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November 18, 1988

POSTED
Amdt. 154 to DPR-52
See Correction Letter of 1-13-89

12)

Docket Nos. 50-259/260/296

Mr. S. A. White
Senior Vice President, Nuclear Power
Tennessee Valley Authority
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1101 Market Street
Chattanooga, Tennessee 37402-2801

SBlack
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ARM/LFMB
BFN Rdg. File

Dear Mr. White:

SUBJECT: CHANGES TO DEFINITIONS OF MODE OF OPERATION, CORE ALTERATION, AND REACTOR CONDITIONS FOR BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3 (TAC 00068, 00069, 00070) (TS 240)

The Commission has issued the enclosed Amendment Nos. 158, 154, and 129to 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 May 31, 1988. The amendments modify the Technical Specifications by changing the definitions of Mode of Operations, Core Alteration, and Reactor Conditions.

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

Sincerely,

Original Signed by

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Special Projects

Enclosures:
1. Amendment No. 158 to License No. DPR-33
2. Amendment No. 154 to License No. DPR-52
3. Amendment No. 129 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

MW20080

OFC	:OSP:TVA/LA	:TVA/TP	:OSP:TVA/PM	: OGC	:TVA:AD/P	:	:
NAME	:MSimms:as	:BDLiaw	:GGears	:	:SBlack	:	:
DATE	:9/16/88	: /7/88	:9/2/88	:11/17/88	:11/8/88	:	:

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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.154
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 May 31, 1988 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. 154, 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 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne Black

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 18, 1988

Correction Letter
of 1-13-89

ATTACHMENT TO LICENSE AMENDMENT NO. 154

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf pages* are provided to maintain document completeness.

REMOVE	INSERT
1.0-1	1.0-1*
1.0-2	1.0-2
1.0-3	1.0-3
1.0-4	1.0-4
1.0-5	1.0-5*
1.0-6	1.0-6*
1.0-7	1.0-7
1.0-8	1.0-8*
3.1/4.1-1	3.1/4.1-1
3.1/3.2-2	3.1/4.1-2*
3.2/4.2-33	3.2/4.2-33
3.3/4.3-5	3.3/4.3-5
3.3/4.3-6	3.3/4.3-6*
3.3/4.3-11	3.3/4.3-11
3.3/4.3-12	3.3/4.3-12
3.5/4.5-1	3.5/4.5-1
3.5/4.5-2	3.5/4.5-2
3.5/4.2-2	3.5/4.5-3*
3.5/4.5-4	3.5/4.5-4
3.5/4.5-7	3.5/4.5-7
3.5/4.5-8	3.5/4.5-8
3.5/4.5-9	3.5/4.5-9
3.5/4.5-10	3.5/4.5-10
3.5/4.5-14	3.5/4.5-14*
3.5/4.5-15	3.5/4.5-15
3.6/4.6-5	3.6/4.6-5
3.6/4.6-6	3.6/4.6-6
3.6/4.6-7	3.6/4.6-7
3.6/4.6-8	3.6/4.6-8
3.6/4.6-9	3.6/4.6-9*
3.6/4.6-10	3.6/4.6-10
3.6/4.6-11	3.6/4.6-11
3.6/4.6-12	3.6/4.6-12
3.7/4.7-1	3.7/4.7-1*
3.7/4.7-2	3.7/4.7-2
3.7/4.7-15-3.7/4.7-15a	3.7/4.7-15
3.7/4.7-19	3.7/4.7-19
3.7/4.7-20	3.7/4.7-20

REMOVE	INSERT
3.7/4.7-21	3.7/4.7-21*
3.7/4.7-22	3.7/4.7-22
3.9/4.9-3	3.9/4.9-3*
3.9/4.9-4	3.9/4.9-4
3.9/4.9-9	3.9/4.9-9
3.9/4.9-10	3.9/4.9-10
3.9/4.9-11	3.9/4.9-11
3.9/4.9-12	3.9/4.9-12*
3.9/4.9-13	3.9/4.9-13*
3.9/4.9-14	3.9/4.9-14
3.9/4.9-15	3.9/4.9-15

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit requires unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

B. Limiting Safety System Settings (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

C. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

1. In the event a Limiting Condition for Operation and/or associated requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours unless corrective measures are completed that permit operation under the permissible discovery or until the reactor is placed in an operational condition in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications. This provides actions to be taken for circumstances not directly provided for in the specifications and where occurrence would violate the intent of the specification. For example, if a specification calls for two systems (or subsystems) to be operable and provides for explicit requirements if one system (or subsystem) is inoperable, then if both systems (or subsystems) are inoperable the unit is to be in at least Hot Standby in 6 hours and in Cold Shutdown within the following 30 hours if the inoperable condition is not corrected.

2. When a system, subsystem, train, component, or device is determined to be inoperable solely because its onsite power source is inoperable, or solely because its offsite power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition For Operation, provided:

   (1) its corresponding offsite or diesel power source is operable; and (2) all of its redundant system(s), subsystem(s), train(s), component(s), and device(s) are operable, or likewise satisfy these requirements. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least Hot Standby within 6 hours, and in at least Cold Shutdown within the following 30 hours. This definition is not applicable in Cold Shutdown or Refueling. This provision describes what additional conditions must be satisfied to permit operation to continue consistent with the specifications for power sources, when an offsite or onsite power source is not operable. It specifically prohibits operation when one division is inoperable because its offsite or diesel power source is inoperable and a system, subsystem, train, component, or device in another division is inoperable for another reason. This provision permits the requirements associated with individual systems, subsystems, trains, components, or devices to be consistent with the requirements of the associated electrical power source. It allows operation to be governed by the time limit of the requirements associated with the Limiting Condition For Operation for the offsite or diesel power source, not the individual requirements for each system, subsystem, train, component, or device that is determined to be inoperable solely because of the inoperability of its offsite or diesel power source.

D. PRIOR TO STARTUP - Prior to withdrawing the first control rod for the purpose of making the reactor critical.

E. Operable - Operability - A system, subsystem, train, component, or device shall be operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

F. Operating - Operating means that a system or component is performing its intended functions in its required manner.

G. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

1.0 DEFINITIONS (Cont'd)

H. Reactor Power Operation - Reactor power operation is any operation in the STARTUP/HOT STANDBY or RUN MODE with the reactor critical and above 1 percent rated power.

I. STARTUP CONDITION - The reactor is in the STARTUP CONDITION when the withdrawal of control rods for the purpose of making the reactor critical has begun, reactor power is less than or equal to 1 percent of rated, and the reactor is in the STARTUP/HOT STANDBY MODE.

J. HOT STANDBY CONDITION - The reactor is in the HOT STANDBY CONDITION when reactor power is less than or equal to 1 percent of rated. The reactor is in the STARTUP/HOT STANDBY MODE, and the reactor is not in the STARTUP CONDITION. The reactor coolant temperature may be greater than 212° F.

Note that a HOT STANDBY CONDITION cannot exist simultaneously with a STARTUP CONDITION due to the difference in intent. A HOT STANDBY CONDITION exists when the reactor mode switch is placed in the STARTUP/HOT STANDBY position (for example, to comply with an LCO) and power level has been reduced to 1 percent or lower. Anytime control rods are being withdrawn for the purpose of increasing reactor power level, the reactor mode switch has been placed in the STARTUP/HOT STANDBY position, and reactor power level is at or below one percent, a STARTUP CONDITION exists.

K. SHUTDOWN CONDITION - The reactor is in the SHUTDOWN CONDITION when the reactor is in the Shutdown or Refuel Mode.

1. HOT SHUTDOWN CONDITION - The reactor is in the HOT SHUTDOWN CONDITION when reactor coolant temperature is greater than 212° F and the reactor is in the SHUTDOWN CONDITION.

