Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

January 10, 2006

TVA-BFN-TS-453

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN P1-35
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

BROWNS FERRY NUCLEAR PLANT (BFN) UNITS 1, 2, AND 3 - TECHNICAL SPECIFICATIONS (TS) CHANGE TS-453 - INSTRUMENT SETPOINT PROGRAM

Pursuant to 10 CFR 50.90, Tennessee Valley Authority (TVA) is submitting a request for a TS change (TS-453) to licenses DPR-33, DPR-52, and DPR-68 for Units 1, 2, and 3, respectively. The scope of this proposed TS change includes those drift susceptible instruments, which are either necessary to ensure compliance with a Safety Limit or critical in ensuring the fuel peak cladding temperature acceptance criterion of 10 CFR 50.46 are met. The proposed change specifies the methodology used for determining, setting, and evaluating as-found setpoints for these instruments.

NRC expressed concerns regarding TVA's setpoint methodology in Reference 1. NRC stated that these concerns must be addressed as part of the review of several proposed TS changes (References 1 and 2). In order to resolve these NRC concerns, TVA proposes to add a footnote that specifies the actions to be taken for the applicable as-found instrument setpoints and references a discussion of the NRC-approved TVA setpoint methodology. The purpose of including this information in the TS is to control critical instrument setpoints and ensure compliance with 10 CFR 50.36. In addition, TVA will evaluate the final Technical Specification Task Force change related to resolution of the setpoint issue within 90 days after its approval by the NRC. With the submittal of this proposed TS,

A001

TVA considers the NRC's constraint should be resolved for the following proposed TS:

- TS-418, Units 2 and 3 - Extended Power Uprate Operation;
- TS-430, Unit 1 - Power Range Neutron Monitor Upgrade;
- TS-431, Unit 1 - Extended Power Uprate Operation;
- TS-433, Unit 1 - 24 Month Fuel Cycle;
- TS-434, Unit 1 - Allowable Value for Reactor Vessel Water Level - Low Level 3;
- TS-437, Unit 1 - Scram Discharge Instrument Volume Setpoint Change; and
- TS-447, Units 1, 2 and 3 - Calibration Interval Extension for HPCI/RCIC Temperature Switches.

TVA requests this amendment be approved expeditiously and that the implementation of this revised TS be within 90 days of NRC approval. The 90 day period is necessary due to the number of surveillance procedures that require revision in order to implement this change.

TVA has determined there are no significant hazards considerations associated with the proposed TS change and the change qualifies for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosures to the Alabama State Department of Public Health.

Enclosure 1 provides TVA's evaluation of the proposed changes. Enclosure 2 provides a mark-up of the proposed TS changes. Enclosure 3 provides a mark-up of the proposed TS Bases changes. Enclosure 4 provides a summary of the regulatory commitments associated with this submittal.

If you have any questions about this amendment, please contact me at (256) 729-2636.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 10, 2006.

Sincerely,

William D. Crouch

William D. Crouch
Manager of Licensing
and Industry Affairs

References:

1. NRC letter to TVA, dated January 6, 2005, "Browns Ferry Units 1, 2, and 3 - Request for Information Regarding Status of Amendments Using Method 3 (TAC Nos. MC1330, MC1427, MC2305, MC3812, MC4070, MC4071, MC4072, MC4161, MC3743, and MC3744)."

2. NRC letter to TVA, dated April 19, 2005, "Browns Ferry Nuclear Plant, Unit 1 - Licensing Action Status and Interdependencies (TAC Nos. MC1330, MC1427, MC2305, MC3812, MC3813, MC3822, MC3960, MC4070, MC4161, MC4659, MC4797, MC5254, and MC5373)."

Enclosures:
1. TVA Evaluation of the Proposed Changes
2. Proposed Technical Specification Changes (mark-up)
3. Changes to Technical Specification Bases Pages (mark-up)
4. List of Regulatory Commitments

Enclosures
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Enclosure 1

Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 Technical Specifications (TS) Change TS-453

Instrument Setpoint Program

TVA Evaluation of Proposed Changes

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1.0 DESCRIPTION

This letter requests a TS change (TS-453) to licenses DPR-33, DPR-52, and DPR-68 for Units 1, 2, and 3, respectively. The scope of this proposed TS change includes those drift susceptible instruments, which are either necessary to ensure compliance with a Safety Limit or critical in ensuring the fuel peak cladding temperature acceptance criterion of 10 CFR 50.46 are met. The proposed change adds a footnote that specifies the actions to be taken for the applicable as-found instrument setpoints and references a discussion of the NRC-approved TVA setpoint methodology. In addition, TVA will evaluate the final Technical Specification Task Force change related to resolution of the setpoint issue within 90 days after its approval by the NRC.

TVA requests the amendment be approved expeditiously in order to allow the processing of the other associated proposed TS changes and that the implementation of the revised TS be within 90 days of NRC approval.

2.0 PROPOSED CHANGE

The proposed change affects drift susceptible instruments, which are either necessary to ensure compliance with a Safety Limit or critical in ensuring the fuel peak cladding temperature acceptance criterion of 10 CFR 50.46 are met. The proposed change adds a footnote that specifies the actions to be taken for the applicable as-found instrument setpoints and references a discussion of the NRC-approved TVA setpoint methodology. The footnote references a new section which will be included in the Updated Final Safety Analysis Report (UFSAR) prior to implementation of the proposed TS change. This new section will summarize the methodology used for determining, setting and evaluating as-found instrument setpoints. The specific TS changes are listed below:

A. In Units 1, 2, and 3 TS 3.3.1.1, Reactor Protection System (RPS) Instrumentation, Table 3.3.1.1-1, "Reactor Protection System Instrumentation," a new footnote will be added to the following instrument functions:

   3. Reactor Vessel Steam Dome Pressure - High;

   4. Reactor Vessel Water Level - Low, Level 3; and

   9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low.

The footnote will state:

> "During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.
>
> Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.
>
> The nominal Trip Setpoint shall be specified on design output documentation. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band shall be specified in Chapter 7 of the Updated Final Safety Analysis Report."

B. In Units 1, 2, and 3 TS 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation, Table 3.3.5.1-1, "Emergency Core Cooling System Instrumentation," a new footnote will be added to the following instrument functions:

1. Core Spray System

    a. Reactor Vessel Water Level - Low Low Low, Level 1;

    b. Drywell Pressure - High; and

    c. Reactor Steam Dome Pressure - Low (Injection Permissive and ECCS Initiation).

2. Low Pressure Coolant Injection (LPCI) System

   a. Reactor Vessel Water Level - Low Low Low, Level 1;

   b. Drywell Pressure - High;

   c. Reactor Steam Dome Pressure - Low (Injection Permissive and ECCS Initiation); and

   d. Reactor Steam Dome Pressure - Low (Recirculation Discharge Valve Permissive).

3. High Pressure Coolant Injection (HPCI) System

   a. Reactor Vessel Water Level - Low Low, Level 2; and

   b. Drywell Pressure - High.

4. Automatic Depressurization System (ADS) Trip System A

   a. Reactor Vessel Water Level - Low Low Low, Level 1;

   b. Drywell Pressure - High; and

   d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory).

5. Automatic Depressurization System (ADS) Trip System B

   a. Reactor Vessel Water Level - Low Low Low, Level 1;

   b. Drywell Pressure - High; and

   d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory).

The footnote will be the same as above.

C. In Units 1, 2, and 3 TS 3.3.5.2, Reactor Core Isolation Cooling (RCIC) System Instrumentation, Table 3.3.5.2-1, "Reactor Core Isolation Cooling System Instrumentation," a new footnote will be added to the following instrument function:

   1. Reactor Vessel Water Level - Low Low, Level 2.

   The footnote will be the same as above.

D. In Units 1, 2, and 3 TS 3.3.6.1, Primary Containment Isolation Instrumentation, Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," a new footnote will be added to the following Main Steam Line Isolation instrument function:

   1.b Main Steam Line Pressure - Low.

   The footnote will be the same as above.

A mark-up of the TS showing the proposed changes is provided in Enclosure 2. A mark-up of the TS Bases showing the proposed changes is provided in Enclosure 3.

3.0 BACKGROUND

3.1 Reason for the Proposed Changes

The underlying reason for the Units 1, 2, and 3 TS change is to resolve NRC concerns regarding TVA's setpoint methodology. NRC expressed concerns regarding TVA's setpoint methodology in Reference 1. NRC has stated that these concerns must be addressed as part of the review of several proposed TS amendments (References 1 and 2). Including this information in the TS will ensure control of critical instrument setpoints and compliance with 10 CFR 50.36.

Analytical Limits represent the values used in the safety analyses to demonstrate that automatic protective actions prevent the plant from exceeding a Safety Limit or 10 CFR 50.46 limit. Typically, the Analytical Limits do not account for instrument characteristics such as drift, repeatability, accident induced error, etc. These phenomena are accounted for in the instrument setpoint calculations. Instrument setpoint and scaling calculations utilize the Analytical Limits to establish the nominal Trip Setpoint, Acceptable As Left (AAL) band, Acceptable As Found (AAF) band and the Allowable Value. The Allowable Value is the value in the TS. If the instrument actuates under normal plant conditions or during a surveillance test at or before the

Allowable Value, the setpoint calculation demonstrates that the instrument would actuate at or before the Analytical Limit under transient or accident conditions, and thus the Safety Limit or 10 CFR 50.46 limit would not be exceeded.

The AAL band accounts for uncertainties such as drift and repeatability so that an instrument would be expected to remain within the AAF band when tested at the end of the next surveillance interval. The AAF band represents the expected band of instrument performance. If an instrument is outside the AAF band, but conservative with respect to the Allowable Value, it would still perform its as designed function and thus could be considered operable, but would be evaluated to determine why its performance is outside the expected AAF band. The initial evaluation, which is performed prior to returning the instrument to operation, would be performed to show that the instrument is not degrading such that it might not function as designed during the next interval of operation.

Instruments which perform outside the Allowable Value during surveillance testing may not be able to perform its design function during a transient or an accident, and thus would be declared inoperable until actions are taken to ensure the channel will perform as designed.

Requiring the channel setpoint to be reset to a value that is within the acceptable as-left tolerance after any instrument calibration is necessary to ensure the channel is in conformance with the assumptions of the supporting instrument setpoint and scaling calculation.

3.2 Safety Limits and 10 CFR 50.46 Requirements

Safety Limits are defined in Section 2.1 of the TS as:

- Reactor Core Safety Limits:
    - With the reactor steam dome pressure less than 785 psig or core flow less than 10 percent rated core flow, thermal power shall be less than or equal to 25 percent rated thermal power (Safety Limit 2.1.1.1).
    - With the reactor steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10 percent rated core flow, the minimum value of the critical power ratio (MCPR) shall be greater than or equal to (*reload specific value*) for two recirculation loop operation or greater than or equal to (*reload specific value*) for single loop operation (Safety Limit 2.1.1.2).

- Reactor vessel water level shall be greater than the top of active irradiated fuel (Safety Limit 2.1.1.3).

- Reactor Coolant System (RCS) Pressure Safety Limit:

  - Reactor steam dome pressure shall be less than or equal to 1325 psig (Safety Limit 2.1.2).

