UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 222
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 30, 1994, and supplemented on November 21, 1994 and March 9, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 222, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

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

Attachment: Changes to the Technical Specifications

Date of Issuance: July 17, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 222

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

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 |
|---|---|
| vii | vii |
| viii | viii* |
| 1.1/2.1-5 | 1.1/2.1-5 |
| --- | 1.1/2.1-5a |
| 1.1/2.1-10 | 1.1/2.1-10 |
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| 3.2/4.2-7 | 3.2/4.2-7 |
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| 3.7/4.7-9 | 3.7/4.7-9* |
| 3.7/4.7-10 | 3.7/4.7-10 |
| --- | 3.7/4.7-24a |
| --- | 3.7/4.7-24b |
| 3.7/4.7-33 | 3.7/4.7-33* |
| 3.7/4.7-34 | 3.7/4.7-34 |

## LIST OF TABLES (Cont'd)

## LIST OF ILLUSTRATIONS

FEB 24 1995

| SAFETY LIMIT | LIMITING SAFETY SYSTEM SETTING |
|---|---|

**1.1.B. Power Transient**

To ensure that the SAFETY LIMITS established in Specification 1.1.A are not exceeded, each required scram shall be initiated by its expected scram signal. The SAFETY LIMIT shall be assumed to be exceeded when scram is accomplished by means other than the expected scram signal.

**2.1.B. Power Transient Trip Settings**

| | | |
|---|---|---|
| 1. | Scram and isolation (PCIS groups 2,3,6) reactor low water level | ≥ 538 in. above vessel zero |
| 2. | Scram--turbine stop valve closure | ≤ 10 percent valve closure |
| 3. | Scram--turbine control valve fast closure or turbine trip | ≥ 550 psig |
| 4. | (Deleted) | |
| 5. | Scram--main steam line isolation | ≤ 10 percent valve closure |
| 6. | Main steam isolation valve closure --nuclear system low pressure | ≥ 825 psig |

**C. Reactor Vessel Water Level**

Whenever there is irradiated fuel in the reactor vessel, the water level shall be greater than or equal to 372.5 inches above vessel zero.

**C. Water Level Trip Settings**

| | | |
|---|---|---|
| 1. | Core spray and LPCI actuation--reactor low water level | ≥ 398 in. above vessel zero |
| 2. | HPCI and RCIC actuation--reactor low water level | ≥ 470 in. above vessel zero |
| 3. | Main steam isolation valve closure--reactor low water level | ≥ 398 in. above vessel zero |

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1.1 BASES (Cont'd)

The safety limit has been established at 372.5 inches above vessel zero to provide a point which can be monitored and also provide adequate margin to assure sufficient cooling.

REFERENCE


1. General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO 10958 and NEDE 10938.

2. General Electric Document No. EAS-65-0687, Setpoint Determination for Browns Ferry Nuclear Plant, Revision 2.

2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Browns Ferry Nuclear Plant have been analyzed in support of planned operating conditions up to the maximum thermal power of 3293 MWt. The analyses were based upon plant operation in accordance with Reference 1.

The transient analyses performed for each reload are described in Reference 2. Models and model conservatisms are also described in this reference.

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

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

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

| Minimum No. Instrument Channels Operable Per Trip Sys(1)(11) | Function | Trip Level Setting | Action (1) | Remarks |
|---|---|---|---|---|
| 2 | Instrument Channel - Low Pressure Main Steam Line | $\geq$ 825 psig (4) | B | 1. Below trip setting initiates Main Steam Line Isolation |
| 2(3) | Instrument Channel - High Flow Main Steam Line | $\leq$ 140% of rated steam flow | B | 1. Above trip setting initiates Main Steam Line Isolation |
| 2(12) | Instrument Channel - Main Steam Line Tunnel High Temperature | $\leq$ 200°F | B | 1. Above trip setting initiates Main Steam Line Isolation. |
| 2(14) | Instrument Channel - Reactor Water Cleanup System Floor Drain High Temperature | 160 - 180°F | C | 1. Above trip setting initiates Isolation of Reactor Water Cleanup Line from Reactor and Reactor Water Return Line. |
| 2 | Instrument Channel - Reactor Water Cleanup System Space High Temperature | 160 - 180°F | C | 1. Same as above |
| 1(15) | Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone | $\leq$ 100 mr/hr or downscale | G | 1. 1 upscale channel or 2 downscale channels will<br>a. Initiate SGTS<br>b. Isolate reactor zone and refueling floor.<br>c. Close atmosphere control system. |

TABLE 3.2.B
INSTRUMENTATION THAT INITATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

| Minimum No. Operable Per Trip Sys(1) | Function | Trip Level Setting | Action | Remarks |
|---|---|---|---|---|
| 2 | Instrument Channel - Reactor Low Water Level | ≥ 470" above vessel zero | A | 1. Below trip setting initiates HPCI. |
| 2 | Instrument Channel - Reactor Low Water Level | ≥ 470" above vessel zero. | A | 1. Multiplier relays initiate RCIC. |
| 2 | Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #1) | ≥ 398" above vessel zero. | A | 1. Below trip setting initiates CSS.<br>Multiplier relays initiate LPCI.<br>2. Multiplier relay from CSS initiates accident signal (15). |
| 2(16) | Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #2) | ≥ 398" above vessel zero. | A | 1. Below trip settings, in conjunction with drywell high pressure, low water level permissive, ADS timer timed out and CSS or RHR pump running, initiates ADS.<br>2. Below trip settings, in conjunction with low reactor water level permissive, ADS timer timed out, ADS high drywell pressure bypass timer timed out, CSS or RHR pump running, initiates ADS. |
| 1(16) | Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184 & 185, SW #1) | ≥ 544" above vessel zero. | A | 1. Below trip setting permissive for initiating signals on ADS. |
| 1 | Instrument Channel - Reactor Low Water Level (LITS-3-52 and 62, SW #1) | ≥ 312 5/16" above vessel zero. (2/3 core height) | A | 1. Below trip setting prevents inadvertent operation of containment spray during accident condition. |

TABLE 3.2.B (Continued)

| Minimum No. Operable Per Trip Sys(1) | Function | Trip Level Setting | Action | Remarks |
|---|---|---|---|---|
| 2(18) | Instrument Channel - Drywell High Pressure (PS-64-58 E-H) | $1 \le p \le 2.5$ psig | A | 1. Below trip setting prevents inadvertent operation of containment spray during accident conditions. |
| 2(18) | Instrument Channel - Drywell High Pressure (PS-64-58 A-D, SW #2) | $\le$ 2.5 psig | A | 1. Above trip setting in conjunction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI.<br>2. Multiplier relay from CSS initiates accident signal. (15) |
| 2(18) | Instrument Channel - Drywell High Pressure (PS-64-58A-D, SW #1) | $\le$ 2.5 psig | A | 1. Above trip setting in conjunction with low reactor pressure initiates LPCI. |
| 2(16)(18) | Instrument Channel - Drywell High Pressure (PS-64-57A-D) | $\le$ 2.5 psig | A | 1. Above trip setting, in conjunction with low reactor water level, low reactor water level permissive, ADS timer timed out, and CSS or RHR pump running, initiates ADS. |

NOTES FOR TABLE 3.2.B

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

   Action:

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

   B. Declare the system or component inoperable.

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

   D. No action required; indicators are considered redundant.

   E. Within 24 hours restore the inoperable channel(s) to OPERABLE status or place the inoperable channel(s) in the tripped condition.

2. In only one trip system.

3. Not considered in a trip system.

4. Deleted

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

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

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

8. Note 1 does not apply to this item.

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

NOTES FOR TABLE 3.2.B (Cont'd)

10. Only one trip system for each cooler fan.

11. In only two of the four 4160-V shutdown boards. See note 13.

12. In only one of the four 4160-V shutdown boards. See note 13.

13. An emergency 4160-V shutdown board is considered a trip system.

14. RHRSW pump would be inoperable. Refer to Section 4.5.C for the requirements of a RHRSW pump being inoperable.

15. The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level ($\geq$ 398" above vessel zero) originating in the core spray system trip system.

16. The ADS circuitry is capable of accomplishing its protective action with one OPERABLE trip system. Therefore, one trip system may be taken out of service for functional testing and calibration for a period not to exceed eight hours.