2. COLD SHUTDOWN CONDITION - The reactor is in the COLD SHUTDOWN CONDITION when reactor coolant temperature is equal to or less than 212° F and the reactor is in the SHUTDOWN CONDITION.

L. COLD CONDITION - The reactor is in the COLD CONDITION when reactor coolant temperature is equal to or less than 212° F in any Mode of Operation (except as defined in K.2 above).

M. Mode of Operation - The reactor mode switch position determines the Mode of Operation of the reactor when there is fuel in the reactor vessel, except that the Mode of Operation may remain unchanged when the reactor mode switch is temporarily moved to another position as permitted by the notes. When there is no fuel in the reactor vessel, the reactor is considered not to be in any Mode of Operation or operational condition. The reactor mode switch may then be in any position or may be INOPERABLE.

1. Startup/Hot Standby Mode - The reactor is in the STARTUP/HOT STANDBY MODE when the reactor mode switch is in the "STARTUP/HOT STANDBY" position. This is often referred to as just the STARTUP MODE.

2. Run Mode - The reactor is in the Run Mode when the reactor mode switch is in the "Run" position.

3. Shutdown Mode - The reactor is in the Shutdown Mode when the reactor mode switch is in the "Shutdown" position.(1)(2)(3)(4)

4. Refuel Mode - The reactor is in the Refuel Mode when the reactor mode switch is in the "Refuel" position.(1)

(1) The reactor mode switch may be placed in any position to perform required tests or maintenance authorized by the shift operations supervisor, provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

(2) The reactor mode switch may be placed in the "Refuel" position while a single control rod drive is being removed from the reactor pressure vessel per specification 3.10.A.5 provided that reactor coolant temperature is equal to or less than 212° F.

(3) The reactor mode switch may be placed in the "Refuel" position while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock is operable.

(4) The reactor mode switch may be placed in the "Startup/Hot Standby" position and withdrawal of selected control rods is permitted for the purpose of determining the operability of the RSCS and RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

N. Rated Power - Rated power refers to operation at a reactor power of 3,293 MWt; this is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power. Design power, the power to which the safety analysis applies, corresponds to 3,440 MWt.

O. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All nonautomatic containment isolation valves on lines connected to the reactor coolant systems or containment which are not required to be open during accident conditions are closed. These valves may be opened to perform necessary operational activities.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or each line which contains an inoperable isolation valve is isolated as required by specification 3.7.D.2.

4. All blind flanges and manways are closed.

P. Secondary Containment Integrity

1. Secondary containment integrity means that the reactor building is intact and the following conditions are met:

a) At least one door in each access opening to the turbine building, control bay and out-of-doors is closed.

b) The standby gas treatment system is operable and can maintain 0.25 inches of water negative pressure in those areas where secondary containment integrity is stated to exist.

c) All secondary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an operable secondary containment automatic isolation system, or

2. Closed by at least one secondary containment automatic isolation valve deactivated in the isolated position.

2. Reactor zone secondary containment integrity means the unit reactor building is intact and the following conditions are met:

a) At least one door between any opening to the turbine building, control bay and out-of-doors is closed.

P. Secondary Containment Integrity (Cont'd)

2. b) The standby gas treatment system is operable and can maintain 0.25 inches water negative pressure on the unit zone.

c) All the unit reactor building ventilation system penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an operable reactor building ventilation system automatic isolation system, or

2. Closed by at least one reactor building ventilation system automatic isolation valve deactivated in the isolated position.

If it is desirable for operational considerations, a reactor zone may be isolated from the other reactor zones and the refuel zone by maintaining at least one closed door in each common passageway between zones.* Reactor zone safety-related features are not compromised by openings between adjacent units or refuel zone, unless it is desired to isolate a given zone.

3. Refuel zone secondary containment integrity means the refuel zone is intact and the following conditions are met:

a) At least one door in each access opening to the out-of-doors is closed.

b) The Standby Gas Treatment System is operable and can maintain 0.25 inches water negative pressure on the refuel zone.

c) All refuel zone ventilation system penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an operable refuel zone ventilation system automatic isolation system, or

2. Closed by at least one refuel zone ventilation system automatic isolation valve deactivated in the isolated position.

If it is desirable for operational considerations, the refuel zone may be isolated from the reactor zones by maintaining all hatches in place between the refuel floor and reactor zones and at least one closed door in each access between the refuel zone and the reactor building.* Refuel zone safety-related features are not compromised by openings between the reactor building unless it is desired to isolate a given zone.

*To effectively control zone isolation, all accesses to the affected zone will be locked or guarded to prevent uncontrolled passage to the unaffected zones.

Q. Operating Cycle - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

R. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the unit after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

S. CORE ALTERATION - The addition, removal, relocation, or movement of fuel, sources, incore instruments, or reactivity controls within the reactor pressure vessel with the head removed and fuel in the vessel. Normal control rod movement with the control rod drive hydraulic system is not defined as a Core Alteration. Normal movement of in-core instrumentation and the traversing in-core probe is not defined as a Core Alteration. Suspension of Core Alterations shall not preclude completion of the movement of a component to a safe conservative position.

T. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

U. Thermal Parameters

1. Minimum Critical Power Ratio (MCPR) - Minimum Critical Power Ratio (MCPR) is the value of the critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power.

2. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

3. Core Maximum Fraction of Limiting Power Density (CMFLPD) - The highest ratio, for all fuel types in the core, of the maximum fuel rod power density (kW/ft) for a given fuel type to the limiting fuel rod power density (kW/ft) for that fuel type.

4. Average Planar Linear Heat Generation Rate (APLHGR) - The Average Planar Heat Generation Rate is applicable to a specific planar height and is equal to the sum of the linear heat generation rates for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

10. Logic - A logic is an arrangement of relays, contacts, and other components that produces a decision output.

    (a) Initiating - A logic that receives signals from channels and produces decision outputs to the actuation logic.

    (b) Actuation - A logic that receives signals (either from initiation logic or channels) and produces decision outputs to accomplish a protective action.

11. Channel Calibration - Shall be the adjustment, as necessary, of the channel output such that it responds with necessary range and accuracy to known values of the parameters which the channel monitors. The channel calibration shall encompass the entire channel including alarm and/or trip functions and shall include the channel functional test. The channel calibration may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated. Non-calibratable components shall be excluded from this requirement, but will be included in channel functional test and source check.

12. Channel Functional Test - Shall be:

    a. Analog Channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

    b. Bistable Channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

13. Source Check - Shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source or multiple of sources.

LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Protection System

Applicability

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective

To assure the operability of the reactor protection system.

Specification

A. When there is fuel in the vessel, the setpoints, minimum number of trip systems, and minimum number of instrument channels that must be OPERABLE for MODE OF OPERATION shall be as given in Table 3.1.A.

B. Two RPS power monitoring channels for each inservice RPS MG sets or alternate source shall be OPERABLE.

1. With one RPS electric power monitoring channel for inservice RPS MG set or alternate power supply INOPERABLE, restore the INOPERABLE channel to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.1 Reactor Protection System

Applicability

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.A and 4.1.B, respectively.

B. The RPS power monitoring system instrumentation shall be determined OPERABLE:

1. At least once per 6 months by performance of channel functional tests.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.1.B. (Cont'd) 2. With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply INOPERABLE, restore at least one to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.	

NOTES FOR TABLE 3.2.F

(1) From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding 30 days unless such instrumentation is sooner made OPERABLE.

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

(3) If the requirements of notes (1) and (2) cannot be met, and if one of the indications cannot be restored in (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN CONDITION within 24 hours.

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

(5) From and after the date that both the acoustic monitor and the temperature indication on any one valve fails to indicate in the control room, continued operation is permissible during the succeeding 30 days, unless one of the two monitoring channels is sooner made OPERABLE. If both the primary and secondary indication on any SRV tailpipe is INOPERABLE, the torus temperature will be monitored at least once per shift to observe any unexplained temperature increase which might be indicative of an open SRV.