With regards to the reactor core Safety Limits, operation with core thermal power below 25% of rated without thermal margin surveillance is conservatively acceptable for complying with the Safety Limits even for reactor operations at natural circulation. Adequate fuel thermal margins are also maintained without further surveillance for the low power conditions that would be present if core natural circulation is below 10% of rated flow.

Critical power correlations are applicable for calculations at pressure greater than 785 psig and core flow greater than 10 percent rated core flow. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Although it is recognized that the onset of transition boiling would not result in damage to Boiling Water Reactor fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit.

The reactor vessel water level Safety Limit has been established at the top of the active fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

With regards to the RCS pressure Safety Limit, reactor steam dome pressure protects the reactor coolant system against overpressurization. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity.

10 CFR 50.46 provides the acceptance criteria for ECCS for light-water nuclear power reactors. 10 CFR 50.46(b)(1) requires the calculated maximum fuel element cladding temperature not exceed 2200°F.

3.3 Protection Function Description

Protection functions are designed to initiate appropriate responses from systems to ensure that Safety Limits are maintained in the event of a design basis accident or transient. This is achieved by specifying Limiting Safety System Settings (LSSS), as well as Limiting Conditions for Operation (LCO) on other reactor system parameters, and equipment performance. TS

are required by 10 CFR 50.36 to contain LSSS defined by the regulation as "...settings for automatic protective devices...so chosen that automatic protective action will correct the abnormal situation before a Safety Limit (SL) is exceeded." For BFN, the LSSS are defined in TS Bases 3.3.1.1 as the Allowable Values, which, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits, including Safety Limits during design basis accidents. The Analytical Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that a Safety Limit is not exceeded. Any automatic protection action that occurs on or before reaching the Analytical Limit therefore ensures that the Safety Limit is not exceeded. However, in practice, the actual trip setpoints for automatic protective devices must be chosen to be more conservative than the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

The trip setpoint for a protective device is chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the Safety Limit would not be exceeded. As such, the trip setpoint accounts for uncertainties in setting the device (e.g., calibration), uncertainties in how the device might actually perform (e.g., repeatability), changes in the point of action of the device over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the trip setpoint ensures that Safety Limits are not exceeded.

At BFN, the nominal Trip Setpoints are specified by design documents, which require an evaluation under the provisions of 10 CFR 50.59 before they can be changed.

3.4 System Description

The instrumentation affected by this TS change involves the following systems:

- Reactor Protection System,
- Emergency Core Cooling Systems,
- Reactor Core Isolation Cooling, and
- Primary Containment Isolation.

A brief description of each system and the affected instrumentation is provided below:

Reactor Protection System

The RPS initiates a reactor scram when one or more monitored parameters exceed their specified limits, to preserve the integrity of the fuel cladding and the RCS, and minimize the energy that must be absorbed following a loss of coolant accident. This can be accomplished either automatically or manually. The protection and monitoring functions of the RPS have been designed to ensure safe operation of the reactor. This is achieved by specifying LSSSs in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance. The LSSS are defined in this specification as the Allowable Values, which in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits, including Safety Limits during Design Basis Accidents and operational transients. Additional information regarding the RPS is provided in UFSAR Section 7.2.

The drift susceptible RPS instrumentation affected by this TS change are:

- Reactor Vessel Steam Dome Pressure - High, which maintains compliance with Safety Limit 2.1.1.2;

- Reactor Vessel Water Level - Low, Level 3, which maintains compliance with Safety Limit 2.1.1.3 and the fuel peak cladding temperature acceptance criterion of 10 CFR 50.46; and

- Turbine Control Valve Fast Closure, Trip Oil Pressure - Low, which maintains compliance with Safety Limit 2.1.1.2.

Each is discussed below:

An increase in the Reactor Pressure Vessel (RPV) pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and thermal power to increase, which could challenge the integrity of the fuel cladding and the reactor coolant pressure boundary. The Reactor Vessel Steam Dome Pressure - High function initiates a scram for transients that result in a pressure increase, thereby counteracting the pressure increase by rapidly reducing core power.

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, a reactor scram is initiated at Level 3 to substantially reduce the heat generated in the fuel

from fission. The Reactor Vessel Water Level - Low, Level 3 function is assumed in the accident analysis of the recirculation line break. The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Fast closure of the Turbine Control Valves (TCVs) results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low function is the primary scram signal for the generator load rejection event. For this event, the reactor scram reduces the amount of energy required to be absorbed.

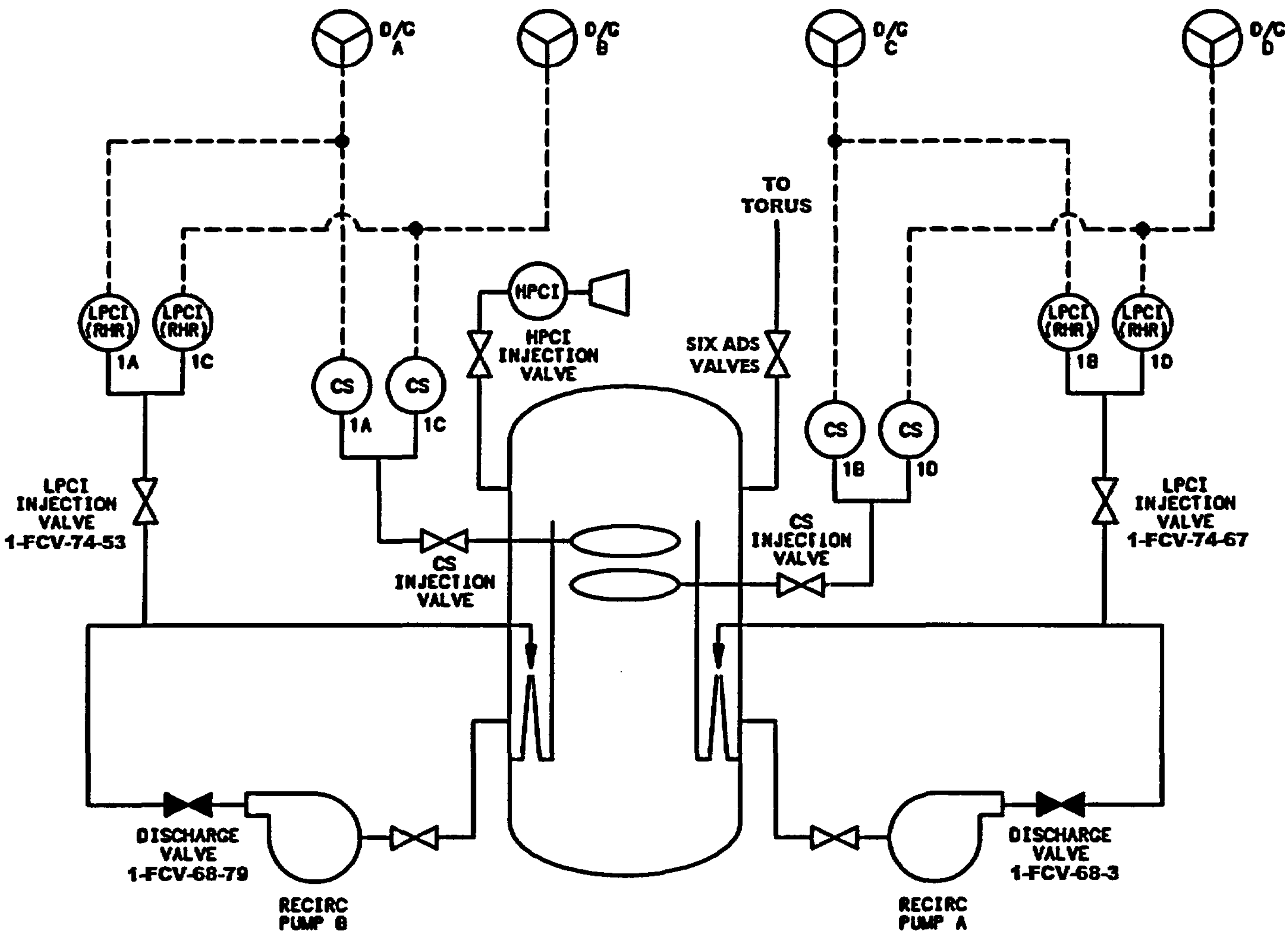
Emergency Core Cooling Systems

As discussed above, the ECCS coupled with a reactor scram ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. As shown in Figure 1, the BFN ECCS consists of the following:

- Core Spray;
- Low Pressure Coolant Injection, which is an operating mode of Residual Heat Removal (RHR) (The other operating modes of RHR include shutdown cooling, containment spray and pool cooling, standby cooling, and supplemental fuel pool cooling.);
- High Pressure Coolant Injection; and
- Automatic Depressurization System.

These systems are designed to limit clad temperature over the complete spectrum of possible break sizes in the nuclear system process barrier, including the design basis break. The design basis break is defined as the complete and sudden circumferential rupture of the largest pipe connected to the reactor vessel (i.e., one of the recirculation loop pipes) with displacement of the ends so that blowdown occurs from both ends.

FIGURE 1
LAYOUT OF THE UNIT 1 EMERGENCY CORE COOLING SYSTEM



The Units 2 and 3 design is similar.

The ECCS affected by this TS change are described below.

1. The Core Spray System instruments, which maintain compliance with the fuel peak cladding temperature acceptance criterion of 10 CFR 50.46, are:

   - Reactor Vessel Water Level - Low Low Low, Level 1;
   - Drywell Pressure - High; and
   - Reactor Steam Dome Pressure - Low (Injection Permissive and ECCS Initiation).

2. The LPCI instruments, which maintain compliance with the fuel peak cladding temperature acceptance criterion of 10 CFR 50.46, are:

   - Reactor Vessel Water Level - Low Low Low, Level 1;
   - Drywell Pressure - High;
   - Reactor Steam Dome Pressure - Low (Injection Permissive and ECCS Initiation); and
   - Reactor Steam Dome Pressure - Low (Recirculation Discharge Valve Permissive).

3. The following HPCI instruments:

   - Reactor Vessel Water Level - Low Low, Level 2, which maintains compliance with Safety Limit 2.1.1.3 and the fuel peak cladding temperature acceptance criterion of 10 CFR 50.46; and
   - Drywell Pressure - High, which maintains compliance with the fuel peak cladding temperature acceptance criterion of 10 CFR 50.46.

4. The ADS instruments, which maintain compliance with the fuel peak cladding temperature acceptance criterion of 10 CFR 50.46, are:

   - Reactor Vessel Water Level - Low Low Low, Level 1;
   - Drywell Pressure - High; and
   - Reactor Vessel Water Level - Low, Level 3 (Confirmatory).

Low RPV water level and high drywell pressure indicate that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The low pressure ECCS are initiated to ensure that core spray and flooding functions are available to prevent or minimize fuel damage. Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. This ensures that, prior to opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure.

The Core Spray system consists of two independent loops. Each loop consists of two pumps, a spray sparger inside the core shroud and above the core, piping and valves to convey water from the pressure suppression pool to the sparger, and the associated controls and instrumentation. When the system is actuated, water is taken from the pressure suppression pool. Flow then passes through a normally open motor-operated valve in the suction line to each 50 percent capacity pump.