17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 30 percent within four hours.

18. Not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever PRIMARY CONTAINMENT INTEGRITY is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 538 inches above vessel zero closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Groups 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 470 inches above vessel zero (Table 3.2.B) trips the recirculation pumps and initiates the RCIC and HPCI systems. The RCIC and HPCI system initiation opens the turbine steam supply valve which in turn initiates closure of the respective drain valves (Group 7).

The low water level instrumentation set to trip at $\geq$ 398 inches above vessel zero (Table 3.2.B) closes the Main Steam Isolation Valves, the Main Steam Line Drain Valves, and the Reactor Water Sample Valves (Group 1). These trip settings are adequate to prevent core uncovery in the case of a break in the largest line assuming the maximum closing time.

The low reactor water level instrumentation that is set to trip when reactor water level is $\geq$ 398 inches above vessel zero (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation, and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and, in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus, the results given above are applicable here also.

ADS provides for automatic nuclear steam system depressurization, if needed, for small breaks in the nuclear system so that the LPCI and the CSS can operate to protect the fuel from overheating. ADS uses six of the 13 MSRVs to relieve the high pressure steam to the suppression pool. ADS initiates when the following conditions exist: low reactor water level permissive (level 3), low reactor water level (level 1), high drywell pressure or the ADS high drywell pressure bypass timer timed out, and the ADS timer timed out. In addition, at least one RHR pump or two core spray pumps must be running.

The ADS high drywell pressure bypass timer is added to meet the requirements of NUREG 0737, Item II.K.3.18. This timer will bypass the high drywell pressure permissive after a sustained low water level. The worst case condition is a main steam line break outside primary containment with HPCI inoperable. With the ADS high drywell pressure bypass timer analytical limit of 360 seconds, a Peak Cladding Temperature (PCT) of 1500°F will not be exceeded for the worst case event. This temperature is well below the limiting PCT of 2200°F.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the high steam flow instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

j. Continuous Leak Rate Monitor

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

k. Drywell and Torus Surfaces

The interior surfaces of the drywell and torus above the level one foot below the normal water line and outside surfaces of the torus below the water line shall be visually inspected each operating cycle for deterioration and any signs of structural damage with particular attention to piping connections and supports and for signs of distress or displacement.

LIMITING CONDITIONS FOR OPERATION

3.7.A PRIMARY CONTAINMENT

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.

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

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

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

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

SURVEILLANCE REQUIREMENTS

4.7.A PRIMARY CONTAINMENT

3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.

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

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.

b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

TABLE 3.7.A

INSTRUMENTATION FOR CONTAINMENT SYSTEMS

| Minimum No. Operable Per Trip System | Function | Trip Level Setting | Action | Remarks |
|---|---|---|---|---|
| 2 | Instrument Channel - Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21) | 0.5 psid | (1) | Actuates the pressure suppression chamber-reactor building vacuum breakers. |

Footnote:

(1) - Repair in 24 hours. If the function is not OPERABLE in 24 hours, declare the system or component inoperable.

TABLE 4.7.A

CONTAINMENT SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| Function | Functional Test | Calibration | Instrument Check |
|---|---|---|---|
| Instrument Channel- Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21) | Once/month[1] | Once/18 months[2] | None |

Footnotes:

(1) - Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify OPERABILITY of the trip and alarm functions.

(2) - Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the level setting.

containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after two years of operation in the rugged shipboard environment on the US Savannah (ORNL 3726). Pressure drop across the combined HEPA filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow ASTM D3803-89. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

All elements of the heater should be demonstrated to be functional and OPERABLE during the test of heater capacity. Operation of each filter train for a minimum of 10 hours each month will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repaired and test repeated.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability and OPERABILITY of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other systems must be tested daily. This substantiates the availability of the OPERABLE systems and thus reactor operation and refueling operation can continue for a limited period of time.

3.7.D/4.7.D Primary Containment Isolation Valves

The Browns Ferry Containment Leak Rate Program and Procedures contains the list of all the Primary Containment Isolation Valves for which the Technical Specification requirements apply. The procedures are subject to the change control provisions for plant procedures in the administrative controls section of the Technical Specifications. The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a LOCA.

Group 1 - Process lines are isolated by reactor vessel low water level (≥398") in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in Group 1, except the reactor water sample line valves, are also closed when process instrumentation detects excessive main steam line flow, low pressure, or main steam space high temperature. The reactor water sample line valves isolate only on reactor low water level at ≥398".

Group 2 - Isolation valves are closed by reactor vessel low water level (538") or high drywell pressure. The Group 2 isolation signal also "isolates" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the Group 2 isolation signal by a transient or spurious signal.

Group 3 - Process lines are normally in use, and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from nonsafety related causes. To protect the reactor from a possible pipe break

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 237
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 March 30, 1994, and supplemented on November 21, 1994 and March 9, 1995, 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;

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   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

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3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

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Project Directorate II-4
Division of Reactor Projects - I/II
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| vii | vii |
| viii | viii* |
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| 3.7/4.7-10 | 3.7/4.7-10 |
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## LIST OF TABLES (Cont'd)

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## LIST OF ILLUSTRATIONS

SEP 2 2 1993

NOTES FOR TABLE 3.2.K

*(Deleted)
**(Deleted)
***During main condenser offgas treatment system operation

ACTION A

(Deleted)

ACTION B

(Deleted)

ACTION C

(Deleted)

ACTION D

(Deleted)

ACTION E

With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of main condenser offgas treatment system may continue provided that a temporary monitor is installed or grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION F

(Deleted)

Table 3.2.L
Anticipated Transient Without Scram (ATWS) -
Recirculation Pump Test (RPT) Surveillance Instrumentation

| Minimum No. Channels Operable per Trip Sys (1) | Function | Trip Setting | Allowable Value | Action | Remarks |
|---|---|---|---|---|---|
| 2 | ATWS/RPT Logic Reactor Dome Pressure High (PIS-3-204A-D) | 1118 psig | ≤ 1146.5 psig | (2) | Two out of two of the high reactor dome pressure channels or the low reactor vessel level channels in either trip system trips both reactor recirculation pumps. |
| 2 | Reactor Vessel Level Low (LS-3-58 A1-D1) | 483" above vessel zero | ≥ 471.52" above vessel zero | | |

(1) One channel in only one trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided the other channels in that trip system are OPERABLE.

(2) Two trip systems exist, either of which will trip both recirculation pumps. Perform Surveillance/maintenance/calibration on one channel in only one trip system at a time. If a channel is found to be inoperable or if the surveillance/maintenance/calibration period for one channel exceeds 6 consecutive hours, the trip system will be declared inoperable or the channel will be placed in a tripped condition. If in RUN mode and one trip system is inoperable for 72 hours or both trip systems are inoperable, the reactor shall be in at least the HOT STANDBY CONDITION within 6 hours.

TABLE 4.2.B

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

| Function | Functional Test | Calibration | Instrument Check |
|---|---|---|---|
| Instrument Channel Reactor Low Water Level (LIS-3-58A-D, LS-3-58A-D) | (1) (27) | Once/18 Months (28) | Once/day |
| Instrument Channel Reactor Low Water Level (LIS-3-184 & 185) | (1) (27) | Once/18 Months (28) | Once/day |
| Instrument Channel Reactor Low Water Level (LIS-3-52 & 62A) | (1) (27) | Once/18 Months (28) | Once/day |
| Instrument Channel Drywell High Pressure (PIS-64-58E-H) | (1) (27) | Once/18 Months (28) | none |
| Instrument Channel Drywell High Pressure (PIS-64-58A-D) | (1) (27) | Once/18 Months (28) | none |
| Instrument Channel Drywell High Pressure (PIS-64-57A-D) | (1) (27) | Once/18 Months (28) | none |
| Instrument Channel Reactor Low Pressure (PIS-3-74A&B, PS-3-74A&B) (PIS-68-95, PS-68-95) (PIS-68-96, PS-68-96) | (1) (27) | Once/6 Months (28) | none |

TABLE 4.2.B (Continued)