(6) A channel consists of eight sensors, one from each alternating torus bay. Seven sensors must be OPERABLE for the channel to be OPERABLE.

(7) When one of these instruments is INOPERABLE for more than seven days, in lieu of any other report required by Specification 6.7.2, prepare and submit a Special Report to the Commission pursuant to Specification 6.7.3 within the next seven days outlining the action taken, the cause of inoperability, and the plans and schedule for restoring the system to OPERABLE status.

(8) With the plant in REACTOR POWER OPERATION, STARTUP CONDITION, HOT STANDBY CONDITION OR HOT SHUTDOWN CONDITION and with the number of OPERABLE channels less than the required OPERABLE channels, either restore the INOPERABLE channel(s) to OPERABLE status within 72 hours, or initiate the preplanned alternate method of monitoring the appropriate parameter.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.3.B. Control Rods

1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the SHUTDOWN CONDITION when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

2. The control rod drive housing support system shall be in place during reactor REACTOR POWER OPERATION or when the reactor coolant system is pressurized above atmospheric with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

4.3.B. Control Rods

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

   a. Verify that the control rod is following the drive by observing a response in the nuclear instrumentation each time a rod is moved when the reactor is operating above the preset power level of the RSCS.

   b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

3.a Whenever the reactor is in the startup or run modes below 20% rated power, the Rod Sequence Control System (RSCS) shall be OPERABLE, except that the RSCS constraints may be suspended by means of the individual rod bypass switches for
1 - special criticality tests, or

2 - control rod scram timing per 4.3.C.1.

When RSCS is bypassed on individual rods for these exceptions, RWM must be operable per 3.3.B.3.b and a second party verification may not be used in lieu of RWM.

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

3.a.1 The Rod Sequence Control System (RSCS) shall be demonstrated to be OPERABLE for a reactor startup by the following checks:

a. Performance of the comparator check of group notch circuits within 8 hours prior to control rod withdrawal for the purpose of making the reactor critical

b. Selecting and attempting to withdraw an out-of-sequence control rod after withdrawal of the first insequence control rod.

c. Attempting to withdraw a control rod more than one notch prior to other control rod movement after the group notch mode is automatically initiated.

3.a.2 The Rod Sequence Control System (RCS) shall be demonstrated to be OPERABLE for a reactor shutdown by the following checks:

a. Performance of the comparator check of the group notch circuits within 8 hours prior to automatic initiation of the group notch mode.

LIMITING CONDITIONS FOR OPERATION

3.3.C. Scram Insertion Times

2. The average of the scram insertion times for the three fastest OPERABLE control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

a. The maximum scram insertion time for 90% insertion of any OPERABLE control rod shall not exceed 7.00 seconds.

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be placed in SHUTDOWN CONDITION until the cause has been determined and corrective actions have been taken as appropriate.

SURVEILLANCE REQUIREMENTS

4.3.C. Scram Insertion Times

2. At 16-week intervals, 10% of the OPERABLE control rod drives shall be scram-timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

D. Reactivity Anomalies

During the STARTUP test program and STARTUP following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.E. If Specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN CONDITION within 24 hours.	4.3.E. Surveillance requirements are as specified in 4.3.C and .D above.
F. Scram Discharge Volume (SDV)	F. Scram Discharge Volume (SDV)
1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.	1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation. 1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE monthly.
2. In the event any SDV drain or vent valve becomes INOPERABLE, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.	2. When it is determined that any SDV drain or vent valve is INOPERABLE, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.
3. If redundant drain or vent valves become INOPERABLE, the reactor shall be in HOT STANDBY CONDITION within 24 hours.	3. No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and containment cooling systems.

Objective

To assure the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. The CSS shall be OPERABLE:

(1) PRIOR TO STARTUP from a COLD CONDITION, or

(2) when there is irradiated fuel in the vessel and when the reactor vessel pressure is greater than atmospheric pressure, except as specified in Specification 3.5.A.2.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the surveillance requirements of the core and containment cooling systems when the corresponding limiting condition for operation is in effect.

Objective

To verify the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. Core Spray System Testing.

	Item	Frequency
a.	Simulated Automatic Actuation test	Once/ Operating Cycle
b.	Pump Operability	Once/ month
c.	Motor Operated Valve Operability	Once/ month
d.	System flow rate: Each loop shall deliver at least 6250 gpm against a system head corresponding to a	Once/3 months

LIMITING CONDITIONS FOR OPERATION

3.5.A Core Spray System (CSS)

2. If one CSS loop is INOPERABLE, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS loop and the RHR system (LPCI mode) and the diesel generators are OPERABLE.

3. If Specification 3.5.A.1 or Specification 3.5.A.2 cannot be met, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one core spray loop with one OPERABLE pump and associated diesel generator shall be OPERABLE, except with the reactor vessel head removed as specified in 3.5.A.5 or PRIOR TO STARTUP as specified in 3.5.A.1.

SURVEILLANCE REQUIREMENTS

4.5.A Core Spray System (CSS)

4.5.A.1.d (Cont'd)

105 psi differential pressure between the reactor vessel and the primary containment.

e. Check Valve — Once/Operating Cycle

2. When it is determined that one core spray loop is INOPERABLE, at a time when operability is required, the other core spray loop, the RHRS (LPCI mode), and the diesel generators shall be demonstrated to be OPERABLE immediately. The OPERABLE core spray loop shall be demonstrated to be OPERABLE daily thereafter.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.A Core Spray System (CSS) 5. When irradiated fuel is in the reactor vessel and the reactor vessel head is removed, core spray is not required provided work is not in progress which has the potential to drain the vessel, provided the fuel pool gates are open and the fuel pool is maintained above the low level alarm point, and provided one RHRSW pump and associated valves supplying the standby coolant supply are OPERABLE.	

LIMITING CONDITIONS FOR OPERATION

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

1. The RHRS shall be OPERABLE:

   (1) PRIOR TO STARTUP from a COLD CONDITION; or

   (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

2. With the reactor vessel pressure less than 105 psig, the RHR may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.

SURVEILLANCE REQUIREMENTS

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

1. a.	Simulated Automatic Actuation Test	Once/ Operating Cycle
b.	Pump Operability	Once/ month
c.	Motor Operated valve operability	Once/ month
d.	Pump Flow Rate	Once/3 months
e.	Testable Check Valve	Once/ Operating Cycle

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12,000 gpm against an indicated system pressure of 250 psig.

2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

LIMITING CONDITIONS FOR OPERATION

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

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

11. When there is irradiated fuel in the reactor and the reactor vessel pressure is greater than atmospheric, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered INOPERABLE if cross-connect capability can be restored to service within 5 hours.)

SURVEILLANCE REQUIREMENTS

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

8. No additional surveillance required.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE monthly.

10. No additional surveillance required.

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE monthly when the cross-connect capability is required.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)	4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)
12. If three RHR pumps or associated heat exchangers located on the unit cross-connection in the adjacent units are INOPERABLE for any reason (including valve inoperability, pipe break, etc.), the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pump and associated diesel generator are OPERABLE.	12. When it is determined that three RHR pumps or associated heat exchangers located on the unit cross-connection in the adjacent units are INOPERABLE at a time when operability is required, the remaining RHR pump and associated heat exchanger on the unit cross-connection and the associated diesel generator shall be demonstrated to be OPERABLE immediately and every 15 days thereafter until the INOPERABLE pump and associated heat exchanger are returned to normal service.
13. If RHR cross-connection flow or heat removal capability is lost, the unit may remain in operation for a period not to exceed 10 days unless such capability is restored.	13. No additional surveillance required.
14. All recirculation pump discharge valves shall be OPERABLE PRIOR TO STARTUP (or closed if permitted elsewhere in these specifications).	14. All recirculation pump discharge valves shall be tested for operability during any period of COLD SHUTDOWN CONDITION exceeding 48 hours, if operability tests have not been performed during the preceding 31 days.