The LPCI is an operating mode of the RHR System, with two LPCI subsystems. During LPCI operation, the four RHR pumps take suction from the pressure suppression pool and discharge to the reactor vessel into the core region through both of the recirculation loops. Two pumps discharge to each recirculation loop. Once an initiation signal is received by the LPCI control circuitry, the signal is sealed in until manually reset.

The HPCI System consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping for the system is provided from the Condensate Storage Tank (CST) and the suppression pool. The HPCI System may be initiated by either automatic or manual means.

In case the capability of the HPCI is not sufficient to maintain the reactor water level, the ADS functions to reduce the reactor pressure so that flow from the LPCI and the Core Spray System enters the reactor vessel in time to cool the core and limit fuel cladding temperature. The ADS uses six of the nuclear system main steam relief valves to relieve the high pressure steam to the pressure suppression pool.

Additional information regarding the ECCS is provided in UFSAR Chapter 6.

Reactor Core Isolation Cooling

The RCIC System is designed to operate either automatically or manually following RPV isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of the RPV water level. Under these conditions, the HPCI and RCIC systems perform similar functions.

The RCIC System consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping is provided from the CST and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV.

The RCIC instrumentation function affected by this TS change is the Reactor Vessel Water Level - Low Low, Level 2, which maintains compliance with Safety Limit 2.1.1.3. Additional information regarding the RCIC system is provided in UFSAR Section 4.7.

Primary Containment Isolation

Low main steam line pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The primary containment isolation system instrumentation affected by this TS change is the Main Steam Line Pressure - Low function, which is directly assumed in the analysis of the pressure regulator failure. For this event, the closure of the Main Steam Isolation Valves (MSIVs) ensures that the RPV temperature change limit (100 degrees F/hr) is not reached. In addition, this function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. This function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram.

Additional information regarding the Primary Containment Isolation system is provided in UFSAR Section 7.3.

4.0 TECHNICAL ANALYSIS

4.1 Setpoint Methodology

As evidenced below, TVA's method for performing setpoint calculations and the implementing programmatic controls were previously explicitly reviewed and approved by NRC as part of the BFN docket.

Prior to Unit 2 restart, TVA developed its current setpoint calculation methodology based on Method 3 of Instrument Society of America (ISA) S67.04.02. NRC (including NRR staff) performed an inspection (Reference 3) to assess the adequacy of the testing, calibration, maintenance, and configuration control of safety-related instrumentation. Section 5 of Inspection Report 89-06 states:

> "The latest procedure used by the licensee for setpoint calculations is the Division of Nuclear Engineering (DNE), Electrical Engineering Branch (EEB), instruction EEB-TI-28, Revision 1, dated October 24, 1988. … Procedure EEB-TI-28 incorporates the guidance found in RG 1.105 and ISA Standard 67.04 and is acceptable for assuring that setpoints are established and held within specified limits for nuclear safety-related instruments used in nuclear power plants. The guidance provided by this procedure was reflected in the setpoint calculations which were reviewed during this inspection and are identified in the scope paragraph. The methodology of determining instrument loop errors and using them in the accuracy calculation reviewed is acceptable."

In order to support the restart of BFN Unit 2, TVA submitted (Reference 4) a request to revise the TS low water level setpoint. On January 2, 1991, NRC-approved (Reference 5) the requested amendment, which stated:

> "The amendment changes the Technical Specifications (TS) to incorporate a revised trip setpoint for the Level 1 low reactor pressure vessel (RPV) water level based on new calculation methodology."

In addition, the accompanying Safety Evaluation stated:

> "TVA performed a Setpoint and Scaling Calculation to determine the accuracy of the instruments and loops. This accuracy was compared to the required accuracies to assure that there is sufficient margin between the setpoints and the operating limits, and the Safety Limits. The calculations reviewed by the staff at TVA's Rockville offices were as follows:

Instrument No.	Calculation No.	Revision No.
2-LT-3-56A	ED-Q2003-88122	3
2-LT-3-56B	ED-Q2003-88123	3
2-LT-3-56C	ED-Q2003-88124	3
2-LT-3-56D	ED-Q2003-88125	3
2-LT-3-58A	ED-Q2003-880126	4
2-LT-3-58B	ED-Q2003-880127	4
2-LT-3-58C	ED-Q2003-880128	4
2-LT-3-58D	ED-Q2003-880129	4

> The staff's review of the calculations verified that TVA addressed instrument and loop errors for normal operation and accident conditions. … The methodology for determination of instrument setpoints used by TVA was in accordance with Regulatory Guide (RG) 1.105 that endorses Instrument Society of America (ISA) Standard ISA-S67.04 - 1982 "Setpoint for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants". This standard provides guidance for ensuring that setpoints stay within TS limits. …
>
> The proposed changes to the LSSS *(limiting safety system setting)* and SL *(Safety Limit)* settings were deemed acceptable because they are based on a value derived by approved calculational means. This change ensures that trips occur within the analytical limit used to confirm the design bases of the plant."

This NRC approved setpoint methodology continues to be used and has formed the basis for subsequent NRC approval of TS changes. For example, the NRC approved (Reference 6) a change in the reactor vessel water level Safety Limit and limiting safety system setting for BFN Units 1 and 3 by Amendments 222 and 196, respectively. The Safety Evaluation states:

> "The methodology used by the licensee to determine the LSSS is in accordance with the Instrument Society of America Standard ISA-S67.04 - 1982 "Setpoints for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants." This methodology is consistent with the guidance of Regulatory Guide 1.105. Therefore, the proposed LSSS is acceptable."

In addition, as part of NRC's review of the Units 1, 2, and 3 TS for a 24-month fuel cycle (Reference 7), NRC endorsed TVA's method of evaluation of as-found and as-left values in TVA's maintenance program and TVA's method of addressing failures through the corrective action program.

4.2 Compliance with Current Regulations and Commitments

As discussed above, NRC has previously concluded that the methodology used by TVA to determine the LSSS is in accordance with the Instrument Society of America Standard ISA-S67.04 - 1982 "Setpoints for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants" and that this methodology is consistent with the guidance of Regulatory Guide 1.105.

TVA has reviewed its Licensing Basis and determined that no commitments are affected by this proposed change.

5.0 REGULATORY SAFETY ANALYSIS

The Tennessee Valley Authority (TVA) is submitting an amendment request to licenses DPR-33, DPR-52, and DPR-68 for Units 1, 2, and 3, respectively. The scope of this proposed Technical Specification (TS) change includes those drift susceptible instruments, which are either necessary to ensure compliance with a Safety Limit or critical in ensuring the fuel peak cladding temperature acceptance criterion of 10 CFR 50.46 are met. The proposed change adds a footnote that specifies the actions to be taken for the applicable as-found instrument setpoints and references a discussion of the NRC-approved TVA setpoint methodology. The purpose of including this information in the TS is to control critical instrument setpoints and ensure compliance with 10 CFR 50.36.

5.1 No Significant Hazards Consideration

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment", as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

Including references to TVA's methodology for determining, setting, and evaluating as-found instrument setpoints in the TS is an administrative change. There will be no change to the manner in which Safety Limits, Analytical Limits, or Allowable Values are determined. No changes are proposed in the manner in which the Reactor Protection System (RPS), Emergency Core Cooling System (ECCS), Reactor Core Isolation Cooling (RCIC), or Primary Containment Isolation systems provide plant protection or which create new modes of plant operation.

The proposed request will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   There are no hardware changes nor are there any changes in the method by which any plant system performs a safety function. This request does not affect the normal method of plant operation. The proposed amendment does not introduce new equipment, which could create a new or different kind of accident.

   No new external threats, release pathways, or equipment failure modes are created. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this request. Therefore, the implementation of the proposed amendment will not create a possibility for an accident of a new or different type than those previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No

Including references to TVA's methodology for determining, setting, and evaluating as-found instrument setpoints in the TS is an administrative change. No changes are proposed in the manner in which the RPS, ECCS, RCIC, or Primary Containment Isolation systems satisfy the Updated Final Safety Analysis Report requirements for accident mitigation or unit safe shutdown. There will be no change to Safety Limits, Analytical Limits, Allowable Values, or post-Loss Of Coolant Accident peak clad temperatures. For these reasons, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, TVA concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

As discussed above, NRC has previously concluded that the methodology used by TVA to determine the LSSS is in accordance with the Instrument Society of America Standard ISA-S67.04 - 1982 "Setpoints for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants" and that this methodology is consistent with the guidance of Regulatory Guide 1.105.

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

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(9). Therefore, pursuant to 10 CFR 51.22(b), no

environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. NRC letter to TVA, dated January 6, 2005, "Browns Ferry Units 1, 2, and 3 - Request for Information Regarding Status of Amendments Using Method 3 (TAC Nos. MC1330, MC1427, MC2305, MC3812, MC4070, MC4071, MC4072, MC4161, MC3743, and MC3744)."

2. NRC letter to TVA, dated April 19, 2005, "Browns Ferry Nuclear Plant, Unit 1 - Licensing Action Status and Interdependencies (TAC Nos. MC1330, MC1427, MC2305, MC3812, MC3813, MC3822, MC3960, MC4070, MC4161, MC4659, MC4797, MC5254, and MC5373)."

3. NRC letter to TVA, dated May 8, 1989, "Notice of Violation (NRC Inspection Report Nos. 50-259/89-06, 50-260/89-06 and 50-296/89-06)."

4. TVA letter to NRC, dated August 6, 1990, "Browns Ferry Nuclear Plant (BFN) - Unit 2 - TVA BFN Technical Specification (TS) No. 291 - Revision to Level 1 Low Reactor Pressure Vessel (RPV) Water Level."

5. NRC letter to TVA, dated January 2, 1991, "Issuance of Amendment (TAC No. 77279) (TS 291)."

6. NRC letter to TVA, dated July 17, 1995, "Issuance of Technical Specification Amendments for the Browns Ferry Nuclear Plant Units 1, 2, and 3 (TAC NOS. M89248, M89249 and M89250) (TS 318)."

7. NRC letter to TVA, dated November 30, 1998, "Issuance of Amendments - Browns Ferry Nuclear Plan Units 1, 2, and 3 (TAC Nos. MA2081, MA2082, and MA2083)."