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

| Function | Functional Test | Calibration | Instrument Check |
|---|---|---|---|
| Core Spray Auto Sequencing Timers (Normal Power) | (4) | Once/operating cycle | none |
| Core Spray Auto Sequencing Timers (Diesel Power) | (4) | Once/operating cycle | none |
| LPCI Auto Sequencing Timers (Normal Power) | (4) | Once/operating cycle | none |
| LPCI Auto Sequencing Timers (Diesel Power) | (4) | Once/operating cycle | none |
| RHRSW A1, B3, C1, D3 Timers (Normal Power) | (4) | Once/operating cycle | none |
| RHRSW A1, B3, C1, D3 Timers (Diesel Power) | (4) | Once/operating cycle | none |
| ADS Timer | (4) | Once/operating cycle | none |
| ADS High Drywell Pressure Bypass Timer | (4) | Once/operating cycle | none |
| RCIC Steam Line Space Torus Area High Temperature | (1) | Once/3 months | none |
| RCIC Steam Line Space RCIC Pump Room Area High Temperature | (1) | Once/3 months | none |

TABLE 4.2.B (Continued)

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

| Function | Functional Test | Calibration | Instrument Check |
|---|---|---|---|
| Instrument Channel - RHR Pump Discharge Pressure | (1) | Once/3 months | none |
| Instrument Channel - Core Spray Pump Discharge Pressure | (1) | Once/3 months | none |
| Core Spray Sparger to RPV d/p | (1) | Once/3 months | Once/day |
| Trip System Bus Power Monitor | Once/operating Cycle | N/A | none |
| Instrument Channel - Condensate Header Low Level (LS-73-56A, B) | (1) | Once/3 months | none |
| Instrument Channel - Suppression Chamber High Level | (1) | Once/3 months | none |
| Instrument Channel - Reactor High Water Level (LIS-3-208A-D) | (1)(27) | Once/18 months (28) | Once/day |
| Instrument Channel - RCIC Turbine Steam Line High Flow | (1)(27) | Once/18 months (28) | none |
| Instrument Channel - RCIC Steam Supply Low Pressure | Once/31 days | Once/18 months | none |
| Instrument Channel - RCIC Turbine Exhaust Diaphragm High Pressure | Once/31 days | Once/18 months | none |
| HPCI Steam Line Space Torus Area High Temperature | (1) | Once/3 months | none |
| HPCI Steam Line Space HPCI Pump Room Area High Temperature | (1) | Once/3 months | none |

TABLE 4.2.B (Continued)

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

| Function | Functional Test | Calibration | Instrument Check |
|---|---|---|---|
| Instrument Channel - HPCI Turbine Steam Line High Flow | (1)(27) | Once/18 months (28) | none |
| Instrument Channel - HPCI Steam Supply Low Pressure | Once/31 days | Once/18 months | none |
| Instrument Channel - HPCI Turbine Exhaust Diaphragm High Pressure | Once/31 days | Once/18 months | none |
| Core Spray System Logic | Once/18 months | (6) | N/A |
| RCIC System (Initiating) Logic | Once/18 months | N/A | N/A |
| RCIC System (Isolation) Logic | Once/18 months | (6) | N/A |
| HPCI System (Initiating) Logic | Once/18 months | (6) | N/A |
| HPCI System (Isolation) Logic | Once/18 months | (6) | N/A |
| ADS Logic | Once/18 months | (6) | N/A |
| LPCI (Initiating) Logic | Once/18 months | (6) | N/A |
| LPCI (Containment Spray) Logic | Once/18 months | (6) | N/A |
| Core Spray System Auto Initiation Inhibit (Core Spray Auto Initiation) | Once/18 months (7) | N/A | N/A |
| LPCI Auto Initiation Inhibit (LPCI Auto Initiation) | Once/18 months (7) | N/A | N/A |

TABLE 4.2.F

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

| Instrument Channel | Calibration Frequency | Instrument Check |
|---|---|---|
| 1) Reactor Water Level (LI-3-58A&B) | Once/18 months | Each Shift |
| 2) Reactor Pressure (PI-3-74A&B) | Once/6 months | Each Shift |
| 3) Drywell Pressure (PI-64-67B) and XR-64-50 | Once/6 months | Each Shift |
| 4) Drywell Temperature (TI-64-52AB) and XR-64-50 | Once/6 months | Each Shift |
| 5) Suppression Chamber Air Temperature (XR-64-52) | Once/6 months | Each Shift |
| 8) Control Rod Position | N/A | Each Shift |
| 9) Neutron Monitoring | (2) | Each Shift |
| 10) Drywell Pressure (PS-64-67B) | Once/6 months | N/A |
| 11) Drywell Pressure (PIS-64-58A) | Once/18 months | N/A |
| 12) Drywell Temperature (TS-64-52A) | Once/6 months | N/A |
| 13) Timer (IS-64-67A) | Once/6 months | N/A |
| 14) CAD Tank Level | Once/6 months | Once/day |
| 15) Containment Atmosphere Monitors | Once/6 months | Once/day |

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

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

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

j. Continuous Leak Rate Monitor

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

k. Drywell and Torus Surfaces

The interior surfaces of the drywell and torus above the level one foot below the normal water line and outside surfaces of the torus below the water line shall be visually inspected each operating cycle for deterioration and any signs of structural damage with particular attention to piping connections and supports and for signs of distress or displacement.

LIMITING CONDITIONS FOR OPERATION

3.7.A Primary Containment

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.

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

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

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

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

SURVEILLANCE REQUIREMENTS

4.7.A Primary Containment

3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.

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

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.

b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

TABLE 3.7.A

INSTRUMENTATION FOR CONTAINMENT SYSTEMS

| Minimum No. Operable Per Trip System | Function | Trip Level Setting | Action | Remarks |
|---|---|---|---|---|
| 2 | Instrument Channel - Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21) | 0.5 psid | (1) | Actuates the pressure suppression chamber-reactor building vacuum breakers. |

Footnote:

(1) - Repair in 24 hours. If the function is not OPERABLE in 24 hours, declare the system or component inoperable.

TABLE 4.7.A

CONTAINMENT SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| Function | Functional Test | Calibration | Instrument Check |
|---|---|---|---|
| Instrument Channel-Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21) | Once/month(1) | Once/18 months(2) | None |

Footnotes:

(1) - Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify OPERABILITY of the trip and alarm functions.

(2) - Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the level setting.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 196
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 30, 1994, and supplemented on November 21, 1994 and March 9, 1995, 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-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.196 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment: Changes to the Technical Specifications

Date of Issuance: July 17, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 196

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

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 |
|---|---|
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| viii | viii* |
| 1.1/2.1-5 | 1.1/2.1-5 |
| --- | 1.1/2.1-5a |
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| 3.2/4.2-44 | 3.2/4.2-44* |
| 3.2/4.2-45 | 3.2/4.2-45 |
| 3.2/4.2-46 | 3.2/4.2-46 |
| 3.2/4.2-53 | 3.2/4.2-53 |
| 3.2/4.2-54 | 3.2/4.2-54* |
| 3.2/4.2-64 | 3.2/4.2-64 |
| 3.2/4.2-65 | 3.2/4.2-65 |
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| --- | 3.7/4.2-23b |
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| 3.7/4.7-32 | 3.7/4.7-32* |
| 3.7/4.7-33 | 3.7/4.7-33 |

<u>LIST OF TABLES</u> (Cont'd)

# LIST OF ILLUSTRATIONS

| SAFETY LIMIT | LIMITING SAFETY SYSTEM SETTING |
|---|---|
| 1.1.B. Power Transient | 2.1.B. Power Transient Trip Settings |

**1.1.B. Power Transient**

To ensure that the SAFETY LIMITS established in Specification 1.1.A are not exceeded, each required scram shall be initiated by its expected scram signal. The SAFETY LIMIT shall be assumed to be exceeded when scram is accomplished by means other than the expected scram signal.

**C. Reactor Vessel Water Level**

Whenever there is irradiated fuel in the reactor vessel, the water level shall be greater than or equal to 372.5 inches above vessel zero.