LIMITING CONDITIONS FOR OPERATION

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

1. PRIOR TO STARTUP from a COLD CONDITION, 9 RHRSW pumps must be OPERABLE, with 7 pumps (including one of pumps D1, D2, B2 or B1) assigned to RHRSW service and 2 automatically starting pumps assigned to EECW service.

SURVEILLANCE REQUIREMENTS

4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)

1. a. Each of the RHRSW pumps normally assigned to automatic service on the EECW headers will be tested automatically each time the diesel generators are tested. Each of the RHRSW pumps and all associated essential control valves for the EECW headers and RHR heat exchanger headers shall be demonstrated to be OPERABLE once every three months.

   b. Annually each RHRSW pump shall be flow-rate tested. To be considered OPERABLE, each pump shall pump at least 4500 gpm through its normally assigned flow path.

LIMITING CONDITIONS FOR OPERATION

3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Cont'd)

2. During REACTOR POWER OPERATION, RHRSW pumps must be OPERABLE and assigned to service as indicated in Table 3.5-1 for the specified time limits.

3. During unit 2 REACTOR POWER OPERATION, any two RHRSW pumps (D1, D2, B1, and B2) normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection must be OPERABLE except as specified in 3.5.C.4 and 3.5.C.5 below.

SURVEILLANCE REQUIREMENTS

4.5.C. RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Cont'd)

2. a. If no more than two RHRSW pumps are INOPERABLE, increased surveillance is not required.

   b. When three RHRSW pumps are INOPERABLE, the remaining pumps, associated essential control valves, and associated diesel generators shall be operated weekly.

   c. When four RHRSW pumps are INOPERABLE, the remaining pumps, associated essential control valves, and associated diesel generators shall be operated daily.

3. Routine surveillance for these pumps is specified in 4.5.C.1.

LIMITING CONDITIONS FOR OPERATION

3.5.E High Pressure Coolant Injection System (HPCIS)

2. If the HPCI system is INOPERABLE, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS (LPCI), and RCICS are operable.

3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 122 psig or less within 24 hours.

F. Reactor Core Isolation Cooling System (RCICS)

1. The RCICS shall be OPERABLE:

(1) prior to STARTUP from a Cold Condition; or

(2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 122 psig, except as specified in 3.5.F.2.

SURVEILLANCE REQUIREMENTS

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

4.5.E.1 (Cont'd)

e. Flow Rate at 150 psig — Once/operating cycle

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

2. When it is determined that the HPCIS is INOPERABLE the ADS actuation logic, the RCICS, the RHRS (LPCI), and the CSS shall be demonstrated to be OPERABLE immediately. The RCICS and ADS logic shall be demonstrated to be OPERABLE daily thereafter.

F. Reactor Core Isolation Cooling System (RCICS)

1. RCIC Subsystem testing shall be performed as follows:

a. Simulated Automatic Actuation Test — Once/operating cycle

b. Pump Operability — Once/month

c. Motor-Operated Valve Operability — Once/month

LIMITING CONDITIONS FOR OPERATION

3.5.F. Reactor Core Isolation Cooling System (RCICS)

2. If the RCICS is INOPERABLE, the reactor may remain in operation for a period not to exceed 7 days if the HPCIS is OPERABLE during such time.

3. If Specifications 3.5.F.1 or 3.5.F.2 are not met, an orderly shutdown shall be initiated and the reactor shall be depressurized to less than 122 psig within 24 hours.

G. Automatic Depressurization System (ADS)

1. Four of the six valves of the Automatic Depressurization System shall be OPERABLE:

   (1) PRIOR TO STARTUP from a COLD CONDITION, or

SURVEILLANCE REQUIREMENTS

4.5.F. Reactor Core Isolation Cooling System (RCICS)

4.5.F.1 (Cont'd)

d.	Flow Rate at normal reactor vessel operating pressure	Once/3 months
e.	Flow Rate at 150 psig	Once/ operating cycle

The RCIC pump shall deliver at least 600 gpm during each flow test.

2. When it is determined that the RCICS is INOPERABLE, the HPCIS shall be demonstrated to be OPERABLE immediately.

G. Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:

   a. A simulated automatic actuation test shall be performed prior to STARTUP after each

LIMITING CONDITIONS FOR OPERATION

3.6.B. Coolant Chemistry

1. PRIOR TO STARTUP and at steaming rates less than 100,000 lb/hr, the following limits shall apply.

   a. Conductivity, mmho/cm at 25°C — 2.0

   b. Chloride, ppm — 0.1

2. At steaming rates greater than 100,000 lb/hr, the following limits shall apply.

   a. Conductivity, mmho/cm at 25°C — 1.0

   b. Chloride, ppm — 0.2

SURVEILLANCE REQUIREMENTS

4.6.B. Coolant Chemistry

1. Reactor coolant shall be continuously monitored for conductivity.

   a. Whenever the continuous conductivity monitor is inoperable and the condensate demineralizers are bypassed, a sample of reactor coolant shall be analyzed for conductivity every 4 hours. If the condensate demineralizers are in service, a sample of reactor coolant shall be analyzed for conductivity every 8 hours.

   b. Once a week the continuous monitor shall be checked with an in-line flow cell. This in-line conductivity calibration shall be performed every 24 hours whenever the reactor coolant conductivity is >1.0 mmho/cm at 25°C.

2. During startup prior to pressurizing the reactor above atmospheric pressure, measurements of reactor water quality shall be performed to show conformance with 3.6.B.1 of limiting conditions.

LIMITING CONDITIONS FOR OPERATION

3.6.B. Coolant Chemistry

3. At steaming rates greater than 100,000 lb/hr, the reactor water quality may exceed Specification 3.6.B.2 only for the time limits specified below. Exceeding these time limits of the following maximum quality limits shall be cause for placing the reactor in the COLD SHUTDOWN CONDITION.

   a. Conductivity time above 1 mmho/cm at 25°C - 2 weeks/year.
      Maximum Limit 10 mmho/cm at 25°C

   b. Chloride concentration time above 0.2 ppm - 2 weeks/year.
      Maximum Limit - 0.5 ppm.

   c. The reactor shall be placed in the SHUTDOWN CONDITION if pH <5.6 or >8.6 for a 24-hour period.

SURVEILLANCE REQUIREMENTS

4.6.B. Coolant Chemistry

3. Whenever the reactor is operating (including HOT STANDBY CONDITION) measurements of reactor water quality shall be performed according to the following schedule:

   a. Chloride ion content shall be measured at least once every 96 hours.

   b. Chloride ion content shall be measured at least every 8 hours whenever reactor conductivity is >1.0 mmho/cm at 25°C.

   c. A sample of primary coolant shall be measured for pH at least once every 8 hours whenever the reactor coolant conductivity is >1.0 mmho/cm at 25°C.

LIMITING CONDITIONS FOR OPERATION

3.6.B. Coolant Chemistry

4. When the reactor is not pressurized, except during the STARTUP CONDITION, the reactor water shall be maintained within the following limits.

   a. Conductivity - 10 mmho/cm at 25°C

   b. Chloride - 0.5 ppm

   c. pH shall be between 5.3 and 8.6.

5. When the time limits or maximum conductivity or chloride concentration limits are exceeded, an orderly shutdown shall be initiated immediately. The reactor shall be brought to the COLD SHUTDOWN CONDITION as rapidly as cooldown rate permits.

6. Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 3.2 mCi/gm of dose equivalent I-131.

SURVEILLANCE REQUIREMENTS

4.6.B. Coolant Chemistry

4. Whenever the reactor is not pressurized, a sample of the reactor coolant shall be analyzed at least every 96 hours for chloride ion content and pH.

5. During equilibrium power operation an isotopic analysis, including quantitative measurements for at least I-131, I-132, I-133, and I-134 shall be performed monthly on a coolant liquid sample.

6. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the equilibrium concentration specified in 3.6.B.6 and one of the following conditions are met:

LIMITING CONDITIONS FOR OPERATION

3.6.B. Coolant Chemistry

3.6.B.6 (Cont'd)

This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient the iodine concentrations shall not exceed 26 mCi/gm whenever the reactor is critical. The reactor shall not be operated more than 5% of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds 26 mCi/gm, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

SURVEILLANCE REQUIREMENTS

4.6.B. Coolant Chemistry

4.6.B.6 (Cont'd)

a. During the STARTUP CONDITION

b. Following a significant power change**

c. Following an increase in the equilibrium off-gas level exceeding 10,000 mCi/sec (at the steam jet air ejector) within a 48-hour period.

d. Whenever the equilibrium iodine limit specified in 3.6.B.6 is exceeded.

The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until a stable iodine concentration below the limiting value (3.2 mCi/gm) is established. However, at least 3 consecutive samples shall be taken in all cases. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration. If the total iodine activity of the sample is below 0.32 mCi/gm, an isotopic analysis to determine equivalent I-131 is not required.

** For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.C. Coolant Leakage	4.6.C. Coolant Leakage
1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.	1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.
b. Anytime the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below.	
c. During the first 24 hours in the RUN mode following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.	

LIMITING CONDITIONS FOR OPERATION

3.6.C Coolant Leakage

2. Both the sump and air sampling systems shall be OPERABLE during REACTOR POWER OPERATION. From and after the date that one of these systems is made or found to be INOPERABLE for any reason, REACTOR POWER OPERATION is permissible only during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

   The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

3.6.D. Relief Valves

1. When more than one relief valves are known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.C Coolant Leakage

2. With the air sampling system INOPERABLE, grab samples shall be obtained and analyzed at least once every 24 hours.

4.6.D. Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. Once during each operating cycle, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

LIMITING CONDITIONS FOR OPERATION

3.6.E. Jet Pumps

1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.D. Relief Valves

3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.

4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

   a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

   b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

   c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

LIMITING CONDITIONS FOR OPERATION

3.6.F Recirculation Pump Operation

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.

2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

3. Steady-state operation with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature of

SURVEILLANCE REQUIREMENTS

4.6.E. Jet Pumps

2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating with the equalizer valve closed, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

4.6.F. Recirculation Pump Operation

1. Recirculation pump speeds shall be checked and logged at least once per day.

2. No additional surveillance required.

3. Before starting either recirculation pump during steady-state operation, check and log the loop discharge temperature and dome saturation temperature.

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 the 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.

LIMITING CONDITIONS FOR OPERATION

3.7.B. Standby Gas Treatment System

* 3. From and after the date that one train of the standby gas treatment system is made or found to be INOPERABLE for any reason, REACTOR POWER OPERATION and fuel handling is permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE, provided that during such 7 days all active components of the other two standby gas treatment trains shall be OPERABLE.

4. If these conditions cannot be met, the reactor shall be placed in a condition for which the standby gas treatment system is not required.

* Not applicable with no fuel in any reactor vessel

SURVEILLANCE REQUIREMENTS

4.7.B. Standby Gas Treatment System

4.7.B.2 (Cont'd)

d. Each train shall be operated a total of at least 10 hours every month.

e. Test sealing of gaskets for housing doors shall be performed utilizing chemical smoke generators during each test performed for compliance with Specification 4.7.B.2.a and Specification 3.7.B.2.a.

3. a. Once per operating cycle automatic initiation of each branch of the standby gas treatment system shall be demonstrated from each unit's controls.

b. At least once per year manual operability of the bypass valve for filter cooling shall be demonstrated.

* c. When one train of the standby gas treatment system becomes INOPERABLE the other two trains shall be demonstrated to be OPERABLE within 2 hours and daily thereafter.

LIMITING CONDITIONS FOR OPERATION

3.7.E. Control Room Emergency Ventilation

* 1. Except as specified in Specification 3.7.E.3 below, both control room emergency pressurization systems shall be OPERABLE at all times when any reactor vessel contains irradiated fuel.

2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

   b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal at a velocity when tested in accordance with ASTM D3803 (130°C, 95% R.H.).

   c. System flow rate shall be shown to be within ±10% design flow when tested in accordance with ANSI N510-1975.

* LCO not applicable until just prior to withdrawing the first control rod for the purpose of making the reactor critical from the unit 2 cycle 5 outage.

SURVEILLANCE REQUIREMENTS

4.7.E Control Room Emergency Ventilation

1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to to be less than 6 inches of water at system design flow rate (± 10%).

2. a. The tests and sample analysis of Specification 3.7.E.2 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

   b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

   c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

   d. Each circuit shall be operated at least 10 hours every month.

LIMITING CONDITIONS FOR OPERATION

3.7.E. Control Room Emergency Ventilation

* 3. From and after the date that one of the control room emergency pressurization systems is made or found to be INOPERABLE for any reason, REACTOR POWER OPERATION or refueling operations is permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE.

* 4. If these conditions cannot be met, reactor shutdown shall be initiated and all reactors shall be in COLD SHUTDOWN within 24 hours for reactor operations and refueling operations shall be terminated within 2 hours.

* LCO not applicable until just prior to withdrawing the first control rod for the purpose of making the reactor critical from the unit 2 cycle 5 outage.

SURVEILLANCE REQUIREMENTS

4.7.E. Control Room Emergency Ventilation

3. At least once every 18 months, automatic initiation of the control room emergency pressurization system shall be demonstrated.

4. During the simulated automatic actuation test of this system (see Table 4.2.G), it shall be verified that the following dampers operate as indicated:

   Close: FCO-150 B, D, E, and F
   Open: FCO-151
   FCO-152

LIMITING CONDITIONS FOR OPERATION

3.7.F. Primary Containment Purge System

1. The primary containment shall be normally vented and purged through the primary containment purge system. The standby gas treatment system may be used when primary containment purge system is INOPERABLE.

2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

   b. The results of laboratory carbon sample analysis shall show ≥85% radioactive methyl iodide removal when tested in accordance with ASTM D3803 (130°C 95% R.H.).

   c. System flow rate shall be shown to be within ±10% of design flow when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS

4.7.F. Primary Containment Purge System

1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8.5 inches of water at system design flow rate (± 10%).

2. a. The tests and sample analysis of Specification 3.7.F.2 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first or after 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

   b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

   c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

LIMITING CONDITIONS FOR OPERATION

3.7.G. Containment Atmosphere Dilution System (CAD)

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

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

   b. A minimum supply of 2,500 gallons of liquid nitrogen per system.

2. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE whenever the reactor is in the RUN MODE.

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

SURVEILLANCE REQUIREMENTS

4.7.G. Containment Atmosphere Dilution System (CAD)

1. System Operability

   a. At least once per month cycle each solenoid operated air/nitrogen valve through at least one complete cycle of full travel and verify that each manual valve in the flow path is open.

   b. Verify that the CAD System contains a minimum supply of 2,500 gallons of liquid nitrogen twice per week.

LIMITING CONDITIONS FOR OPERATION

3.9.A. Auxiliary Electrical Equipment

3.9.A.1.c. (Cont'd)

(4) The Athens 161-kV line is available to the units 1 and 2 shutdown boards through a common station-service transformer when unit 1 is in Cold Shutdown and unit 3 is not claiming the Athens line as an offsite source.

NOTE FOR (3) AND (4):

With no cooling tower pumps or fans running, a cooling tower transformer may be substituted for a common station-service transformer.