Enclosure 2

Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3
Technical Specifications (TS) Change TS-453

Instrument Setpoint Program

Proposed Technical Specification Changes (mark-up)

The following footnote will be added where indicated on the TS pages:

During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Downscale	1	2	F	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	≥ 3% RTP
e. Inop	1,2	2	G	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	NA
3. Reactor Vessel Steam Dome Pressure - High[d]	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1055 psig
4. Reactor Vessel Water Level - Low, Level 3[d]	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 538 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5[a]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5[a]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(d) INSERT FOOTNOTE

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8. Turbine Stop Valve - Closure	≥ 30% RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 10% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low[d]	≥ 30% RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 550 psig
10. Reactor Mode Switch - Shutdown Position	1,2	1	G	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
	5[a]	1	H	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
11. Manual Scram	1,2	1	G	SR 3.3.1.1.8 SR 3.3.1.1.14	NA
	5[a]	1	H	SR 3.3.1.1.8 SR 3.3.1.1.14	NA
12. RPS Channel Test Switches	1,2	2	G	SR 3.3.1.1.4	NA
	5[a]	2	H	SR 3.3.1.1.4	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(d) INSERT FOOTNOTE

Table 3.3.5.1-1 (page 1 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level - Low Low Low, Level 1(e)	1,2,3, 4(a), 5(a)	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure High(e)	1,2,3	4(b)	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Steam Dome Pressure - Low (Injection Permissive and ECCS Initiation)(e)	1,2,3	4(b) 2 per trip system	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
	4(a), 5(a)	4 2 per trip system	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
d. Core Spray Pump Discharge Flow - Low (Bypass)	1,2,3, 4(a), 5(a)	2 1 per subsystem	E	SR 3.3.5.1.2 SR 3.3.5.1.5	≥ 1647 gpm and ≤ 2910 gpm
e. Core Spray Pump Start - Time Delay Relay					
Pumps A,B,C,D (with diesel power)	1,2,3, 4(a), 5(a)	4 1 per pump	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump A (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump B (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump C (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 12 seconds and ≤ 16 seconds

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Channels affect Common Accident Signal Logic. Refer to LCO 3.8.1, "AC Sources - Operating."

(e) INSERT FOOTNOTE

Table 3.3.5.1-1 (page 2 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System (continued)					
e. Core Spray Pump Start - Time Delay Relay (continued)					
Pump D (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 18 seconds and ≤ 24 seconds
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level - Low Low Low, Level 1(e)	1,2,3, 4(a), 5(a)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure - High(e)	1,2,3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Steam Dome Pressure - Low (Injection Permissive and ECCS Initiation)(e)	1,2,3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
	4(a), 5(a)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
d. Reactor Steam Dome Pressure - Low (Recirculation Discharge Valve Permissive)(e)	1(c),2(c), 3(c)	4	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 215 psig and ≤ 245 psig
e. Reactor Vessel Water Level - Level 0	1,2,3	2 1 per subsystem	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 312 5/16 inches above vessel zero

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Deleted.

(c) With associated recirculation pump discharge valve open.

(e) INSERT FOOTNOTE

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
f. Low Pressure Coolant Injection Pump Start - Time Delay Relay					
Pump A,B,C,D (with diesel power)	1,2,3, 4(a), 5(a)	4	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump A (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump B (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump C (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 12 seconds and ≤ 16 seconds
Pump D (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 18 seconds and ≤ 24 seconds
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level - Low Low, Level 2(e)	1, 2(d), 3(d)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 470 inches above vessel zero

(continued)

(a) When the associated subsystem(s) are required to be OPERABLE.

(d) With reactor steam dome pressure > 150 psig.

(e) INSERT FOOTNOTE

Table 3.3.5.1-1 (page 4 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System (continued)					
b. Drywell Pressure - High(e)	1, 2(d), 3(d)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Vessel Water Level - High, Level 8	1, 2(d), 3(d)	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 583 inches above vessel zero
d. Condensate Header Level - Low	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ Elev. 551 feet
e. Suppression Pool Water Level - High	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≤ 7 inches above instrument zero
f. High Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)	1, 2(d), 3(d)	1	E	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 671 gpm
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level - Low Low Low, Level 1(e)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure - High(e)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 115 seconds

(continued)

(d) With reactor steam dome pressure > 150 psig.

(e) INSERT FOOTNOTE

Table 3.3.5.1-1 (page 5 of 6)

Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. ADS Trip System A (continued)					
d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory)(e)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 544 inches above vessel zero
e. Core Spray Pump Discharge Pressure - High	1, 2(d), 3(d)	4	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 175 psig and ≤ 195 psig
f. Low Pressure Coolant Injection Pump Discharge Pressure - High	1, 2(d), 3(d)	8	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 90 psig and ≤ 110 psig
g. Automatic Depressurization System High Drywell Pressure Bypass Timer	1, 2(d), 3(d)	2	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 322 seconds
5. ADS Trip System B					
a. Reactor Vessel Water Level - Low Low Low, Level 1(e)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure - High(e)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 115 seconds
d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory)(e)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 544 inches above vessel zero

(continued)

(d) With reactor steam dome pressure > 150 psig.

(e) INSERT FOOTNOTE

Table 3.3.5.2-1 (page 1 of 1)
Reactor Core Isolation Cooling System Instrumentation

FUNCTION	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low, Level 2(a)	4	B	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.4	≥ 470 inches above vessel zero
2. Reactor Vessel Water Level - High, Level 8	2	C	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.4	≤ 583 inches above vessel zero

(a) INSERT FOOTNOTE

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 398 inches above vessel zero
b. Main Steam Line Pressure - Low(c)	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 825 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 140% rated steam flow
d. Main Steam Tunnel Temperature - High	1,2,3	8	D	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 200°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low, Level 3	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 538 inches above vessel zero
b. Drywell Pressure - High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 2.5 psig
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 90 psi
b. HPCI Steam Supply Line Pressure - Low	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 100 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig

(continued)

(c) INSERT FOOTNOTE

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3[b]	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	1	3[b]	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17	NA
3. Reactor Vessel Steam Dome Pressure - High[d]	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig
4. Reactor Vessel Water Level - Low, Level 3[d]	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 528 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5[a]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

(d) INSERT FOOTNOTE

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level - High (continued)					
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 46 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 46 gallons
8. Turbine Stop Valve - Closure	≥ 30% RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 10% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low(d)	≥ 30% RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 550 psig
10. Reactor Mode Switch - Shutdown Position	1,2	1	G	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
	5(a)	1	H	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
11. Manual Scram	1,2	1	G	SR 3.3.1.1.8 SR 3.3.1.1.14	NA
	5(a)	1	H	SR 3.3.1.1.8 SR 3.3.1.1.14	NA
12. RPS Channel Test Switches	1,2	2	G	SR 3.3.1.1.4	NA
	5(a)	2	H	SR 3.3.1.1.4	NA
13.Deleted					

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(d) INSERT FOOTNOTE

Table 3.3.5.1-1 (page 1 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level - Low Low Low, Level 1(e)	1,2,3, 4(a), 5(a)	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure - High(e)	1,2,3	4(b)	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Steam Dome Pressure - Low (Injection Permissive and ECCS Initiation)(e)	1,2,3	4(b) 2 per trip system	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
	4(a), 5(a)	4 2 per trip system	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
d. Core Spray Pump Discharge Flow - Low (Bypass)	1,2,3, 4(a), 5(a)	2 1 per subsystem	E	SR 3.3.5.1.2 SR 3.3.5.1.5	≥ 1647 gpm and ≤ 2910 gpm
e. Core Spray Pump Start - Time Delay Relay					
Pumps A,B,C,D (with diesel power)	1,2,3, 4(a), 5(a)	4 1 per pump	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump A (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump B (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump C (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 12 seconds and ≤ 16 seconds

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Channels affect Common Accident Signal Logic. Refer to LCO 3.8.1, "AC Sources - Operating."

(e) INSERT FOOTNOTE

Table 3.3.5.1-1 (page 2 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System (continued)					
e. Core Spray Pump Start - Time Delay Relay (continued)					
Pump D (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 18 seconds and ≤ 24 seconds
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level - Low Low Low, Level 1(e)	1,2,3, 4(a), 5(a)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure - High(e)	1,2,3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Steam Dome Pressure - Low (Injection Permissive and ECCS Initiation)(e)	1,2,3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
	4(a), 5(a)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
d. Reactor Steam Dome Pressure - Low (Recirculation Discharge Valve Permissive)(e)	1(c), 2(c), 3(c)	4	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 215 psig and ≤ 245 psig
e. Reactor Vessel Water Level - Level 0	1,2,3	2 1 per subsystem	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 312 5/16 inches above vessel zero

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Deleted.

(c) With associated recirculation pump discharge valve open.

(e) INSERT FOOTNOTE

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
f. Low Pressure Coolant Injection Pump Start - Time Delay Relay					
Pump A,B,C,D (with diesel power)	1,2,3, 4[a], 5[a]	4	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump A (with normal power)	1,2,3, 4[a], 5[a]	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump B (with normal power)	1,2,3, 4[a], 5[a]	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump C (with normal power)	1,2,3, 4[a], 5[a]	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 12 seconds and ≤ 16 seconds
Pump D (with normal power)	1,2,3, 4[a], 5[a]	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 18 seconds and ≤ 24 seconds
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level - Low Low, Level 2[e]	1, 2[d], 3[d]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 470 inches above vessel zero

(continued)

(a) When the associated subsystem(s) are required to be OPERABLE.

(d) With reactor steam dome pressure > 150 psig.

(e) INSERT FOOTNOTE

Table 3.3.5.1-1 (page 4 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System (continued)					
b. Drywell Pressure - High(e)	1, 2(d), 3(d)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Vessel Water Level - High, Level 8	1, 2(d), 3(d)	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 583 inches above vessel zero
d. Condensate Header Level - Low	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ Elev. 551 feet
e. Suppression Pool Water Level - High	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≤ 7 inches above instrument zero
f. High Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)	1, 2(d), 3(d)	1	E	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 671 gpm
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level - Low Low Low, Level 1(e)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure - High(e)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 115 seconds

(continued)

(d) With reactor steam dome pressure > 150 psig.

(e) INSERT FOOTNOTE

Table 3.3.5.1-1 (page 5 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. ADS Trip System A (continued)					
d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory)(e)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 528 inches above vessel zero
e. Core Spray Pump Discharge Pressure - High	1, 2(d), 3(d)	4	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 175 psig and ≤ 195 psig
f. Low Pressure Coolant Injection Pump Discharge Pressure - High	1, 2(d), 3(d)	8	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 90 psig and ≤ 110 psig
g. Automatic Depressurization System High Drywell Pressure Bypass Timer	1, 2(d), 3(d)	2	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 322 seconds
5. ADS Trip System B					
a. Reactor Vessel Water Level - Low Low Low, Level 1(e)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure - High(e)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 115 seconds
d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory)(e)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 528 inches above vessel zero

(continued)

(d) With reactor steam dome pressure > 150 psig.