**2.1.B. Power Transient Trip Settings**

| | | |
|---|---|---|
| 1. | Scram and isolation (PCIS groups 2,3,6) reactor low water level | ≥ 538 in. above vessel zero |
| 2. | Scram—turbine stop valve closure | ≤ 10 percent valve closure |
| 3. | Scram—turbine control valve fast closure or turbine trip | ≥ 550 psig |
| 4. | (Deleted) | |
| 5. | Scram—main steam line isolation | ≤ 10 percent valve closure |
| 6. | Main steam isolation valve closure —nuclear system low pressure | ≥ 825 psig |

**C. Water Level Trip Settings**

| | | |
|---|---|---|
| 1. | Core spray and LPCI actuation—reactor low water level | ≥ 398 in. above vessel zero |
| 2. | HPCI and RCIC actuation—reactor low water level | ≥ 470 in. above vessel zero |
| 3. | Main steam isolation valve closure—reactor low water level | ≥ 398 in. above vessel zero |

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1.1 BASES (Cont'd)

The safety limit has been established at 372.5 inches above vessel zero to provide a point which can be monitored and also provide adequate margin to assure sufficient cooling.

REFERENCE


1. General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO 10958 and NEDE 10938.

2. General Electric Document No. EAS-65-0687, Setpoint Determination for Browns Ferry Nuclear Plant, Revision 2.

## 2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Browns Ferry Nuclear Plant have been analyzed in support of planned operating conditions up to the maximum thermal power of 3293 MWt. The analyses were based upon plant operation in accordance with Reference 1.

The transient analyses performed for each reload are described in Reference 2. Models and model conservatisms are also described in this reference.

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

| Min. No. of Operable Instr. Channels Per Trip System (1)(23) | Trip Function | Trip Level Setting | Modes in Which Function Must Be Operable | | | | Action (1) |
|---|---|---|---|---|---|---|---|
| | | | Shut-down | Refuel (7) | Startup/ Hot Standby | Run | |
| 1 | Mode Switch in Shutdown | | X | X | X | X | 1.A |
| 1 | Manual Scram | | X | X | X | X | 1.A |
| | IRM (16) | | | | | | |
| 3 | High Flux | ≤120/125 Indicated on scale | X(22) | X(22) | X | (5) | 1.A |
| 3 | Inoperative | | | X | X | (5) | 1.A |
| | APRM (16)(24)(25) | | | | | | |
| 2 | High Flux (Fixed Trip) | ≤ 120% | | | | X | 1.A or 1.B |
| 2 | High Flux (Flow Biased) | See Spec. 2.1.A.1 | | | | X | 1.A or 1.B |
| 2 | High Flux | ≤ 15% rated power | | X(21) | X(17) | (15) | 1.A |
| 2 | Inoperative | (13) | | X(21) | X(17) | X | 1.A |
| 2 | Downscale | ≥ 3 Indicated on Scale | | (11) | (11) | X(12) | 1.A or 1.B |
| 2 | High Reactor Pressure (PIS-3-22AA,BB,C,D) | ≤ 1055 psig | | X(10) | X | X | 1.A |
| 2 | High Drywell Pressure (14) (PIS-64-56 A-D) | ≤ 2.5 psig | | X(8) | X(8) | X | 1.A |
| 2 | Reactor Low Water Level (14) (LIS-3-203 A-D) | ≥ 538" above vessel zero | | X | X | X | 1.A |

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

| Min. No. of Operable Instr. Channels Per Trip System (1)(23) | Trip Function | Trip Level Setting | Modes in Which Function Must Be Operable: Shut-down | Refuel (7) | Startup/ Hot Standby | Run | Action (1) |
|---|---|---|---|---|---|---|---|
| 2 | High Water Level in West Scram Discharge Tank (LS-85-45A-D) | ≤ 50 Gallons | X(2) | X(2) | X | X | 1.A |
| 2 | High Water Level in East Scram Discharge Tank (LS-85-45E-H) | ≤ 50 Gallons | X(2) | X(2) | X | X | 1.A |
| 4 | Main Steam Line Isolation Valve Closure | ≤10% Valve Closure | | | | X(6) | 1.A or 1.C |
| 2 | Turbine Control Valve Fast Closure or Turbine Trip | ≥550 psig | | | | X(4) | 1.A or 1.D |
| 4 | Turbine Stop Valve Closure | ≤10% Valve Closure | | | | X(4) | 1.A or 1.D |
| 2 | Turbine First Stage Pressure Permissive (PIS-1-81A&B) (PIS-1-91A&B) | not ≥154 psig | | X(18) | X(18) | X(18) | 1.A or 1.D (19) |

TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

| | Group (2) | Functional Test | Minimum Frequency(3) |
|---|---|---|---|
| Mode Switch in Shutdown | A | Place Mode Switch in Shutdown | Each Refueling Outage |
| Manual Scram | A | Trip Channel and Alarm | Every 3 Months |
| IRM | | | |
| High Flux | C | Trip Channel and Alarm (4) | Once Per Week During Refueling and Before Each Startup |
| Inoperative | C | Trip Channel and Alarm (4) | Once Per Week During Refueling and Before Each Startup |
| APRM | | | |
| High Flux (15% Scram) | C | Trip Output Relays (4) | Before Each Startup and Weekly When Required to be Operable |
| High Flux (Flow Biased) | B | Trip Output Relays (4) | Once/Week |
| High Flux (Fixed Trip) | B | Trip Output Relays (4) | Once/Week |
| Inoperative | B | Trip Output Relays (4) | Once/Week |
| Downscale | B | Trip Output Relays (4) | Once/Week |
| Flow Bias | B | (6) | (6) |
| High Reactor Pressure (PIS-3-22AA,BB,C,D) | B | Trip Channel and Alarm (7) | Once/Month |
| High Drywell Pressure (PIS-64-56 A-D) | B | Trip Channel and Alarm (7) | Once/Month |
| Reactor Low Water Level (LIS-3-203 A-D) | B | Trip Channel and Alarm (7) | Once/Month |

TABLE 4.1.A (Continued)

| | Group (2) | Functional Test | Minimum Frequency(3) |
|---|---|---|---|
| High Water Level in Scram Discharge Tank Float Switches (LS-85-45C-F) | A | Trip Channel and Alarm | Once/Month |
| Electronic Level Switches (LS-85-45A, B, G, H) | B | Trip Channel and Alarm (7) | Once/Month |
| Main Steam Line Isolation Valve Closure | A | Trip Channel and Alarm | Once/3 Months (8) |
| Turbine Control Valve Fast Closure or turbine trip | A | Trip Channel and Alarm | Once/Month (1) |
| Turbine First Stage Pressure Permissive (PIS-1-81A and B, PIS-1-91A and B) | B | Trip Channel and Alarm (7) | Every three months |
| Turbine Stop Valve Closure | A | Trip Channel and Alarm | Once/Month (1) |

TABLE 4.1.B
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

| Instrument Channel | Group (1) | Calibration | Minimum Frequency(2) |
|---|---|---|---|
| IRM High Flux | C | Comparison to APRM on Controlled Startups (6) | Note (4) |
| APRM High Flux | | | |
| Output Signal | B | Heat Balance | Once Every 7 Days |
| Flow Bias Signal | B | Calibrate Flow Bias Signal (7) | Once/Operating Cycle |
| LPRM Signal | B | TIP System Traverse (8) | Every 1000 Effective Full Power Hours |
| High Reactor Pressure (PIS-3-22AA,BB,C,D) | B | Standard Pressure Source | Once/6 Months(9) |
| High Drywell Pressure (PIS-64-56 A-D) | B | Standard Pressure Source | Once/18 Months(9) |
| Reactor Low Water Level (LIS-3-203 A-D) | B | Pressure Standard | Once/18 Months(9) |
| High Water Level in Scram Discharge Volume | | | |
| Float Switches (LS-85-45C-F) | A | Calibrated Water Column (5) | Note (5) |
| Electronic Lvl Switches (LS-85-45-A, B, G, H) | B | Calibrated Water Column | Once/Operating Cycle (9) |
| Main Steam Line Isolation Valve Closure | A | Note (5) | Note (5) |
| Turbine First Stage Pressure Permissive (PIS-1-81A&B, PIS-1-91A&B) | B | Standard Pressure Source | Once/18 Months(9) |
| Turbine Control Valve Fast Closure or Turbine Trip | A | Standard Pressure Source | Once/Operating Cycle |
| Turbine Stop Valve Closure | A | Note (5) | Note (5) |

NOTES FOR TABLE 4.1.B

1. A description of three groups is included in the Bases of this specification.

2. Calibrations are not required when the systems are not required to be OPERABLE or are tripped. If calibrations are missed, they shall be performed prior to returning the system to an OPERABLE status.