SURVEILLANCE REQUIREMENTS

4.9.A. Auxiliary Electrical System

4.9.A.1.b (Cont'd)

(3) On diesel generator breaker trip, the loads are shed from the emergency buses and the diesel output breaker recloses on the autostart signal, the emergency buses are energized with permanently connected loads, the autoconnected emergency loads are energized through load sequencing, and the diesel operates for greater than or equal to five minutes while its generator is loaded with the emergency loads.

c. Once a month the quantity of diesel fuel available shall be logged.

d. Each diesel generator shall be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.

e. Once a month a sample of diesel fuel shall be checked for quality. The quality shall be within acceptable limits specified in Table 1 of the latest revision to ASTM D975 and logged.

LIMITING CONDITIONS FOR OPERATION

3.9.A. Auxiliary Electrical Equipment

2. The reactor shall not be started up (made critical) from the HOT STANDBY CONDITION unless all of the following conditions are satisfied:

   a. At least one offsite power source is available as specified in 3.9.A.1.c.

   b. Three units 1 and 2 diesel generators shall be OPERABLE.

   c. An additional source of power consisting of one of the following:

      (1) A second offsite power source available as specified in 3.9.A.1.c.

      (2) A fourth OPERABLE units 1 and 2 diesel generator.

   d. Requirements 3.9.A.3 through 3.9.A.6 are met.

SURVEILLANCE REQUIREMENTS

4.9.A. Auxiliary Electrical System

2. DC Power System - Unit Batteries (250-V), Diesel-Generator Batteries (125-V) and Shutdown Board Batteries (250-V)

   a. Every week the specific gravity, voltage and temperature of the pilot cell and overall battery voltage shall be measured and logged.

   b. Every three months the measurement shall be made of voltage of each cell to nearest 0.1 volt, specific gravity of each cell, and temperature of every fifth cell. These measurements shall be logged.

   c. At least once every 24 months, a battery rated discharge (capacity) test shall be performed and the voltage, time, and output current measurements shall be logged.

LIMITING CONDITIONS FOR OPERATION

3.9.B. Operation With Inoperable Equipment

3. When one of the units 1 and 2 diesel generator is INOPERABLE, continued REACTOR POWER OPERATION is permissible during the succeeding 7 days, provided that 2 offsite power sources are available as specified in 3.9.A.1.c and all of the CS, RHR (LPCI and containment cooling) systems, and the remaining three units 1 and 2 diesel generators are OPERABLE. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

4. When one units 1 and 2 4-kV shutdown board is INOPERABLE, continued REACTOR POWER OPERATION is permissible for a period of 5 days provided that 2 offsite power sources are available as specified in 3.9.A.1.c and the remaining 4-kV shutdown boards and associated diesel generators, CS, RHR (LPCI and containment cooling) systems, and all 480-V emergency power boards are OPERABLE. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

SURVEILLANCE REQUIREMENTS

4.9.B. Operation With Inoperable Equipment

3. When one of the units 1 and 2 diesel generators is found to be INOPERABLE, all of the CS, RHR (LPCI and containment cooling) systems and the remaining diesel generators and associated boards shall be demonstrated to be OPERABLE immediately and daily thereafter.

4. When one 4-kV shutdown board is found to be INOPERABLE, all remaining 4-kV shutdown boards and associated diesel generators, CS, and RHR (LPCI and containment cooling) systems supplied by the remaining 4-kV shutdown boards shall be demonstrated to be operable immediately and daily thereafter.

LIMITING CONDITIONS FOR OPERATION

3.9.B. Operation With Inoperable Equipment

5. When one of the shutdown buses is INOPERABLE, REACTOR POWER OPERATION is permissible for a period of 7 days.

6. When one of the 480-V diesel auxiliary boards becomes INOPERABLE, REACTOR POWER OPERATION is permissible for a period of 5 days.

7. From and after the date that one of the three 250-V unit batteries and/or its associated battery board is found to be INOPERABLE for any reason, continued REACTOR POWER OPERATION is permissible during the succeeding 7 days. Except for routine surveillance testing, NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period, and the plans to return the failed component to an OPERABLE state.

SURVEILLANCE REQUIREMENTS

4.9.B. Operation With Inoperable Equipment

5. When a shutdown bus is found to be INOPERABLE, all 1 and 2 diesel generators shall be proven OPERABLE immediately and daily thereafter.

6. When one units 1 and 2 diesel auxiliary board is found to be INOPERABLE, the remaining diesel auxiliary board and each unit 1 and 2 diesel generator shall be proven OPERABLE immediately and daily thereafter.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9.B Operation With Inoperable Equipment	
8. From and after the date that one of the 250-V shutdown board batteries and/or its associated battery board is found to be INOPERABLE for any reason, continued REACTOR POWER OPERATION is permissible during the succeeding five days in accordance with 3.9.B.7.	
9. When one division of the logic system is INOPERABLE, continued REACTOR POWER OPERATION is permissible under this condition for seven days, provided the CSCS requirements listed in Specification 3.9.B.3 are satisfied. The NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period, and the plans to return the failed component to an OPERABLE state.	
10. (deleted)	
11. The following limiting conditions for operation exist for the undervoltage relays which start the diesel generators on the 4-kV shutdown boards.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.9.B. Operation With Inoperable Equipment

3.9.B.11 (Cont'd)

a. The loss of voltage relay channel which starts the diesel generator for a complete loss of voltage on a 4-kV shutdown board may be INOPERABLE for 10 days provided the degraded voltage relay channel on that shutdown board is OPERABLE (within the surveillance schedule of 4.9.A.4.b).

b. The degraded voltage relay channel which starts the diesel generator for degraded voltage on a 4-kV shutdown board may be INOPERABLE for 10 days provided the loss of voltage relay channel on that shutdown board is OPERABLE (within the surveillance schedule of 4.9.A.4.b).

c. One of the three phase-to-phase degraded voltage relays provided to detect a degraded voltage on a 4-kV shutdown board may be INOPERABLE for 15 days provided both of the following conditions are satisfied.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

3.9.B. Operation With Inoperable Equipment

3.9.B.11.c. (Cont'd)

1. The other two phase-to-phase degraded voltage relays on that 4-kV shutdown board are OPERABLE (within the surveillance schedule of 4.9.A.4.b).

2. The loss of voltage relay channel on that shutdown board is OPERABLE (within the surveillance schedule of 4.9.A.4.b).

d. The degraded voltage relay channel and the loss of voltage relay channel on a 4-kV shutdown board may be INOPERABLE for 5 days provided the other shutdown boards and undervoltage relays are OPERABLE. (Within the surveillance schedule of 4.9.A.4.b).

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.9.B. Operation With Inoperable Equipment

12. When one 480-V shutdown board is found to be INOPERABLE, the reactor will be placed in the HOT STANDBY CONDITION within 12 hours and COLD SHUTDOWN CONDITION within 24 hours.

13. If one 480-V RMOV board mg set is INOPERABLE, REACTOR POWER OPERATION may continue for a period not to exceed seven days, provided the remaining 480-V RMOV board mg sets and their associated loads remain OPERABLE.

14. If any two 480-V RMOV board mg sets become INOPERABLE, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

15. If the requirements for operating in the conditions specified by 3.9.B.1 through 3.9.B.14 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.9.C. Operation in Cold Shutdown

Whenever the reactor is in COLD SHUTDOWN CONDITION with irradiated fuel in the reactor, the availability of electric power shall be as specified in Section 3.9.A except as specified herein.

1. At least two units 1 and 2 diesel generators and their associated 4-kV shutdown boards shall be OPERABLE.

2. An additional source of power energized and capable of supplying power to the units 1 and 2 shutdown boards consisting of at least one of the following:

   a. One of the offsite power sources specified in 3.9.A.1.c.

   b. A third OPERABLE diesel generator.

3. At least one 480-V shutdown board for each unit must be OPERABLE.

4. One 480-V RMOV board mg set is required for each RMOV board (2D or 2E) required to support operation of the RHR system in accordance with 3.5.B.9.