(e) INSERT FOOTNOTE

Table 3.3.5.2-1 (page 1 of 1)
Reactor Core Isolation Cooling System Instrumentation

FUNCTION	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low, Level 2(a)	4	B	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.4	≥ 470 inches above vessel zero
2. Reactor Vessel Water Level - High, Level 8	2	C	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.4	≤ 583 inches above vessel zero

(a) INSERT FOOTNOTE

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 398 inches above vessel zero
b. Main Steam Line Pressure - Low(c)	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 825 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 140% rated steam flow
d. Main Steam Tunnel Temperature - High	1,2,3	8	D	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 200°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low, Level 3	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 528 inches above vessel zero
b. Drywell Pressure - High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 2.5 psig
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 90 psi
b. HPCI Steam Supply Line Pressure - Low	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 100 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig

(continued)

(c) INSERT FOOTNOTE

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3[b]	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	1	3[b]	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17	NA
3. Reactor Vessel Steam Dome Pressure - High[d]	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig
4. Reactor Vessel Water Level - Low, Level 3[d]	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 528 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5[a]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

(d) INSERT FOOTNOTE

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level - High					
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 46 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 46 gallons
8. Turbine Stop Valve - Closure	≥ 30% RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 10% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low(d)	≥ 30% RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 550 psig
10. Reactor Mode Switch - Shutdown Position	1,2	1	G	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
	5(a)	1	H	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
11. Manual Scram	1,2	1	G	SR 3.3.1.1.8 SR 3.3.1.1.14	NA
	5(a)	1	H	SR 3.3.1.1.8 SR 3.3.1.1.14	NA
12. RPS Channel Test Switches	1,2	2	G	SR 3.3.1.1.4	NA
	5(a)	2	H	SR 3.3.1.1.4	NA
13. Deleted					

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(d) INSERT FOOTNOTE

Table 3.3.5.1-1 (page 1 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level — Low Low Low, Level 1[f]	1,2,3, 4[a], 5[a]	4[b]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure — High[f]	1,2,3	4[b]	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Steam Dome Pressure — Low (Injection Permissive and ECCS Initiation)[f]	1,2,3	4[b] 2 per trip system	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
	4[a], 5[a]	4 2 per trip system	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
d. Core Spray Pump Discharge Flow — Low (Bypass)	1,2,3, 4[a], 5[a]	2 1 per subsystem	E	SR 3.3.5.1.2 SR 3.3.5.1.5	≥ 1647 gpm and ≤ 2910 gpm
e. Core Spray Pump Start — Time Delay Relay					
Pumps A,B,C,D (with diesel power)	1,2,3, 4[a], 5[a]	4 1 per pump	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump A (with normal power)	1,2,3, 4[a], 5[a]	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump B (with normal power)	1,2,3, 4[a], 5[a]	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump C (with normal power)	1,2,3, 4[a], 5[a]	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 12 seconds and ≤ 16 seconds

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Channels affect Common Accident Signal Logic. Refer to LCO 3.8.1, "AC Sources - Operating."

(f) INSERT FOOTNOTE

Table 3.3.5.1-1 (page 2 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System (continued)					
e. Core Spray Pump Start — Time Delay Relay (continued)					
Pump D (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 18 seconds and ≤ 24 seconds
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level — Low Low Low, Level 1(f)	1,2,3, 4(a), 5(a)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure — High(f)	1,2,3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Steam Dome Pressure — Low (Injection Permissive and ECCS Initiation)(f)	1,2,3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
	4(a), 5(a)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
d. Reactor Steam Dome Pressure — Low (Recirculation Discharge Valve Permissive)(f)	1(c),2(c), 3(c)	4	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 215 psig and ≤ 245 psig
e. Reactor Vessel Water Level — Level 0	1,2,3	2 1 per subsystem	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 312 5/16 inches above vessel zero

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Deleted.

(c) With associated recirculation pump discharge valve open.

(f) INSERT FOOTNOTE

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
f. Low Pressure Coolant Injection Pump Start — Time Delay Relay					
Pump A,B,C,D (with diesel power)	1,2,3, 4[a], 5[a]	8[e]	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump A (with normal power)	1,2,3, 4[a], 5[a]	2 1 per trip system	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump B (with normal power)	1,2,3, 4[a], 5[a]	2 1 per trip system	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump C (with normal power)	1,2,3, 4[a], 5[a]	2 1 per trip system	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 12 seconds and ≤ 16 seconds
Pump D (with normal power)	1,2,3, 4[a], 5[a]	2 1 per trip system	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 18 seconds and ≤ 24 seconds
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level — Low Low, Level 2[f]	1, 2[d], 3[d]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 470 inches above vessel zero

(continued)

(a) When the associated subsystem(s) are required to be OPERABLE.

(d) With reactor steam dome pressure > 150 psig.

(e) Pumps A, B, C, and D have 2 relays each (1 per trip system).

(f) INSERT FOOTNOTE

Table 3.3.5.1-1 (page 4 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System (continued)					
b. Drywell Pressure — High(f)	1, 2(d),3(d)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Vessel Water Level — High, Level 8	1, 2(d), 3(d)	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 583 inches above vessel zero
d. Condensate Header Level — Low	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ Elev. 551 feet
e. Suppression Pool Water Level — High	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≤ 7 inches above instrument zero
f. High Pressure Coolant Injection Pump Discharge Flow—Low (Bypass)	1, 2(d), 3(d)	1	E	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 671 gpm
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level — Low Low Low, Level 1(f)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure — High(f)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 115 seconds

(continued)

(d) With reactor steam dome pressure > 150 psig.

(f) INSERT FOOTNOTE

Table 3.3.5.1-1 (page 5 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. ADS Trip System A (continued)					
d. Reactor Vessel Water Level — Low, Level 3 (Confirmatory)(f)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 528 inches above vessel zero
e. Core Spray Pump Discharge Pressure — High	1, 2(d), 3(d)	4	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 175 psig and ≤ 195 psig
f. Low Pressure Coolant Injection Pump Discharge Pressure — High	1, 2(d), 3(d)	8	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 90 psig and ≤ 110 psig
g. Automatic Depressurization System High Drywell Pressure Bypass Timer	1, 2(d), 3(d)	2	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 322 seconds
5. ADS Trip System B					
a. Reactor Vessel Water Level — Low Low Low, Level 1(f)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure — High(f)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 115 seconds
d. Reactor Vessel Water Level — Low, Level 3 (Confirmatory)(f)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 528 inches above vessel zero

(continued)

(d) With reactor steam dome pressure > 150 psig.

(f) INSERT FOOTNOTE

Table 3.3.5.2-1 (page 1 of 1)
Reactor Core Isolation Cooling System Instrumentation

FUNCTION	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low, Level 2[a]	4	B	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.4	≥ 470 inches above vessel zero
2. Reactor Vessel Water Level - High, Level 8	2	C	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.4	≤ 583 inches above vessel zero

(a) INSERT FOOTNOTE

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 398 inches above vessel zero
b. Main Steam Line Pressure - Low(c)	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 825 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 140% rated steam flow
d. Main Steam Tunnel Temperature - High	1,2,3	8	D	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 200°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low, Level 3	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 528 inches above vessel zero
b. Drywell Pressure - High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 2.5 psig
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 90 psi
b. HPCI Steam Supply Line Pressure - Low	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 100 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig

(continued)

(c) INSERT FOOTNOTE

Enclosure 3

Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3
Technical Specifications (TS) Change TS-453

Instrument Setpoint Program

Changes to Technical Specification Bases Pages
(mark-up)

INSERT A

and contained in design output documents. The setpoint calculations are in accordance with the setpoint methodology described in Chapter 7 of the Updated Final Safety Analysis Report.

INSERT B

The Acceptable As Found band is based on a statistical combination of possible measurable uncertainties (i.e., setting tolerance, drift, temperature effects, and measurement and test equipment [M&TE]). When a channel's as-found value is conservative to the Allowable Value but outside the Acceptable As Found band (tolerance range), the channel may be degraded. An initial determination shall be made to validate the OPERABILITY. This initial determination will verify that the instrument will continue to behave in accordance with design-basis assumptions. The purpose of the initial determination is to ensure confidence in the instrument performance prior to returning the instrument to service. The technician performing the surveillance will evaluate the instrument's ability to maintain a stable setpoint within the as-left tolerance. The technician's evaluation will be reviewed by on shift personnel during the approval of the surveillance data prior to returning the channel back to service at the completion of the surveillance. This shall constitute the initial determination of operability.

After the surveillance is completed, the channel's as-found condition will be documented in the Corrective Action Program. As part of the activities of the Corrective Action Program, additional evaluations and potential corrective actions will be performed as necessary to ensure that any as-found setting which is conservative to the Allowable Value but outside the Acceptable As Found band is evaluated for long-term reliability trends.

When a channel is found to exceed the channel's Allowable Value or fails to be reset to with the Acceptable As Left band, the channel shall be declared inoperable.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

RPS instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 10). Functions not specifically credited in the accident analysis are retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The OPERABILITY of the RPS is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.1.1-1. Each Function must have a required number of OPERABLE channels per RPS trip system, with their setpoints within the specified Allowable Value, where appropriate. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint).

INSERT A

Allowable Values are specified for each RPS Function specified in the Table. Nominal trip setpoints are specified in the setpoint ~~calculations.~~ The nominal setpoints are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Reactor Vessel Steam Dome Pressure - High (PIS-3-22AA, PIS-3-22BB, PIS-3-22C, and PIS-3-22D)

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. The Reactor Vessel Steam Dome Pressure - High Function initiates a scram for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 4, reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux - High signal, not the Reactor Vessel Steam Dome Pressure - High signal), along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

High reactor pressure signals are initiated from four pressure transmitters that sense reactor pressure. The Reactor Vessel Steam Dome Pressure - High Allowable Value is chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

Four channels of Reactor Vessel Steam Dome Pressure - High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required to be OPERABLE in MODES 1 and 2 when the RCS is pressurized and the potential for pressure increase exists.

INSERT B

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. Reactor Vessel Water Level - Low, Level 3 (LIS-3-203A, LIS-3-203B, LIS-3-203C, and LIS-3-203D) (continued)

The Function is required in MODES 1 and 2 where considerable energy exists in the RCS resulting in the limiting transients and accidents. ECCS initiations at Reactor Vessel Water Level - Low Low, Level 2 and Low Low Low, Level 1 provide sufficient protection for level transients in all other MODES.

INSERT B

5. Main Steam Isolation Valve - Closure

MSIV closure results in loss of the main turbine and the condenser as a heat sink for the nuclear steam supply system and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on a Main Steam Isolation Valve - Closure signal before the MSIVs are completely closed in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. However, for the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux - High Function, along with the S/RVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is not assumed in the overpressurization analysis. Additionally, MSIV closure is assumed in the transients analyzed in Reference 7 (e.g., low steam line pressure, manual closure of MSIVs, high steam line flow).

The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (PS-47-142, PS-47-144, PS-47-146, and PS-47-148) (continued)

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP, since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

INSERT B

10. Reactor Mode Switch - Shutdown Position

The Reactor Mode Switch - Shutdown Position Function provides signals, via the manual scram logic channels, directly to the scram pilot solenoid power circuits. These manual scram logic channels are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with four channels, each of which provides input into one of the RPS logic channels.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a, 2.a. Reactor Vessel Water Level - Low Low Low, Level 1 (LS-3-58A-D) (continued)

The Reactor Vessel Water Level - Low Low Low, Level 1 Allowable Value is chosen to allow time for the low pressure injection/spray subsystems to activate and provide adequate cooling.

Four channels of Reactor Vessel Water Level - Low Low Low, Level 1 Function are only required to be OPERABLE when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Refer to LCO 3.5.1 and LCO 3.5.2, "ECCS - Shutdown," for Applicability Bases for the low pressure ECCS subsystems.

