENCLOSURE 4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 185 TO FACILITY OPERATING LICENSE NO. DPR-33,

AMENDMENT NO. 198 TO FACILITY OPERATING LICENSE NO. DPR-52,

AND AMENDMENT NO. 157 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

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

1.0 INTRODUCTION

By letter dated October 30, 1990, the Tennessee Valley Authority (the licensee) submitted a request for changes to the Browns Ferry Nuclear Plant, Units 1, 2, and 3 Technical Specifications (TS). The proposed amendment would change the BFN technical specifications administratively and revise the bases section for flood protection to be consistent with the FSAR. The changes are being made to resolve open issues from NRC inspection reports, to resolve an open item in an NRC safety evaluation, and to correct errors in previous technical specification submittals and implementation.

2.0 EVALUATION

Each change is itemized below, followed by the staff evaluation.

Change 1 - Def. 1.0.II (Dose Equivalent I-131) was added to the technical specifications by Amendments 132, 128, and 103 to Units 1, 2, and 3 respectively. The approved definition included the units "micro-Curie per gram" for the concentration of I-131. The current technical specifications for all three units erroneously have "mCi/gm" for the concentration of I-131. This change corrects the technical specifications to be as approved by the NRC. The staff finds this acceptable.

Change 2 - Note 2 is being deleted from the Notes for Table 3.2.A for all three units. Note 2 has been in the notes since the original issuance of the Unit 1 technical specifications (TS). However, this note has never been attached to any item in the table. TVA has researched the similar sections of TSs for six other Boiling Water Reactors (BWRs) with custom TSs and has not found a similar note which requires testing of other channels. Also, the General Electric BWR Standard TSs (NUREG 0123) do not have a similar note or requirement for isolation actuation instrumentation in Table 3.3.2-1. Operations takes the actions in Note 1 whenever instrument channels are tripped so Note 2 is unnecessary. Functional testing of the instrument channels within the required surveillance intervals provides reasonable assurance that the instrument channels are OPERABLE. Additional testing prior to tripping a channel is unnecessary. The note is therefore being deleted from the table. The staff finds this acceptable.

Note 5 is also being deleted from the Notes for Table 3.2.A. This note appeared in the original technical specifications for Unit 1 and was associated with "Instrument Channel - Reactor High Water Level." When the Unit 2 technical specifications were issued (Amendment 3 for Unit 1), this reactor high water level setpoint was deleted because it was not required for BWRs similar to BFN. However, the note was never deleted from the Notes for Table 3.2.A. This change removes this unnecessary note for all three units. The staff finds this acceptable.

Change 3 - Amendments 108, 102, and 75 for Units 1, 2, and 3, respectively, were issued by NRC on August 13, 1984 and added Note 13 to Table 3.2.A (Instrument Channel - High Radiation Main Steam Line Tunnel). TVA did not receive the amendment until about August 22, 1984. On August 23, 1984, TVA sent a letter (TS 199) to NRC requesting another change to the same page in Table 3.2.A. Because the amendments approved August 13, 1984 had not been received when the letter TS 199 was in the approval cycle, it went to NRC without the reference to Note 13. NRC approved the change requested in TS 199 on August 19, 1986 (Amendment 125 for Unit 2) utilizing the "old" version of the page from Table 3.2.A. Therefore, the reference to Note 13 was inadvertently deleted. The staff finds the proposed correction to be acceptable.

That same August 23, 1984 submittal included the "<" symbol in front of the "3" for Unit 2. Amendment 102 for Unit 2 did not include this symbol so it should be deleted. Units 1 and 3 are correct as is. Unit 2 must be revised to match Units 1 and 3 and to meet the intent of Amendment 102. The staff finds this acceptable.

Change 4 - A typographical error exists in the range of the noble gas monitors as currently listed. The units currently listed (Ci/cc) would be too high and would not provide the required monitoring range information. The error exists because TVA's submittal to NRC dated June 20, 1989, TS 266 Supplement 1 - Correction to Tables 3.2.F and 4.2.F, erroneously omitted the "μ" symbol. This change would reflect the correct range (μCi/cc) of the instruments. The staff finds this acceptable.

Change 5 - The calibration frequency for instruments PI-3-74A&B for Unit 2 is being corrected to be once per six months. Amendment 167 to the BFN Unit 2 technical specifications was approved by NRC on July 7, 1989. That amendment revised the calibration frequency for the reactor pressure instruments (Page 3.2./4.2-54) to a more conservative interval of 6 months. This was based on the recommendation of Tobar, Inc., the manufacturer of these instruments. The staff finds this acceptable.

Amendment 171 to the BFN Unit 2 technical specifications was approved by NRC on August 22, 1989. An overleaf page for this amendment was Page 3.2/4.2-54, the page which includes the reactor pressure instruments. The clean pages for Amendment 171 were sent to NRC on July 25, 1989. The BFN technical specification clerk had apparently not yet received Amendment 167 and therefore sent a clean Page 3.2/4.2-54 with the old calibration frequency of 12 months on it.

When Amendment 171 was issued, it had the incorrect calibration frequency (12 months) which had been transmitted on the clean page. This proposed change revises Pages 3.2/4.2-54 to correctly indicate that the calibration frequency for the Reactor Pressure instruments is 6 months as approved by Amendment 167. The staff finds this acceptable.

Change 6 - Table 4.2.L (Anticipated Transient Without Scram [ATWS] - Recirculation Pump Trip [RPT] Instrumentation Surveillance) is being revised to correct the title and to incorporate the correct instrument numbers into the table. The staff finds this acceptable.

Change 7 - The statement in the bases for Technical Specification 3.2 that "...however, the plant flood protection is always in place and does not depend in any way on advanced warning..." is being deleted to agree with the plant Final Safety Analysis Report (FSAR). The flood doors to the reactor and radwaste buildings are the flood protection referred to as always being in place in the technical specification 3.2 bases. The FSAR was revised in 1987 by Amendment 5 to reflect the practice of leaving the flood doors open under normal circumstances. The FSAR had previously indicated that these doors are normally closed. The Unreviewed Safety Question Determination in support of the FSAR revision concluded that because of the constant surveillance and control exercised by TVA over the Tennessee River and the relatively short amount of time required to close the flood doors, leaving them normally open would not degrade plant flood protection. Administrative controls are in place to ensure the flood doors are closed when they are needed to provide flood protection. The staff finds this acceptable.

Changes 8 and 9 - The paragraphs in Bases Section 3.5.E (HPCI) and 3.5.F (RCIC) regarding net positive suction head for the HPCI and RCIC systems were inadvertently deleted by TVA when TS 274 was submitted to NRC. Amendments 173, 176, and 144 were therefore approved for Units 1, 2, and 3 respectively with these paragraphs deleted. They are still applicable and are therefore being re-inserted. The staff finds this acceptable.

Change 10 - A statement is being added to Bases Section 3.6.F/4.6.F (Recirculation Pump Operation) to note the intention to scram the reactor if the operating recirculation pump trips when in single loop operation. TVA's committment to make this change was documented in NRC's safety evaluation supporting Unit 2 Amendment 174 (TAC 73435). The staff finds this acceptable.

Change 11 - Paragraph 3.7.A.1.F (Primary Containment) for Unit 1 only is being revised to correct the "greater than" symbol (>) to a "less than" symbol (<) to reflect the requirement to depressurize the reactor to less than 200 psig. This change corrects a typographical error and makes this paragraph consistent with the equivalent Unit 2 and 3 requirements. The staff finds this acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 27049). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: D. H. Moran

Date: August 23, 1991