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

SAFETY EVALUATION BY THE OFFICE OF SPECIAL PROJECTS

SUPPORTING AMENDMENT NO. 158 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 154 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 129 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 May 31, 1988, the Tennessee Valley Authority (TVA or the licensee) requested an amendment to Appendix A of the Browns Ferry Nuclear Plant (BFN), Units 1, 2 and 3 Technical Specifications (TS). The proposed amendment would modify the TS by changing definitions and adding notes to remove ambiguity from the definitions of Mode of Operation, Core Alteration, and Reactor Conditions.

2.0 EVALUATION

2.1 Proposed Changes

The specific proposed changes to the BFN TS are as follows:

1. Definitions 1.0.M, 1.0.M.1, 1.0.M.2, 1.0.M.3 and 1.0.M.4, Mode of Operation, are changed to:

   a. Directly link the mode of operation to the position of the reactor mode switch.

   b. Permit the position of the reactor mode switch to be temporarily changed for performance of a test or other operation while the unit does not change its mode of operation.

   c. Make these definitions applicable only when there is fuel in the reactor vessel. The unit would not be considered to be in any defined mode of operation or operational condition without fuel in the vessel.

   d. Delete extraneous information which describes the selection functions of the reactor mode switch and specify exceptions to the definitions of modes of operation and operational conditions relative to the reactor mode switch position.

2. Definition 1.0.S, Core Alterations, is changed to:

   a. Specify the core components whose addition, removal, relocation, or movement within the reactor vessel constitutes a core alteration (fuel, sources, incore instruments, and reactivity controls.)

   b. Specify that handling of these core components only constitutes core alteration when there is fuel in the vessel and the vessel head is removed.

   c. Permit a core alteration to be completed as necessary to leave the unit in a safe, conservative condition when suspension of the core alteration is required.

3. Definitions 1.0.D, 1.0.H., 1.0.I, 1.0.J, 1.0.K, 1.0.L, and 1.0.X are revised to make these definitions consistent with the changes described above for definitions 1.0.M, 1.0.M.1, 1.0.M.2, 1.0.M.3, and 1.0.M.4. Specifically:

   a. Definition 1.0.H, Reactor Power Operation, is changed to reference the startup/hot standby and run modes instead of the mode switch positions of startup/hot standby and run.

   b. Definition 1.0.I, Hot Standby Condition, is the new definition 1.0.J. It is revised to reference the reactor mode instead of the mode switch position, to delete any reference to reactor pressure and main steam isolation valve position, and to allow reactor coolant temperature to be below 212°F. A note has been added to this definition to help distinguish between hot standby and startup conditions.

   c. The new definition 1.0.I, Startup Condition, will define the term startup condition as the withdrawal of control rods for the purpose of making the reactor critical, with the reactor in the startup/hot standby mode, and reactor power less than or equal to one percent of rated power.

   d. Definition 1.0.J, Cold Condition, is the new definition 1.0.L. It is also revised to explicitly state that it is applicable to any mode of operation.

   e. Current definitions 1.0.K, 1.0.L, and 1.0.X, Hot Shutdown, Cold Shutdown, and Shutdown, respectively, are revised and combined into the new definition 1.0.K for shutdown condition. The new shutdown condition definition includes two subdivisions - hot shutdown condition and cold shutdown condition. The new definition of shutdown conditions includes the refueling mode as well as the shutdown mode. The two subdivisions will reference the shutdown condition instead of the shutdown mode.

f. The new definition 1.0.D, Prior to Startup, will define the phrase "prior to startup" as meaning prior to the withdrawal of control rods for the purpose of making the reactor critical.

4. The remaining changes to the TS involve revisions to achieve consistency with the new TS definitions. This necessitates the revision of a number of limiting conditions for operation and surveillance requirements. In addition, on all pages which are submitted for change, any terms or phrases which are defined in the TS definitions sections (Section 1.0) will be printed entirely in uppercase letters.

2.2 Evaluation of Planned Changes

The proposed changes to definition 1.0.M, Mode of Operation, will directly link the mode of operation to a position of the reactor mode switch. The changes will eliminate extraneous information so that each mode of operation is defined in a straightforward manner. The change is justified in that the intent and requirement of this section are not changed, only clarified. Therefore, the staff finds the changes acceptable.

The proposed change to definition 1.0.M is to allow a unit not to be considered in any mode of operation when there is no fuel in the reactor vessel, thus allowing the reactor mode switch to be in any position or inoperable, without fuel in the vessel. Since the reactor mode switch performs no safety function when the reactor is defueled, there is no reason to restrict it to any position, or require its operability. This change will allow the reactor mode switch position to be changed or disabled as necessary to permit testing and maintenance on a defueled unit without imposing restrictions on that unit. This change is acceptable.

Footnote (1) to Definition 1.0.M will allow the operator to move the mode switch to any position to perform required tests or maintenance without changing the mode of operation, provided that all control rods are verified to remain fully inserted. Second person verification by a licensed operator or other technically qualified member of the plant staff of all-rods-in condition is required. This note is necessary to allow testing or maintenance which may require the mode switch to be in a position other than that for the current mode of operation. This note applies to the shutdown and refuel modes. With the reactor mode switch in the shutdown position, the reactor is designed to be shutdown (i.e., subcritical) with all control rods fully inserted. To enforce this condition, the mode switch in the shutdown position provides a scram signal to the Reactor Protection System (bypassed after two seconds) and a rod withdrawal block signal so that all control rods will remain fully inserted. The administrative requirement for second person verification that all rods remain fully inserted will effectively compensate for the scram and rod block signals which will be bypassed when the mode switch is moved from the shutdown position. Therefore, the use of this note for the shutdown mode will not allow any plant conditions different from those currently allowed by the TS. With the mode switch in the refuel position, interlocks ensure that during fuel movements in or over the core all control rods remain fully inserted and that no more than one control rod can be withdrawn from its fully inserted position. The administrative requirement for second person verification of the all-rods-in condition will clearly meet the intended function of these interlocks when the interlocks are bypassed by moving the mode switch from the refuel position. Therefore, the use of this note for the refuel mode will meet the safety design basis of the refueling interlocks and will not allow any plant conditions different from those currently allowed by the TS. Since the addition of footnote (1) to this definition will not allow any plant conditions to exist which are different from those currently allowed by the technical specifications, this proposed change will not adversely affect safety and is therefore acceptable.

Footnote (2) relating to the mode of operation definition will allow placing the mode switch in the refuel position to perform maintenance on a single control rod drive per specification 3.10.A.5, if the reactor coolant temperature is below 212°F. This note applies to the shutdown mode only, and the reactor would be considered to be in the shutdown mode with the mode switch in the refuel position under the terms of the note. The proposed note requires that all refueling interlocks be operable (per specification 3.10.A.1) so that the one-rod-out interlock of the refuel position will not prevent any further control withdrawal if any single rod is not at its fully inserted position. Since this note will be used to remove control rod drives from the reactor vessel, the control blade associated with that drive will be disabled in the fully withdrawn position and will temporarily be incapable of being inserted. To compensate for this condition, the control rods which face adjacent and diagonally adjacent to the withdrawn rod will be electrically disarmed in the full-in position per specification 3.10.A.5 since these rods would have the highest control rod worth. In this manner, it is ensured that the reactor will remain subcritical since the shutdown margin analysis assumes the single-rod-out condition. Since it is ensured that the reactor will remain subcritical with the required shutdown margin, no assumptions for any accident analysis are changed, and the addition of this footnote will not adversely impact nuclear safety. Therefore the change is acceptable.

Footnote (3) relating to the mode of operation definition will allow placing the mode switch in the refuel position to recouple or withdraw a single control rod provided that the one-rod-out interlock is operable. This note applies to the shutdown mode only, and the reactor will be considered to be in the shutdown mode with the mode switch in the refuel position under the terms of this note. Since the proposed note requires the one-rod-out interlock of the refuel mode switch position to be operable, no more than one control rod will be withdrawn from the full-in position at a time. This interlock will ensure that the reactor will remain subcritical at all times, since the shutdown margin analysis assumes a single rod-out condition. Control rod drives which are moved under the terms of this note will be operable so the adjacent drives need not be disarmed. Since it is assured that the reactor in the shutdown mode will remain subcritical with the required shutdown margin, no assumptions of any accident analysis are changed, and the addition of this footnote will not adversely affect safety and is acceptable.