INSERT B

1.b, 2.b. Drywell Pressure - High (PIS-64-58A-D)

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS is initiated upon receipt of the Drywell Pressure - High Function in order to minimize the possibility of fuel damage. The Drywell Pressure - High is also utilized in the development of the Common Accident Signal which initiates the DGs and EECW System. (Refer to LCO 3.8.1, "AC Sources - Operating" for operability requirements of the Common Accident Signal Logic). The Drywell Pressure - High Function, along with the Reactor Steam Dome Pressure - Low Function, are directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b, 2.b. Drywell Pressure - High (PIS-64-58A-D) (continued)

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

The Drywell Pressure - High Function is required to be OPERABLE when ECCS is required to be OPERABLE in conjunction with times when the primary containment is required to be OPERABLE. Thus, four channels of the CS and LPCI Drywell Pressure - High Function are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude ECCS initiation. In MODES 4 and 5, the Drywell Pressure - High Function is not required, since there is insufficient energy in the reactor to pressurize the primary containment to Drywell Pressure - High setpoint. Refer to LCO 3.5.1 for Applicability Bases for the low pressure ECCS subsystems.

INSERT B

1.c, 2.c. Reactor Steam Dome Pressure - Low (Injection Permissive and ECCS Initiation)
(PIS-3-74A and B; PIS-68-95 and 96)

Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. This ensures that, prior to opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure. The Reactor Steam Dome Pressure - Low is also utilized in the development of the Common Accident Signal which initiates the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.c, 2.c. Reactor Steam Dome Pressure - Low (Injection Permissive and ECCS Initiation)
(PIS-3-74A and B; PIS-68-95 and 96) (continued)

DGs and EECW System. (Refer to LCO 3.8.1, "AC Sources - Operating," for operability requirements of the Common Accident Signal Logic). The Reactor Steam Dome Pressure - Low is one of the Functions assumed to be OPERABLE and capable of permitting initiation of the ECCS during the transients analyzed in References 1 and 3. In addition, the Reactor Steam Dome Pressure - Low Function is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

The Reactor Steam Dome Pressure - Low signals are initiated from four pressure transmitters that sense the reactor dome pressure.

The Allowable Value is low enough to prevent overpressurizing the equipment in the low pressure ECCS, but high enough to ensure that the ECCS injection prevents the fuel peak cladding temperature from exceeding the limits of 10 CFR 50.46.

INSERT B

Four channels of Reactor Steam Dome Pressure - Low Function are only required to be OPERABLE when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.d. Reactor Steam Dome Pressure - Low (Recirculation Discharge Valve Permissive)
(PS-3-74A and B; PS-68-95 and 96) (continued)

The Reactor Steam Dome Pressure - Low signals are initiated from four pressure transmitters that sense the reactor dome pressure.

The Allowable Value is chosen to ensure that the valves close prior to commencement of LPCI injection flow into the core, as assumed in the safety analysis.

Four channels of the Reactor Steam Dome Pressure - Low Function are only required to be OPERABLE in MODES 1, 2, and 3 with the associated recirculation pump discharge valve open. With the valve(s) closed, the function of the instrumentation has been performed; thus, the Function is not required. In MODES 4 and 5, the loop injection location is not critical since LPCI injection through the recirculation loop in either direction will still ensure that LPCI flow reaches the core (i.e., there is no significant reactor steam dome back pressure).

INSERT B

2.e. Reactor Vessel Water Level - Level 0 (LIS-3-52 and 62A)

The Level 0 Function is provided as a permissive to allow the RHR System to be manually aligned from the LPCI mode to the suppression pool cooling/spray or drywell spray modes. The permissive ensures that water in the vessel is approximately two thirds core height before the manual transfer is allowed. This ensures that LPCI is available to prevent or minimize fuel damage. This function may be overridden during accident conditions as allowed by plant procedures. Reactor Vessel Water Level - Level 0 Function is implicitly assumed in the analysis of the recirculation line break (Ref. 2) since the analysis assumes that no LPCI flow diversion occurs when

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	<u>3.a. Reactor Vessel Water Level - Low Low, Level 2</u> (LIS-3-58A-D) (continued)

one of the Functions assumed to be OPERABLE and capable of initiating HPCI during the transients analyzed in References 1, 2, and 3. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level - Low Low, Level 2 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level - Low Low, Level 2 Allowable Value is high enough such that for complete loss of feedwater flow, the Reactor Core Isolation Cooling (RCIC) System flow with HPCI assumed to fail will be sufficient to avoid initiation of low pressure ECCS at Reactor Vessel Water Level - Low Low Low, Level 1.

Four channels of Reactor Vessel Water Level - Low Low, Level 2 Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for HPCI Applicability Bases.

INSERT B

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3.b. Drywell Pressure - High (PIS-64-58A-D)

High pressure in the drywell could indicate a break in the RCPB. The HPCI System is initiated upon receipt of the Drywell Pressure - High Function in order to minimize the possibility of fuel damage. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible to be indicative of a LOCA inside primary containment.

Four channels of the Drywell Pressure - High Function are required to be OPERABLE when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for HPCI Applicability Bases.

INSERT B

3.c. Reactor Vessel Water Level - High, Level 8 (LIS-3-208B and 208D)

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Level 8 signal is used to trip the HPCI turbine to prevent overflow into the main steam lines (MSLs). The Reactor Vessel Water Level - High, Level 8 Function is not assumed in the accident and transient analyses. It was retained since it is a potentially significant contributor to risk, thus it meets Criterion 4 of the NRC Policy Statement (Ref. 5).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.a, 5.a. Reactor Vessel Water Level - Low Low Low, Level 1 (LS-3-58A-D) (continued)

Reactor Vessel Water Level - Low Low Low, Level 1 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Low Low, Level 1 Function are required to be OPERABLE only when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip system A, while the other two channels input to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

The Reactor Vessel Water Level - Low Low Low, Level 1 Allowable Value is chosen to allow time for the low pressure core flooding systems to initiate and provide adequate cooling.

INSERT B

4.b, 5.b. Drywell Pressure - High (PIS-64-57A-D)

High pressure in the drywell could indicate a break in the RCPB. Therefore, ADS receives one of the signals necessary for initiation from this Function in order to minimize the possibility of fuel damage. The Drywell Pressure - High is assumed to be OPERABLE and capable of initiating the ADS during the accidents analyzed in Reference 2. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Drywell Pressure - High signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.b, 5.b. Drywell Pressure - High (PIS-64-57A-D) (continued)

Four channels of Drywell Pressure - High Function are only required to be OPERABLE when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip system A, while the other two channels input to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

INSERT B

4.c, 5.c. Automatic Depressurization System Initiation Timer

The purpose of the Automatic Depressurization System Initiation Timer is to delay depressurization of the reactor vessel to allow the HPCI System time to maintain reactor vessel water level. Since the rapid depressurization caused by ADS operation is one of the most severe transients on the reactor vessel, its occurrence should be limited. By delaying initiation of the ADS Function, the operator is given the chance to monitor the success or failure of the HPCI System to maintain water level, and then to decide whether or not to allow ADS to initiate, to delay initiation further by recycling the timer, or to inhibit initiation permanently. The Automatic Depressurization System Initiation Timer Function is assumed to be OPERABLE for the accident analyses of Reference 2 that require ECCS initiation.

There are two Automatic Depressurization System Initiation Timer relays, one in each of the two ADS trip systems. The Allowable Value for the Automatic Depressurization System Initiation Timer is chosen so that there is still time after depressurization for the low pressure ECCS subsystems to provide adequate core cooling.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.c, 5.c. Automatic Depressurization System Initiation Timer (continued)

Two channels of the Automatic Depressurization System Initiation Timer Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.d, 5.d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory) (LIS-3-184 and 185)

The Reactor Vessel Water Level - Low, Level 3 (Confirmatory) Function is used by the ADS only as a confirmatory low water level signal. ADS receives one of the signals necessary for initiation from Reactor Vessel Water Level - Low Low Low, Level 1 signals. In order to prevent spurious initiation of the ADS due to spurious Level 1 signals, a Level 3 (Confirmatory) signal must also be received before ADS initiation commences.

Reactor Vessel Water Level - Low, Level 3 (Confirmatory) signals are initiated from two level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

Two channels of Reactor Vessel Water Level - Low, Level 3 (Confirmatory) Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

INSERT B

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Reactor Vessel Water Level - Low Low, Level 2 (LIS-3-58A-D) (continued)

Reactor Vessel Water Level - Low Low, Level 2 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level - Low Low, Level 2 Allowable Value is set high enough such that for complete loss of feedwater flow, the RCIC System flow with high pressure coolant injection assumed to fail will be sufficient to avoid initiation of low pressure ECCS at Level 1.

Four channels of Reactor Vessel Water Level - Low Low, Level 2 Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC initiation. Refer to LCO 3.5.3 for RCIC Applicability Bases.

INSERT B

2. Reactor Vessel Water Level - High, Level 8 (LIS-3-208A and 208C)

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Level 8 signal is used to close the RCIC steam supply valve to prevent overflow into the main steam lines (MSLs).

Reactor Vessel Water Level - High, Level 8 signals for RCIC are initiated from two level transmitters from the narrow range water level measurement instrumentation, which sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low (PIS-1-72, 76, 82, 86)

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves excluding the Recirculation Loop Sample valves.

INSERT B

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

RPS instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 10). Functions not specifically credited in the accident analysis are retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The OPERABILITY of the RPS is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.1.1-1. Each Function must have a required number of OPERABLE channels per RPS trip system, with their setpoints within the specified Allowable Value, where appropriate. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint).

INSERT A

Allowable Values are specified for each RPS Function specified in the Table. Nominal trip setpoints are specified in the setpoint ~~calculations~~. The nominal setpoints are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Reactor Vessel Steam Dome Pressure - High (PIS-3-22AA, PIS-3-22BB, PIS-3-22C and PIS-3-22D)

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. The Reactor Vessel Steam Dome Pressure - High Function initiates a scram for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 4, reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux - High signal, not the Reactor Vessel Steam Dome Pressure - High signal), along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

High reactor pressure signals are initiated from four pressure transmitters that sense reactor pressure. The Reactor Vessel Steam Dome Pressure - High Allowable Value is chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

Four channels of Reactor Vessel Steam Dome Pressure - High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required to be OPERABLE in MODES 1 and 2 when the RCS is pressurized and the potential for pressure increase exists.

INSERT B

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. Reactor Vessel Water Level - Low, Level 3 (LIS-3-203A, LIS-3-203B, LIS-3-203C, and LIS-3-203D) (continued)

The Function is required in MODES 1 and 2 where considerable energy exists in the RCS resulting in the limiting transients and accidents. ECCS initiations at Reactor Vessel Water Level - Low Low, Level 2 and Low Low Low, Level 1 provide sufficient protection for level transients in all other MODES.

INSERT B

5. Main Steam Isolation Valve - Closure

MSIV closure results in loss of the main turbine and the condenser as a heat sink for the nuclear steam supply system and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on a Main Steam Isolation Valve - Closure signal before the MSIVs are completely closed in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. However, for the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux - High Function, along with the S/RVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is not assumed in the overpressurization analysis. Additionally, MSIV closure is assumed in the transients analyzed in Reference 7 (e.g., low steam line pressure, manual closure of MSIVs, high steam line flow).