3. (Deleted)

4. Required frequency is initial startup following each refueling outage.

5. Physical inspection and actuation of these position switches will be performed once per operating cycle.

6. On controlled startups, overlap between the IRMs and APRMs will be verified.

7. The Flow Bias Signal Calibration will consist of calibrating the sensors, flow converters, and signal offset networks during each operating cycle. The instrumentation is an analog type with redundant flow signals that can be compared. The flow comparator trip and upscale will be functionally tested according to Table 4.2.C to ensure the proper operation during the operating cycle. Refer to 4.1 Bases for further explanation of calibration frequency.

8. A complete TIP system traverse calibrates the LPRM signals to the process computer. The individual LPRM meter readings will be adjusted as a minimum at the beginning of each operating cycle before reaching 100 percent power.

9. Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the trip level setting.

## 3.1 BASES

The Reactor Protection System automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the LIMITING CONDITIONS FOR OPERATION necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection trip system is supplied, via a separate bus, by its own high inertia, ac motor-generator set. Alternate power is available to either Reactor Protection System bus from an electrical bus that can receive standby electrical power. The RPS monitoring system provides an isolation between nonclass 1E power supply and the class 1E RPS bus. This will ensure that failure of a nonclass 1E reactor protection power supply will not cause adverse interaction to the class 1E Reactor Protection System.

The Reactor Protection System is made up of two independent trip systems (refer to Section 7.2, FSAR). There are usually four channels provided to monitor each critical parameter, with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic such that either channel trip will trip that trip system. The simultaneous tripping of both trip systems will produce a reactor scram.

This system meets the intent of IEEE-279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2-out-of-3 system and somewhat less than that of a 1-out-of-2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each trip system logic has one instrument channel. When the minimum condition for operation on the number of OPERABLE instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor. Three APRM instrument channels are provided for each protection trip system.

Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast closure, and turbine stop valve closure are discussed in Specifications 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system (120/125 scram) in conjunction with the APRM system (15 percent scram) provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The discharge volume tank accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

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

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

| Minimum No. Instrument Channels Operable Per Trip Sys(1)(11) | Function | Trip Level Setting | Action (1) | Remarks |
|---|---|---|---|---|
| 2 | Instrument Channel – Low Pressure Main Steam Line (PIS-1-72, 76, 82, 86) | $\geq$ 825 psig (4) | B | 1. Below trip setting initiates Main Steam Line Isolation |
| 2(3) | Instrument Channel – High Flow Main Steam Line (PdIS-1-13A-D, 25A-D, 36A-D, 50A-D) | $\leq$ 140% of rated steam flow | B | 1. Above trip setting initiates Main Steam Line Isolation |
| 2(12) | Instrument Channel – Main Steam Line Tunnel High Temperature | $\leq$ 200°F | B | 1. Above trip setting initiates Main Steam Line Isolation. |
| 1(15) | Instrument Channel – Reactor Building Ventilation High Radiation – Reactor Zone | $\leq$ 100 mr/hr or downscale | G | 1. 1 upscale channel or 2 downscale channels will<br>a. Initiate SGTS<br>b. Isolate reactor zone and refueling floor.<br>c. Close atmosphere control system. |
| 2 | Instrument Channel Reactor Water Cleanup System Main Steam Valve Vault (TIS-069-834A-D) | $\leq$ 201.0°F | C | Above Trip Setting initiates Isolation of Reactor Water Cleanup Lines to and from the Reactor |
| 2 | Instrument Channel Reactor Water Cleanup System Pipe Trench (TIS-069-835A-D) | $\leq$ 135.0°F | C | Above Trip Setting initiates Isolation of Reactor Water Cleanup Lines to and from the Reactor |

TABLE 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

| Minimum No. Operable Per Trip Sys(1) | Function | Trip Level Setting | Action | Remarks |
|---|---|---|---|---|
| 2 | Instrument Channel - Reactor Low Water Level (LIS-3-58A-D) | ≥ 470" above vessel zero. | A | 1. Below trip setting initiates HPCI. |
| 2 | Instrument Channel - Reactor Low Water Level (LIS-3-58A-D) | ≥ 470" above vessel zero. | A | 1. Multiplier relays initiate RCIC. |
| 2 | Instrument Channel - Reactor Low Water Level (LS-3-58A-D) | ≥ 398" above vessel zero. | A | 1. Below trip setting initiates CSS.<br>Multiplier relays initiate LPCI.<br>2. Multiplier relay from CSS initiates accident signal (15). |
| 2(16) | Instrument Channel - Reactor Low Water Level (LS-3-58A-D) | ≥ 398" above vessel zero. | A | 1. Below trip settings, in conjunction with drywell high pressure, low water level permissive, ADS timer timed out and CSS or RHR pump running, initiates ADS.<br>2. Below trip settings, in conjunction with low reactor water level permissive, ADS timer timed out, ADS high drywell pressure bypass timer timed out, CSS or RHR pump running, initiates ADS. |
| 1(16) | Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184, 185) | ≥ 544" above vessel zero. | A | 1. Below trip setting permissive for initiating signals on ADS. |
| 1 | Instrument Channel - Reactor Low Water Level (LIS-3-52 and LIS-3-62A) | ≥ 312 5/16" above vessel zero. (2/3 core height) | A | 1. Below trip setting prevents inadvertent operation of containment spray during accident condition. |

TABLE 3.2.B (Continued)

| Minimum No. Operable Per Trip Sys(1) | Function | Trip Level Setting | Action | Remarks |
|---|---|---|---|---|
| 2(18) | Instrument Channel - Drywell High Pressure (PIS-64-58 E-H) | 1≤ p≤2.5 psig | A | 1. Below trip setting prevents inadvertent operation of containment spray during accident conditions. |
| 2(18) | Instrument Channel - Drywell High Pressure (PIS-64-58 A-D) | ≤ 2.5 psig | A | 1. Above trip setting in conjunction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI.<br>2. Multiplier relay from CSS initiates accident signal. (15) |
| 2(18) | Instrument Channel - Drywell High Pressure (PIS-64-58A-D) | ≤ 2.5 psig | A | 1. Above trip setting in conjunction with low reactor pressure initiates LPCI. |
| 2(16)(18) | Instrument Channel - Drywell High Pressure (PIS-64-57A-D) | ≤ 2.5 psig | A | 1. Above trip setting, in conjunction with low reactor water level, low reactor water level permissive, ADS timer timed out, and CSS or RHR pump running, initiates ADS. |

TABLE 3.2.B (Continued)

| Minimum No. Operable Per Trip Sys(1) | Function | Trip Level Setting | Action | Remarks |
|---|---|---|---|---|
| 2 | Instrument Channel - Reactor Low Pressure (PIS-3-74A&B) (PIS-68-95, 96) | 450 psig ± 15 | A | 1. Below trip setting permissive for opening CSS and LPCI admission valves. |
| 2 | Instrument Channel - Reactor Low Pressure (PS-3-74A&B) (PS-68-95, 96) | 230 psig ± 15 | A | 1. Recirculation discharge valve actuation. |
| 2 | Core Spray Auto Sequencing Timers (5) | $6 \le t \le 8$ sec. | B | 1. With diesel power<br>2. One per motor |
| 2 | LPCI Auto Sequencing Timers (5) | $0 \le t \le 1$ sec. | B | 1. With diesel power<br>2. One per motor |
| 1 | RHRSW A3, B1, C3, and D1 Timers | $13 \le t \le 15$ sec. | A | 1. With diesel power<br>2. One per pump |
| 2 | Core Spray and LPCI Auto Sequencing Timers (6) | $0 \le t \le 1$ sec.<br>$6 \le t \le 8$ sec.<br>$12 \le t \le 16$ sec.<br>$18 \le t \le 24$ sec. | B | 1. With normal power<br>2. One per CSS motor<br>3. Two per RHR motor |
| 1 | RHRSW A3, B1, C3, and D1 Timers | $27 \le t \le 29$ sec. | A | 1. With normal power<br>2. One per pump |