Footnote (4) relating to the mode of operation definition will allow placing the mode switch in the startup/hot standby position to test the Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS). This note applies to the shutdown mode only, and the reactor would be considered to be in the shutdown mode with the mode switch in the startup/hot standby position. This exception is necessary because certain features of the RWM and RSCS cannot be tested unless the mode switch is placed in the startup/hot standby position. The testing required involves selection and withdrawal of control rods to verify that the RWM and RSCS are enforcing rod patterns correctly. Since this test will only have one rod withdrawn from its fully inserted position at a time, the reactor cannot achieve criticality and the intent of the shutdown mode will be maintained. The addition of this note will only clarify existing requirements for testing of the RWM and RSCS and will not change any procedures for operation or testing of these systems. Since this note only clarifies existing surveillance requirements, but does not change their intent or application, it will not adversely impact safety and is acceptable.

The changes to definition 1.0.S, Core Alterations, are proposed to clarify the components to which this definition will apply. Core alterations will be limited to fuel, sources, incore instruments, and reactivity controls, which are the components which can contribute to an accident during core alterations. This definition applies when the vessel head is removed and when fuel is in the vessel. The proposed changes to definition 1.0.S allow completion of a movement of a component to a safe conservative position when core alterations are suspended. This change will prevent leaving a component in an intermediate position (such as a fuel bundle suspended from the refueling bridge) for an extended period of time. This addition will not allow any new moves to be initiated when core alterations are suspended. Since the proposed changes to specification 1.0.S are clarification only, the intent of this definition is unchanged and there is no impact on safety. The new definition for core alterations is acceptable.

The proposed definition 1.0.H, Reactor Power Operation, is only changed to reference the startup/hot standby and run modes instead of mode switch positions of startup and run. This is a necessary administrative change to provide consistency between the mode and condition definitions and is therefore acceptable.

The proposed definition 1.0.J, Hot Standby Condition, will alter the current definition of hot standby to remove the reference to reactor vessel pressure being limited to 1055 psig and remove the reference to the main steam isolation valve (MSIV) position. This definition is necessary to describe the condition of the reactor at low power levels such as prior to entering the startup condition, and during a controlled shutdown after reactor power drops below one percent of rated thermal power in the startup condition, but the distinction between the two is found in the intended condition of operation toward which the reactor is proceeding. A note has been added to the hot standby definition to make this distinction clear in the TS.

The deletion of the reference to reactor vessel pressure found in the current definition of the hot standby condition will not change plant operation or requirements in any way. Vessel pressure limits are well documented in other sections of TS such as the limited safety system settings for relief valve setpoints and nuclear system high pressure scram setpoint, and Table 3.1.A for RPS scram instrumentation requirements. This change is administrative in nature and is therefore acceptable.

The deletion of the requirement to have the main steam isolation valves (MSIVs) closed in the hot standby condition will allow the reactor to use the main condenser as a heat sink in this condition. The existing requirement to have the MSIVs closed has made the hot standby condition an undesirable condition because it isolates the reactor from the main condenser and forces the use of relief valves to control reactor pressure. Since no accident or transient analysis involving MSIV closure assumes that the MSIVs are closed initially, and since this change will not affect any MSIV isolation function, no assumptions used in any accident or transient analysis are invalidated by this proposed change. Therefore, there is no effect on nuclear safety resulting from allowing the MSIVs to be open in the hot standby condition. This change is acceptable.

The change to allow reactor coolant temperature to drop below 212°F will allow the reactor to be in a cold, more conservative state while in the hot standby condition. Hence, this change does not affect nuclear safety and is acceptable.

The proposed definition 1.0.I, Startup Condition, defines the condition of starting up, as opposed to the startup mode. This condition will be in existence when the reactor is in the startup/hot standby mode, reactor power is less than one percent of rated thermal power, and the withdrawal of control rods for the purpose of making the reactor critical has begun. This definition is necessary to describe the condition of the reactor between the time of initially withdrawing control rods and the time of reaching reactor power operation. This proposed definition is justifiable because it is consistent with safe operation for all of its applications within the TS.

The proposed definition 1.0.L, Cold Condition, is revised to explicitly state its applicability in any mode of operation. This is an administrative change that does not alter any current TS requirements or allow any new operational conditions. This change will not have any impact on nuclear safety, and is acceptable.

The proposed definition 1.0.K, Shutdown Condition, will provide an explicit definition for the shutdown condition and will consolidate the hot shutdown and cold shutdown definitions into one section. This definition will allow the reactor to be in the shutdown mode or in the refuel mode and be considered in the shutdown condition. In this condition, the reactor mode switch will be in either the shutdown or refuel positions. These mode switch positions, will allow at most, only one control rod at a time to be withdrawn from the fully inserted position. Since the reactor is analyzed for adequate shutdown margin with the analytically determined highest worth rod fully withdrawn, it is ensured that the reactor will always be subcritical while in the shutdown condition. Thus, the change is acceptable.

The hot shutdown condition will simply be defined as the condition which exists when the reactor is in the shutdown condition with average reactor coolant temperature greater than 212°F. This is similar to the current definition of hot shutdown, with the exception that the new definition of shutdown condition will be referenced instead of the shutdown mode. The intention of this definition is not changed by this amendment request since, as before, the reactor will always be subcritical with average coolant temperature equal to or less than 212°F when in the cold shutdown condition. This change is acceptable.

The cold shutdown condition will simply be defined as the condition which exists when the reactor is in the shutdown condition with average reactor coolant temperature equal to or less than 212°F. This is similar to the current definition of cold shutdown, with the exception that the new definition of shutdown condition will be referenced instead of the shutdown mode. The intention of this definition is not changed by this amendment request since, as before, the reactor will always be subcritical with average coolant temperature equal to or less than 212°F when in the cold shutdown condition. This change is acceptable.

The proposed definition 1.0.D, Prior to Startup, will provide a definition of the phrase "prior to startup" to explicitly state the intention of this phrase. Prior to startup will be defined as prior to the withdrawal of control rods for the purpose of going critical. "Startup" used in this phrase will be referring to the startup condition, and not to the startup mode. This definition is necessary to make this distinction between modes and conditions. This change is justified because it is consistent with safe operation for all of its applications within the TS, as is noted in the following paragraphs.

The phrase "prior to startup" appears in the LCO's for the Core Spray System (CSS), the Residual Heat Removal (RHR) System, the High Pressure Coolant Injection (HPCI) System, the Reactor Core Isolation Cooling System (RCIC) System, and the Automatic Depressurization System (ADS). Each of these core and containment cooling systems is required by the TS to be operable prior to startup from a cold condition. By the proposed definition, these systems would be required to be operable before withdrawing control rods with the intention of going critical. With the reactor in a cold condition and with all control rods inserted, the reactor is physically in the same condition as if it were in cold shutdown. In the cold shutdown condition, these core and containment cooling systems are not required to be operable. Therefore, the proposed changes will not allow any physical plant configuration different from those currently allowed by the TS and are acceptable.

The remaining TS changes are administrative in nature and involve making the TS consistent with respect to the revised definitions. The new definitions necessitate the revision of a number of limiting conditions for operation and surveillance requirements. In addition, the terms or phrases which are defined in the TS definitions section have been printed entirely in upper case letters. These changes are both administrative and clarifying in nat re and are therefore acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

The amendments involve a change to a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and/or changes to the surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (53 FR 30144) on August 10, 1988 and consulted with the State of Alabama. No public comments were received and the State of Alabama did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: J. Kelly and G. Gears

Dated: November 18, 1988