The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (PS-47-142, PS-47-144, PS-47-146, and PS-47-148) (continued)

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP, since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

INSERT B

10. Reactor Mode Switch - Shutdown Position

The Reactor Mode Switch - Shutdown Position Function provides signals, via the manual scram logic channels, directly to the scram pilot solenoid power circuits. These manual scram logic channels are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with four channels, each of which provides input into one of the RPS logic channels.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a, 2.a. Reactor Vessel Water Level - Low Low Low, Level 1 (LS-3-58A-D) (continued)

The Reactor Vessel Water Level - Low Low Low, Level 1 Allowable Value is chosen to allow time for the low pressure injection/spray subsystems to activate and provide adequate cooling.

Four channels of Reactor Vessel Water Level - Low Low Low, Level 1 Function are only required to be OPERABLE when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Refer to LCO 3.5.1 and LCO 3.5.2, "ECCS - Shutdown," for Applicability Bases for the low pressure ECCS subsystems.

INSERT B

1.b, 2.b. Drywell Pressure - High (PIS-64-58A-D)

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS is initiated upon receipt of the Drywell Pressure - High Function in order to minimize the possibility of fuel damage. The Drywell Pressure - High is also utilized in the development of the Common Accident Signal which initiates the DGs and EECW System. (Refer to LCO 3.8.1, "AC Sources - Operating" for operability requirements of the Common Accident Signal Logic). The Drywell Pressure - High Function, along with the Reactor Steam Dome Pressure - Low Function, are directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b, 2.b. Drywell Pressure - High (PIS-64-58A-D) (continued)

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

The Drywell Pressure - High Function is required to be OPERABLE when ECCS is required to be OPERABLE in conjunction with times when the primary containment is required to be OPERABLE. Thus, four channels of the CS and LPCI Drywell Pressure - High Function are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude ECCS initiation. In MODES 4 and 5, the Drywell Pressure - High Function is not required, since there is insufficient energy in the reactor to pressurize the primary containment to Drywell Pressure - High setpoint. Refer to LCO 3.5.1 for Applicability Bases for the low pressure ECCS subsystems.

INSERT B

1.c, 2.c. Reactor Steam Dome Pressure - Low (Injection Permissive and ECCS Initiation)
(PIS-3-74A and B; PIS-68-95 and 96)

Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. This ensures that, prior to opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure. The Reactor Steam Dome Pressure - Low is also utilized in the development of the Common Accident Signal which initiates the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.c, 2.c. Reactor Steam Dome Pressure - Low (Injection Permissive and ECCS Initiation) (PIS-3-74A and B; PIS-68-95 and 96) (continued)

DGs and EECW System. (Refer to LCO 3.8.1, "AC Sources - Operating," for operability requirements of the Common Accident Signal Logic). The Reactor Steam Dome Pressure - Low is one of the Functions assumed to be OPERABLE and capable of permitting initiation of the ECCS during the transients analyzed in References 1 and 3. In addition, the Reactor Steam Dome Pressure - Low Function is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

The Reactor Steam Dome Pressure - Low signals are initiated from four pressure transmitters that sense the reactor dome pressure.

The Allowable Value is low enough to prevent overpressurizing the equipment in the low pressure ECCS, but high enough to ensure that the ECCS injection prevents the fuel peak cladding temperature from exceeding the limits of 10 CFR 50.46.

INSERT B

Four channels of Reactor Steam Dome Pressure - Low Function are only required to be OPERABLE when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.d. Reactor Steam Dome Pressure - Low (Recirculation Discharge Valve Permissive)
(PS-3-74A and B; PS-68-95 and 96) (continued)

The Reactor Steam Dome Pressure - Low signals are initiated from four pressure transmitters that sense the reactor dome pressure.

The Allowable Value is chosen to ensure that the valves close prior to commencement of LPCI injection flow into the core, as assumed in the safety analysis.

Four channels of the Reactor Steam Dome Pressure - Low Function are only required to be OPERABLE in MODES 1, 2, and 3 with the associated recirculation pump discharge valve open. With the valve(s) closed, the function of the instrumentation has been performed; thus, the Function is not required. In MODES 4 and 5, the loop injection location is not critical since LPCI injection through the recirculation loop in either direction will still ensure that LPCI flow reaches the core (i.e., there is no significant reactor steam dome back pressure).

INSERT B

2.e. Reactor Vessel Water Level - Level 0 (LIS-3-52 and 62A)

The Level 0 Function is provided as a permissive to allow the RHR System to be manually aligned from the LPCI mode to the suppression pool cooling/spray or drywell spray modes. The permissive ensures that water in the vessel is approximately two thirds core height before the manual transfer is allowed. This ensures that LPCI is available to prevent or minimize fuel damage. This function may be overridden during accident conditions as allowed by plant procedures. Reactor Vessel Water Level - Level 0 Function is implicitly assumed in the analysis of the recirculation line break (Ref. 2) since the analysis assumes that no LPCI flow diversion occurs when

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.a. Reactor Vessel Water Level - Low Low, Level 2 (LIS-3-58A-D) (continued)

one of the Functions assumed to be OPERABLE and capable of initiating HPCI during the transients analyzed in References 1, 2, and 3. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level - Low Low, Level 2 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level - Low Low, Level 2 Allowable Value is high enough such that for complete loss of feedwater flow, the Reactor Core Isolation Cooling (RCIC) System flow with HPCI assumed to fail will be sufficient to avoid initiation of low pressure ECCS at Reactor Vessel Water Level - Low Low Low, Level 1.

Four channels of Reactor Vessel Water Level - Low Low, Level 2 Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for HPCI Applicability Bases.

INSERT B

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3.b. Drywell Pressure - High (PIS-64-58A-D)

High pressure in the drywell could indicate a break in the RCPB. The HPCI System is initiated upon receipt of the Drywell Pressure - High Function in order to minimize the possibility of fuel damage. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible to be indicative of a LOCA inside primary containment.

Four channels of the Drywell Pressure - High Function are required to be OPERABLE when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for HPCI Applicability Bases.

INSERT B →

3.c. Reactor Vessel Water Level - High, Level 8 (LIS-3-208B and 208D)

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Level 8 signal is used to trip the HPCI turbine to prevent overflow into the main steam lines (MSLs). The Reactor Vessel Water Level - High, Level 8 Function is not assumed in the accident and transient analyses. It was retained since it is a potentially significant contributor to risk, thus it meets Criterion 4 of the NRC Policy Statement (Ref. 5).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.a, 5.a. Reactor Vessel Water Level - Low Low Low, Level 1 (LS-3-58A-D) (continued)

Reactor Vessel Water Level - Low Low Low, Level 1 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Low Low, Level 1 Function are required to be OPERABLE only when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip system A, while the other two channels input to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

The Reactor Vessel Water Level - Low Low Low, Level 1 Allowable Value is chosen to allow time for the low pressure core flooding systems to initiate and provide adequate cooling.

INSERT B

4.b, 5.b. Drywell Pressure - High (PIS-64-57A-D)

High pressure in the drywell could indicate a break in the RCPB. Therefore, ADS receives one of the signals necessary for initiation from this Function in order to minimize the possibility of fuel damage. The Drywell Pressure - High is assumed to be OPERABLE and capable of initiating the ADS during the accidents analyzed in Reference 2. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Drywell Pressure - High signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.b, 5.b. Drywell Pressure - High (PIS-64-57A-D) (continued)

Four channels of Drywell Pressure - High Function are only required to be OPERABLE when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip system A, while the other two channels input to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

INSERT B

4.c, 5.c. Automatic Depressurization System Initiation Timer

The purpose of the Automatic Depressurization System Initiation Timer is to delay depressurization of the reactor vessel to allow the HPCI System time to maintain reactor vessel water level. Since the rapid depressurization caused by ADS operation is one of the most severe transients on the reactor vessel, its occurrence should be limited. By delaying initiation of the ADS Function, the operator is given the chance to monitor the success or failure of the HPCI System to maintain water level, and then to decide whether or not to allow ADS to initiate, to delay initiation further by recycling the timer, or to inhibit initiation permanently. The Automatic Depressurization System Initiation Timer Function is assumed to be OPERABLE for the accident analyses of Reference 2 that require ECCS initiation.

There are two Automatic Depressurization System Initiation Timer relays, one in each of the two ADS trip systems. The Allowable Value for the Automatic Depressurization System Initiation Timer is chosen so that there is still time after depressurization for the low pressure ECCS subsystems to provide adequate core cooling.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.c, 5.c. Automatic Depressurization System Initiation Timer (continued)

Two channels of the Automatic Depressurization System Initiation Timer Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.d, 5.d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory) (LIS-3-184 and 185)

The Reactor Vessel Water Level - Low, Level 3 (Confirmatory) Function is used by the ADS only as a confirmatory low water level signal. ADS receives one of the signals necessary for initiation from Reactor Vessel Water Level - Low Low Low, Level 1 signals. In order to prevent spurious initiation of the ADS due to spurious Level 1 signals, a Level 3 (Confirmatory) signal must also be received before ADS initiation commences.

Reactor Vessel Water Level - Low, Level 3 (Confirmatory) signals are initiated from two level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

Two channels of Reactor Vessel Water Level - Low, Level 3 (Confirmatory) Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

INSERT B

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Reactor Vessel Water Level - Low Low, Level 2 (LIS-3-58A-D) (continued)

Reactor Vessel Water Level - Low Low, Level 2 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level - Low Low, Level 2 Allowable Value is set high enough such that for complete loss of feedwater flow, the RCIC System flow with high pressure coolant injection assumed to fail will be sufficient to avoid initiation of low pressure ECCS at Level 1.

Four channels of Reactor Vessel Water Level - Low Low, Level 2 Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC initiation. Refer to LCO 3.5.3 for RCIC Applicability Bases.

INSERT B

2. Reactor Vessel Water Level - High, Level 8 (LIS-3-208A and 208C)

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Level 8 signal is used to close the RCIC steam supply valve to prevent overflow into the main steam lines (MSLs).

Reactor Vessel Water Level - High, Level 8 signals for RCIC are initiated from two level transmitters from the narrow range water level measurement instrumentation, which sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low (PIS-1-72, 76, 82, 86)

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves excluding the Recirculation Loop Sample valves.

INSERT B

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

RPS instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 10). Functions not specifically credited in the accident analysis are retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The OPERABILITY of the RPS is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.1.1-1. Each Function must have a required number of OPERABLE channels per RPS trip system, with their setpoints within the specified Allowable Value, where appropriate. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint).

INSERT A

Allowable Values are specified for each RPS Function specified in the Table. Nominal trip setpoints are specified in the setpoint ~~calculations.~~ The nominal setpoints are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Reactor Vessel Steam Dome Pressure - High
(PIS-3-22AA, PIS-3-22BB, PIS-3-22C and PIS-3-22D)

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. The Reactor Vessel Steam Dome Pressure - High Function initiates a scram for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 4, reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux - High signal, not the Reactor Vessel Steam Dome Pressure - High signal), along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

High reactor pressure signals are initiated from four pressure transmitters that sense reactor pressure. The Reactor Vessel Steam Dome Pressure - High Allowable Value is chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

Four channels of Reactor Vessel Steam Dome Pressure - High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required to be OPERABLE in MODES 1 and 2 when the RCS is pressurized and the potential for pressure increase exists.