TABLE 3.2.B (Continued)

| Minimum No. Operable Per Trip Sys(1) | Function | Trip Level Setting | Action | Remarks |
|---|---|---|---|---|
| 2 | Instrument Channel – RHR Discharge Pressure | 100 ±10 psig | A | 1. Below trip setting defers ADS actuation. |
| 2 | Instrument Channel CSS Pump Discharge Pressure | 185 ±10 psig | A | 1. Below trip setting defers ADS actuation. |
| 1(3) | Core Spray Sparger to Reactor Pressure Vessel d/p | 2 psid ±0.4 | A | 1. Alarm to detect core spray sparger pipe break. |
| 1 | RHR (LPCI) Trip System bus power monitor | N/A | C | 1. Monitors availability of power to logic systems. |
| 1 | Core Spray Trip System bus power monitor | N/A | C | 1. Monitors availability of power to logic systems. |
| 1 | ADS Trip System bus power monitor | N/A | C | 1. Monitors availability of power to logic systems and valves. |

TABLE 3.2.B (Continued)

| Minimum No. Operable Per Trip Sys(1) | Function | Trip Level Setting | Action | Remarks |
|---|---|---|---|---|
| 1 | HPCI Trip System bus power monitor | N/A | C | 1. Monitors availability of power to logic systems. |
| 1 | RCIC Trip System bus power monitor | N/A | C | 1. Monitors availability of power to logic systems. |
| 1(2) | Instrument Channel - Condensate Header Low Level (LS-73-56A & B) | $\geq$ Elev. 551' | A | 1. Below trip setting will open HPCI suction valves to the suppression chamber. |
| 2(2) | Instrument Channel - Suppression Chamber High Level | $\leq$ 7" above instrument zero | A | 1. Above trip setting will open HPCI suction valves to the suppression chamber. |
| 2(2) | Instrument Channel - Reactor High Water Level (LIS-3-208A and LIS-3-208C) | $\leq$ 583" above vessel zero | A | 1. Above trip setting trips RCIC turbine. |
| 1 | Instrument Channel - RCIC Turbine Steam Line High Flow (PDIS-71-1A and 1B) | $\leq$ 450" $H_2O$ (7) | A | 1. Above trip setting isolates RCIC system and trips RCIC turbine. |
| 3(2) | Instrument Channel - RCIC Steam Supply Pressure - Low (PS 71-1A-D) | $\geq$50 psig | A | 1. Below trip setting isolates RCIC system and trips RCIC turbine. |
| 3(2) | Instrument Channel - RCIC Turbine Exhaust Diaphragm Pressure - High (PS 71-11A-D) | $\leq$ 20 psig | A | 1. Above trip setting isolates RCIC system and trips RCIC turbine. |

TABLE 3.2.B (Continued)

| Minimum No. Operable Per Trip Sys(1) | Function | Trip Level Setting | Action | Remarks |
|---|---|---|---|---|
| 2(2) | Instrument Channel - Reactor High Water Level (LIS-3-208B and LIS-3-208D) | ≤583" above vessel zero. | A | 1. Above trip setting trips HPCI turbine. |
| 1 | Instrument Channel - HPCI Turbine Steam Line High Flow (PDIS-73-1A and 1B) | ≤90 psi (7) | A | 1. Above trip setting isolates HPCI system and trips HPCI turbine. |
| 3(2) | Instrument Channel - HPCI Steam Supply Pressure - Low (PS 73-1A-D) | ≥100 psig | A | 1. Below trip setting isolates HPCI system and trips HPCI turbine. |
| 3(2) | Instrument Channel - HPCI Turbine Exhaust Diaphragm (PS 73-20A-D) | ≤20 psig | A | 1. Above trip setting isolates HPCI system and trips HPCI turbine. |
| 1 | Core Spray System Logic | N/A | B | 1. Includes testing auto initiation inhibit to Core Spray Systems in other units. |
| 1 | RCIC System (Initiating) Logic | N/A | B | 1. Includes Group 7 valves.<br>2. Group 7: A Group 7 isolation is automatically actuated by only the following condition:<br>1. The respective turbine steam supply valve not fully closed. |
| 1 | RCIC System (Isolation) Logic | N/A | B | 1. Includes Group 5 valves.<br>2. Group 5: A Group 5 isolation is actuated by any of the following conditions:<br>a. RCIC Steamline Space High Temperature<br>b. RCIC Steamline High Flow<br>c. RCIC Steamline Low Pressure<br>d. RCIC Turbine Exhaust Diaphragm High Pressure |
| 1 (16) | ADS Logic | N/A | A | |

NOTES FOR TABLE 3.2.B

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

   Action:

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

   B. Declare the system or component inoperable.

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

   D. No action required; indicators are considered redundant.

   E. Within 24 hours restore the inoperable channel(s) to OPERABLE status or place the inoperable channel(s) in the tripped condition.

2. In only one trip system.

3. Not considered in a trip system.

4. Deleted.

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

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

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

8. Note 1 does not apply to this item.

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

NOTES FOR TABLE 3.2.B (Continued)

10. Only one trip system for each cooler fan.

11. In only two of the four 4160-V shutdown boards. See note 13.

12. In only one of the four 4160-V shutdown boards. See note 13.

13. An emergency 4160-V shutdown board is considered a trip system.

14. RHRSW pump would be inoperable. Refer to Section 4.5.C for the requirements of a RHRSW pump being inoperable.

15. The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level ($\geq$ 398" above vessel zero) originating in the core spray system trip system.

16. The ADS circuitry is capable of accomplishing its protective action with one OPERABLE trip system. Therefore, one trip system may be taken out of service for functional testing and calibration for a period not to exceed eight hours.

17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 30 percent within four hours.

18. Not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

TABLE 3.2.F

Surveillance Instrumentation

| Minimum # of Operable Instrument Channels | Instrument # | Instrument | Type Indication and Range | Notes |
|---|---|---|---|---|
| 2 | LI-3-58A<br>LI-3-58B | Reactor Water Level | Indicator - 155" to +60" | (1) (2) (3) |
| 2 | PI-3-74A<br>PI-3-74B | Reactor Pressure | Indicator 0-1200 psig | (1) (2) (3) |
| 2 | XR-64-50<br>PI-64-67 | Drywell Pressure | Recorder -15 to +65 psig<br>Indicator -15 to +65 psig | (1) (2) (3) |
| 2 | TI-64-52<br>XR-64-50 | Drywell Temperature | Recorder, Indicator 0-400°F | (1) (2) (3) |
| 1 | XR-64-52 | Suppression Chamber Air Temperature | Recorder 0-400°F | (1) (2) (3) |
| 1 | N/A | Control Rod Position | 6V Indicating Lights ) | (1) (2) (3) (4) |
| 1 | N/A | Neutron Monitoring | SRM, IRM, LPRM 0 to 100% power ) | |
| 1 | PS-64-67 | Drywell Pressure | Alarm at 35 psig ) | (1) (2) (3) (4) |
| 1 | XR-64-50 and<br>PS-64-58 B and<br>IS-64-67 | Drywell Temperature and Pressure and Timer | Alarm if temp. > 281°F and pressure >2.5 psig after 30 minute delay ) | |
| 1 | LI-84-2A | CAD Tank "A" Level | Indicator 0 to 100% | (1) |
| 1 | LI-84-13A | CAD Tank "B" Level | Indicator 0 to 100% | (1) |

TABLE 3.2.F (cont'd)

Surveillance Instrumentation

| Minimum # of Operable Instrument Channels | Instrument # | Instrument | Type Indication and Range | Notes |
|---|---|---|---|---|
| 2 | $H_2M$ - 76 - 94<br>$H_2M$ - 76 - 104 | Drywell and Torus Hydrogen Concentration | 0.1 - 20% | (1) |
| 2 | PdI-64-137<br>PdI-64-138 | Drywell to Suppression Chamber Differential Pressure | Indicator<br>0 to 2 psid | (1) (2) (3) |
| 1/Valve | | Relief Valve Tailpipe Thermocouple Temperature or Acoustic Monitor on Relief Valve Tailpipe | | (5) |
| 2 | LI-64-159A<br>XR-64-159 | Suppression Chamber Water Level-Wide Range | Indicator,<br>Recorder 0-240" | (1) (2) (3) |
| 2 | PI-64-160A<br>XR-64-159 | Drywell Pressure Wide Range | Indicator, Recorder)<br>0-300 psig ) | (1) (2) (3) |
| 2 | TI-64-161<br>TR-64-161<br>TI-64-162<br>TR-64-162 | Suppression Pool Bulk Temperature | Indicator, Recorder)<br>)<br>30° - 230° F )<br>) | (1) (2) (3) (4) (6) |
| 1 | RR-90-272<br>RR-90-273<br>RM-90-272A<br>RM-90-273A | High Range Primary Containment Radiation Monitors and Recorders | Monitor, Recorder<br>1 - $10^7$ R/Hr | (7) (8) |
| 1 | RM-90-306<br>RR-90-360 | Wide Range Gaseous Effluent Radiation Monitor and Recorder | Monitor, Recorder<br>(Noble Gas<br>$10^{-7}$ - $10^{+5}$ μCi/cc) | (7)(8)(9) |



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NOTES FOR TABLE 3.2.K

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*(Deleted)
**(Deleted)
***During main condenser offgas treatment system operation

ACTION A

(Deleted)

ACTION B

(Deleted)

ACTION C

(Deleted)

ACTION D

(Deleted)

ACTION E

With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of main condenser offgas treatment system may continue provided that a temporary monitor is installed or grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION F

(Deleted)

Table 3.2.L
Anticipated Transient Without Scram (ATWS) -
Recirculation Pump Test (RPT) Surveillance Instrumentation

| Minimum No. Channels operable per Trip Sys (1) | Function | Trip Setting | Allowable Value | Action | Remarks |
|---|---|---|---|---|---|
| 2 | ATWS/RPT Logic Reactor Dome Pressure High (PIS-3-204A-D) | 1118 psig | ≤ 1146.5 psig | (2) | Two out of two of the high reactor dome pressure channels or the low reactor vessel level channels in either trip system trips both reactor recirculation pumps. |
| 2 | Reactor Vessel Level Low (LS-3-58 A1-D1) | 483" above vessel zero | ≥ 471.52" above vessel zero | | |

(1) One channel in only one trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided the other channels in that trip system are OPERABLE.

(2) Two trip systems exist, either of which will trip both recirculation pumps. Perform Surveillance/maintenance/calibration on one channel in only one trip system at a time. If a channel is found to be inoperable or if the surveillance/maintenance/calibration period for one channel exceeds 6 consecutive hours, the trip system will be declared inoperable or the channel will be placed in a tripped condition. If in RUN mode and one trip system is inoperable for 72 hours or both trip systems are inoperable, the reactor shall be in at least the HOT STANDBY CONDITION within 6 hours.

TABLE 4.2.A
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

| Function | Functional Test | Calibration Frequency | Instrument Check |
|---|---|---|---|
| Instrument Channel - Reactor Low Water Level (LIS-3-203A-D) | (1)(28) | once/18 months (29) | once/day |
| Instrument Channel - Reactor High Pressure (PS-68-93 & -94) | (31) | once/18 months | None |
| Instrument Channel - Reactor Low Water Level (LIS-3-56A-D) | (1)(28) | once/18 months (29) | once/day |
| Instrument Channel - High Drywell Pressure (PIS-64-56A-D) | (1)(28) | once/18 months (29) | N/A |
| Instrument Channel - Low Pressure Main Steam Line (PIS-1-72, 76, 82, 86) | (28)(27) | once/18 months (29) | None |
| Instrument Channel - High Flow Main Steam Line (PdIS-1-13A-D, 25A-D, 36A-D, 50A-D) | (28)(27) | once/18 months (29) | once/day |

TABLE 4.2.A
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

| Function | Functional Test | Calibration Frequency | Instrument Check |
|---|---|---|---|
| Instrument Channel - Main Steam Line Tunnel High Temperature | once/3 months (27) | once/operating cycle | None |
| Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone | (1) (30) | once/18 months | once/day (8) |
| Instrument Channel - Reactor Building Ventilation High Radiation - Refueling Zone | (1) (30) | once/18 Months | once/day (8) |
| Instrument Channel - SGTS Train A Heaters | (4) | (9) | N/A |
| Instrument Channel - SGTS Train B Heaters | (4) | (9) | N/A |
| Instrument Channel - SGTS Train C Heaters | (4) | (9) | N/A |
| Reactor Building Isolation Timer (refueling floor) | (4) | once/operating cycle | N/A |
| Reactor Building Isolation Timer (reactor zone) | (4) | once/operating cycle | N/A |

TABLE 4.2.B
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

| Function | Functional Test | Calibration | Instrument Check |
|---|---|---|---|
| Instrument Channel - Reactor Low Water Level (LS-3-58A-D, LIS-3-58A-D) | (1)(28) | once/18 months(29) | once/day |
| Instrument Channel - Reactor Low Water Level (LIS-3-184 & 185) | (1)(28) | once/18 months(29) | once/day |
| Instrument Channel - Reactor Low Water Level (LIS-3-52 & 62A) | (1)(28) | once/18 months(29) | once/day |
| Instrument Channel - Drywell High Pressure (PIS-64-58E-H) | (1)(28) | once/18 months(29) | none |
| Instrument Channel - Drywell High Pressure (PIS-64-58A-D) | (1)(28) | once/18 months(29) | none |
| Instrument Channel - Drywell High Pressure (PIS-64-57A-D) | (1)(28) | once/18 months(29) | none |
| Instrument Channel - Reactor Low Pressure (PIS-3-74A & B, PS-3-74A & B) (PIS-68-95, PS-68-95) (PIS-68-96, PS-68-96) | (1)(28) | once/6 months(29) | none |

TABLE 4.2.B (Cont'd)
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

| Function | Functional Test | Calibration | Instrument Check |
|---|---|---|---|
| Core Spray Auto Sequencing Timers (Normal Power) | (4) | once/operating cycle | none |
| Core Spray Auto Sequencing Timers (Diesel Power) | (4) | once/operating cycle | none |
| LPCI Auto Sequencing Timers (Normal Power) | (4) | once/operating cycle | none |
| LPCI Auto Sequencing Timers (Diesel Power) | (4) | once/operating cycle | none |
| RHRSW A3, B1, C3, D1 Timers (Normal Power) | (4) | once/operating cycle | none |
| RHRSW A3, B1, C3, D1 Timers (Diesel Power) | (4) | once/operating cycle | none |
| ADS Timer | (4) | once/operating cycle | none |
| ADS High Drywell Pressure Bypass Timer | (4) | once/operating cycle | none |



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TABLE 4.2.B (Cont'd)
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

| Function | Functional Test | Calibration | Instrument Check |
|---|---|---|---|
| Instrument Channel - RHR Pump Discharge Pressure | (1) | once/3 months | none |
| Instrument Channel - Core Spray Pump Discharge Pressure | (1) | once/3 months | none |
| Core Spray Sparger to RPV d/p | (1) | once/3 months | once/day |
| Trip System Bus Power Monitor | once/operating Cycle | N/A | none |
| Instrument Channel - Condensate Header Level (LS-73-56A, B) | (1) | once/3 months | none |
| Instrument Channel - Suppression Chamber High Level | (1) | once/3 months | none |
| Instrument Channel - Reactor High Water Level (LIS-3-208A-D) | (1)(28) | once/18 months (29) | once/day |
| Instrument Channel - RCIC Turbine Steam Line High Flow | (1)(28) | once/18 months (29) | none |
| Instrument Channel - RCIC Steam Supply Low Pressure | once/31 days | once/18 months | once/day |
| Instrument Channel - RCIC Turbine Exhaust Diaphragm High Pressure | once/31 days | once/18 months | once/day |
| RCIC Steam Line Space Torus Area High Temperature | (1) | once/3 months | none |
| RCIC Steam Line Space RCIC Pump Room Area High Temperature | (1) | once/3 months | none |
| HPCI Steam Line Space Torus Area High Temperature | (1) | once/3 months | none |
| HPCI Steam Line Space HPCI Pump Room Area High Temperature | (1) | once/3 months | none |

TABLE 4.2.B (Cont'd)
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

| Function | Functional Test | Calibration | Instrument Check |
|---|---|---|---|
| Instrument Channel - HPCI Turbine Steam Line High Flow | (1)(28) | once/18 months (29) | none |
| Instrument Channel - HPCI Steam Supply Low Pressure | once/31 days | once/18 months | once/day |
| Instrument Channel - HPCI Turbine Exhaust Diaphragm High Pressure | once/31 days | once/18 months | once/day |
| Core Spray System Logic | once/18 months | (6) | N/A |
| RCIC System (Initiating) Logic | once/18 months | N/A | N/A |
| RCIC System (Isolation) Logic | once/18 months | (6) | N/A |
| HPCI System (Initiating) Logic | once/18 months | (6) | N/A |
| HPCI System (Isolation) Logic | once/18 months | (6) | N/A |
| ADS Logic | once/18 months | (6) | N/A |
| LPCI (Initiating) Logic | once/18 months | (6) | N/A |
| LPCI (Containment Spray) Logic | once/18 months | (6) | N/A |

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

| Instrument Channel | Calibration Frequency | Instrument Check |
|---|---|---|
| 1) Reactor Water Level (LI-3-58A & B) | Once/18 months | Each Shift |
| 2) Reactor Pressure (PI-3-74A & B) | Once/6 months | Each Shift |
| 3) Drywell Pressure | Once/6 months | Each Shift |
| 4) Drywell Temperature | Once/6 months | Each Shift |
| 5) Suppression Chamber Air Temperature | Once/6 months | Each Shift |
| 8) Control Rod Position | N/A | Each Shift |
| 9) Neutron Monitoring | (2) | Each Shift |
| 10) Drywell Pressure (PS-64-67) | Once/6 months | N/A |
| 11) Drywell Pressure (PIS-64-58A) | Once/18 months | N/A |
| 12) Drywell Temperature (TR-64-52) | Once/6 months | N/A |
| 13) Timer (IS-64-67) | Once/6 months | N/A |
| 14) CAD Tank Level | Once/6 months | Once/day |
| 15) Containment Atmosphere Monitors | Once/6 months | Once/day |

TABLE 4.2.F (Cont'd)
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

| Instrument Channel | Calibration Frequency | Instrument Check |
|---|---|---|
| 16) Drywell to Suppression Chamber Differential Pressure | Once/6 months | Each Shift |
| 17) Relief Valve Tailpipe Thermocouple Temperature | N/A | Once/month (24) |
| 18) Acoustic Monitor on Relief Valve Tailpipe | Once/cycle (25) | Once/month (26) |
| 19) Suppression Chamber Water Level-Wide Range (LI-64-159A) (XR-64-159) | Once/cycle | Once/month |
| 20) Drywell Pressure - Wide Range (PI-64-160A) (XR-64-159) | Once/cycle | Once/shift |
| 21) Suppression Pool Bulk Temperature (TI-64-161) (TR-64-161) (TI-64-162) (TR-64-162) | Once/cycle | Once shift |
| 22) High Range Primary Containment Radiation Monitors and Recorders (RR-90-272, RR-90-273, RM-90-272A, RM-90-273A) | Once/18 months (32) | Once/month |
| 23) Wide Range Gaseous Effluent Radiation Monitor and Recorder (RM-90-306 and RR-90-360) | Once/18 months | Once/shift |



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## 3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever PRIMARY CONTAINMENT INTEGRITY is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 538 inches above vessel zero closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Groups 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 470 inches above vessel zero (Table 3.2.B) trips the recirculation pumps and initiates the RCIC and HPCI systems. The RCIC and HPCI system initiation opens the turbine steam supply valve which in turn initiates closure of the respective drain valves (Group 7).

The low water level instrumentation set to trip at $\geq$398 inches above vessel zero (Table 3.2.B) closes the Main Steam Isolation Valves, the Main Steam Line Drain Valves, and the Reactor Water Sample Valves (Group 1). These trip settings are adequate to prevent core uncovery in the case of a break in the largest line assuming the maximum closing time.

The low reactor water level instrumentation that is set to trip when reactor water level is ≥398 inches above vessel zero (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation, and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and, in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus, the results given above are applicable here also.

ADS provides for automatic nuclear steam system depressurization, if needed, for small breaks in the nuclear system so that the LPCI and the CSS can operate to protect the fuel from overheating. ADS uses six of the 13 MSRVs to relieve the high pressure steam to the suppression pool. ADS initiates when the following conditions exist: low reactor water level permissive (level 3), low reactor water level (level 1), high drywell pressure or the ADS high drywell pressure bypass timer timed out, and the ADS timer timed out. In addition, at least one RHR pump or two core spray pumps must be running.

The ADS high drywell pressure bypass timer is added to meet the requirements of NUREG 0737, Item II.K.3.18. This timer will bypass the high drywell pressure permissive after a sustained low water level. The worst case condition is a main steam line break outside primary containment with HPCI inoperable. With the ADS high drywell pressure bypass timer analytical limit of 360 seconds, a Peak Cladding Temperature (PCT) of 1500°F will not be exceeded for the worst case event. This temperature is well below the limiting PCT of 2200°F.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the high steam flow instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

j. Continuous Leak Rate Monitoring

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

k. The interior surfaces of the drywell and torus above the level one foot below the normal water line and outside surfaces of the torus below the water line shall be visually inspected each operating cycle for deterioration and any signs of structural damage with particular attention to piping connections and supports and for signs of distress or displacement.

LIMITING CONDITIONS FOR OPERATION

3.7.A PRIMARY CONTAINMENT

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.

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

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

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

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

SURVEILLANCE REQUIREMENTS

4.7.A PRIMARY CONTAINMENT

3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.

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

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.

b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

## TABLE 3.7.A

## INSTRUMENTATION FOR CONTAINMENT SYSTEMS

| Minimum No. Operable Per Trip System | Function | Trip Level Setting | Action | Remarks |
|---|---|---|---|---|
| 2 | Instrument Channel - Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21) | 0.5 psid | (1) | Actuates the pressure suppression chamber-reactor building vacuum breakers. |

Footnote:

(1) - Repair in 24 hours. If the function is not OPERABLE in 24 hours, declare the system or component inoperable.

TABLE 4.7.A

CONTAINMENT SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| Function | Functional Test | Calibration | Instrument Check |
|---|---|---|---|
| Instrument Channel- Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21) | Once/month(1) | Once/18 months(2) | None |

Footnotes:

(1) - Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify OPERABILITY of the trip and alarm functions.

(2) - Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the level setting.

containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after two years of operation in the rugged shipboard environment on the US Savannah (ORNL 3726). Pressure drop across the combined HEPA filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow ASTM D3803-89. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

All elements of the heater should be demonstrated to be functional and OPERABLE during the test of heater capacity. Operation of each filter train for a minimum of 10 hours each month will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repaired and test repeated.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability and OPERABILITY of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other systems must be tested daily. This substantiates the availability of the OPERABLE systems and thus reactor operation and refueling operation can continue for a limited period of time.

3.7.D/4.7.D Primary Containment Isolation Valves

The Browns Ferry Containment Leak Rate Program and Procedures contains the list of all the Primary Containment Isolation Valves for which the Technical Specification requirements apply. The procedures are subject to the change control provisions for plant procedures in the administrative controls section of the Technical Specifications. The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a LOCA.

Group 1 - Process lines are isolated by reactor vessel low water level ($\geq$398") in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in Group 1, except the reactor water sample line valves, are also closed when process instrumentation detects excessive main steam line flow, low pressure, or main steam space high temperature. The reactor water sample line valves isolate only on reactor low water level at $\geq$398".

Group 2 - Isolation valves are closed by reactor vessel low water level (538") or high drywell pressure. The Group 2 isolation signal also "isolates" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the Group 2 isolation signal by a transient or spurious signal.

Group 3 - Process lines are normally in use, and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from nonsafety related causes. To protect the reactor from a possible pipe break