INSERT B

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. Reactor Vessel Water Level - Low, Level 3 (LIS-3-203A, LIS-3-203B, LIS-3-203C, and LIS-3-203D) (continued)

The Function is required in MODES 1 and 2 where considerable energy exists in the RCS resulting in the limiting transients and accidents. ECCS initiations at Reactor Vessel Water Level - Low Low, Level 2 and Low Low Low, Level 1 provide sufficient protection for level transients in all other MODES.

INSERT B

5. Main Steam Isolation Valve - Closure

MSIV closure results in loss of the main turbine and the condenser as a heat sink for the nuclear steam supply system and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on a Main Steam Isolation Valve - Closure signal before the MSIVs are completely closed in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. However, for the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux - High Function, along with the S/RVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is not assumed in the overpressurization analysis. Additionally, MSIV closure is assumed in the transients analyzed in Reference 7 (e.g., low steam line pressure, manual closure of MSIVs, high steam line flow).

The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (PS-47-142, PS-47-144, PS-47-146, and PS-47-148) (continued)

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP, since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

INSERT B

10. Reactor Mode Switch - Shutdown Position

The Reactor Mode Switch - Shutdown Position Function provides signals, via the manual scram logic channels, directly to the scram pilot solenoid power circuits. These manual scram logic channels are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with four channels, each of which provides input into one of the RPS logic channels.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a, 2.a. Reactor Vessel Water Level - Low Low Low, Level 1 (LS-3-58A-D) (continued)

The Reactor Vessel Water Level - Low Low Low, Level 1 Allowable Value is chosen to allow time for the low pressure injection/spray subsystems to activate and provide adequate cooling.

Four channels of Reactor Vessel Water Level - Low Low Low, Level 1 Function are only required to be OPERABLE when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Refer to LCO 3.5.1 and LCO 3.5.2, "ECCS - Shutdown," for Applicability Bases for the low pressure ECCS subsystems.

INSERT B

1.b, 2.b. Drywell Pressure - High (PIS-64-58A-D)

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS is initiated upon receipt of the Drywell Pressure - High Function in order to minimize the possibility of fuel damage. The Drywell Pressure - High is also utilized in the development of the Common Accident Signal which initiates the DGs and EECW System. (Refer to LCO 3.8.1, "AC Sources - Operating" for operability requirements of the Common Accident Signal Logic). The Drywell Pressure - High Function, along with the Reactor Steam Dome Pressure - Low Function, are directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b, 2.b. Drywell Pressure - High (PIS-64-58A-D) (continued)

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

The Drywell Pressure - High Function is required to be OPERABLE when ECCS is required to be OPERABLE in conjunction with times when the primary containment is required to be OPERABLE. Thus, four channels of the CS and LPCI Drywell Pressure - High Function are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude ECCS initiation. In MODES 4 and 5, the Drywell Pressure - High Function is not required, since there is insufficient energy in the reactor to pressurize the primary containment to Drywell Pressure - High setpoint. Refer to LCO 3.5.1 for Applicability Bases for the low pressure ECCS subsystems.

INSERT B

1.c, 2.c. Reactor Steam Dome Pressure - Low (Injection Permissive and ECCS Initiation)
(PIS-3-74A and B; PIS-68-95 and 96)

Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. This ensures that, prior to opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure. The Reactor Steam Dome Pressure - Low is also utilized in the development of the Common Accident Signal which initiates the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.c, 2.c. Reactor Steam Dome Pressure - Low (Injection Permissive and ECCS Initiation)
(PIS-3-74A and B; PIS-68-95 and 96) (continued)

DGs and EECW System. (Refer to LCO 3.8.1, "AC Sources - Operating," for operability requirements of the Common Accident Signal Logic). The Reactor Steam Dome Pressure - Low is one of the Functions assumed to be OPERABLE and capable of permitting initiation of the ECCS during the transients analyzed in References 1 and 3. In addition, the Reactor Steam Dome Pressure - Low Function is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

The Reactor Steam Dome Pressure - Low signals are initiated from four pressure transmitters that sense the reactor dome pressure.

The Allowable Value is low enough to prevent overpressurizing the equipment in the low pressure ECCS, but high enough to ensure that the ECCS injection prevents the fuel peak cladding temperature from exceeding the limits of 10 CFR 50.46.

INSERT B

Four channels of Reactor Steam Dome Pressure - Low Function are only required to be OPERABLE when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.d. Reactor Steam Dome Pressure - Low (Recirculation Discharge Valve Permissive)
(PS-3-74A and B; PS-68-95 and 96) (continued)

The Reactor Steam Dome Pressure - Low signals are initiated from four pressure transmitters that sense the reactor dome pressure.

The Allowable Value is chosen to ensure that the valves close prior to commencement of LPCI injection flow into the core, as assumed in the safety analysis.

Four channels of the Reactor Steam Dome Pressure - Low Function are only required to be OPERABLE in MODES 1, 2, and 3 with the associated recirculation pump discharge valve open. With the valve(s) closed, the function of the instrumentation has been performed; thus, the Function is not required. In MODES 4 and 5, the loop injection location is not critical since LPCI injection through the recirculation loop in either direction will still ensure that LPCI flow reaches the core (i.e., there is no significant reactor steam dome back pressure).

INSERT B

2.e. Reactor Vessel Water Level - Level 0 (LIS-3-52 and 62A)

The Level 0 Function is provided as a permissive to allow the RHR System to be manually aligned from the LPCI mode to the suppression pool cooling/spray or drywell spray modes. The permissive ensures that water in the vessel is approximately two thirds core height before the manual transfer is allowed. This ensures that LPCI is available to prevent or minimize fuel damage. This function may be overridden during accident conditions as allowed by plant procedures. Reactor Vessel Water Level - Level 0 Function is implicitly assumed in the analysis of the recirculation line break (Ref. 2) since the analysis assumes that no LPCI flow diversion occurs when

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.a. Reactor Vessel Water Level - Low Low, Level 2 (LIS-3-58A-D) (continued)

one of the Functions assumed to be OPERABLE and capable of initiating HPCI during the transients analyzed in References 1, 2, and 3. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level - Low Low, Level 2 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level - Low Low, Level 2 Allowable Value is high enough such that for complete loss of feedwater flow, the Reactor Core Isolation Cooling (RCIC) System flow with HPCI assumed to fail will be sufficient to avoid initiation of low pressure ECCS at Reactor Vessel Water Level - Low Low Low, Level 1.

Four channels of Reactor Vessel Water Level - Low Low, Level 2 Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for HPCI Applicability Bases.

INSERT B

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3.b. Drywell Pressure - High (PIS-64-58A-D)

High pressure in the drywell could indicate a break in the RCPB. The HPCI System is initiated upon receipt of the Drywell Pressure - High Function in order to minimize the possibility of fuel damage. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible to be indicative of a LOCA inside primary containment.

Four channels of the Drywell Pressure - High Function are required to be OPERABLE when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for HPCI Applicability Bases.

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3.c. Reactor Vessel Water Level - High, Level 8 (LIS-3-208B and 208D)

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Level 8 signal is used to trip the HPCI turbine to prevent overflow into the main steam lines (MSLs). The Reactor Vessel Water Level - High, Level 8 Function is not assumed in the accident and transient analyses. It was retained since it is a potentially significant contributor to risk, thus it meets Criterion 4 of the NRC Policy Statement (Ref. 5).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.a, 5.a. Reactor Vessel Water Level - Low Low Low, Level 1 (LS-3-58A-D) (continued)

Reactor Vessel Water Level - Low Low Low, Level 1 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Low Low, Level 1 Function are required to be OPERABLE only when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip system A, while the other two channels input to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

The Reactor Vessel Water Level - Low Low Low, Level 1 Allowable Value is chosen to allow time for the low pressure core flooding systems to initiate and provide adequate cooling.

INSERT B

4.b, 5.b. Drywell Pressure - High (PIS-64-57A-D)

High pressure in the drywell could indicate a break in the RCPB. Therefore, ADS receives one of the signals necessary for initiation from this Function in order to minimize the possibility of fuel damage. The Drywell Pressure - High is assumed to be OPERABLE and capable of initiating the ADS during the accidents analyzed in Reference 2. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Drywell Pressure - High signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.b, 5.b. Drywell Pressure - High (PIS-64-57A-D) (continued)

Four channels of Drywell Pressure - High Function are only required to be OPERABLE when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip system A, while the other two channels input to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

INSERT B

4.c, 5.c. Automatic Depressurization System Initiation Timer

The purpose of the Automatic Depressurization System Initiation Timer is to delay depressurization of the reactor vessel to allow the HPCI System time to maintain reactor vessel water level. Since the rapid depressurization caused by ADS operation is one of the most severe transients on the reactor vessel, its occurrence should be limited. By delaying initiation of the ADS Function, the operator is given the chance to monitor the success or failure of the HPCI System to maintain water level, and then to decide whether or not to allow ADS to initiate, to delay initiation further by recycling the timer, or to inhibit initiation permanently. The Automatic Depressurization System Initiation Timer Function is assumed to be OPERABLE for the accident analyses of Reference 2 that require ECCS initiation.

There are two Automatic Depressurization System Initiation Timer relays, one in each of the two ADS trip systems. The Allowable Value for the Automatic Depressurization System Initiation Timer is chosen so that there is still time after depressurization for the low pressure ECCS subsystems to provide adequate core cooling.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.c, 5.c. Automatic Depressurization System Initiation Timer (continued)

Two channels of the Automatic Depressurization System Initiation Timer Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.d, 5.d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory) (LIS-3-184 and 185)

The Reactor Vessel Water Level - Low, Level 3 (Confirmatory) Function is used by the ADS only as a confirmatory low water level signal. ADS receives one of the signals necessary for initiation from Reactor Vessel Water Level - Low Low Low, Level 1 signals. In order to prevent spurious initiation of the ADS due to spurious Level 1 signals, a Level 3 (Confirmatory) signal must also be received before ADS initiation commences.

Reactor Vessel Water Level - Low, Level 3 (Confirmatory) signals are initiated from two level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

Two channels of Reactor Vessel Water Level - Low, Level 3 (Confirmatory) Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

INSERT B

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Reactor Vessel Water Level - Low Low, Level 2 (LIS-3-58A-D) (continued)

Reactor Vessel Water Level - Low Low, Level 2 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level - Low Low, Level 2 Allowable Value is set high enough such that for complete loss of feedwater flow, the RCIC System flow with high pressure coolant injection assumed to fail will be sufficient to avoid initiation of low pressure ECCS at Level 1.

Four channels of Reactor Vessel Water Level - Low Low, Level 2 Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC initiation. Refer to LCO 3.5.3 for RCIC Applicability Bases.

INSERT B →

2. Reactor Vessel Water Level - High, Level 8 (LIS-3-208A and 208C)

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Level 8 signal is used to close the RCIC steam supply valve to prevent overflow into the main steam lines (MSLs).

Reactor Vessel Water Level - High, Level 8 signals for RCIC are initiated from two level transmitters from the narrow range water level measurement instrumentation, which sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low (PIS-1-72, 76, 82, 86)

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves excluding the Recirculation Loop Sample valves.

INSERT B

(continued)

Enclosure 4

Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3
Technical Specifications (TS) Change TS-453

Instrument Setpoint Program

List of Regulatory Commitments

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1. TVA will evaluate the final Technical Specification Task Force change related to resolution of the setpoint issue within 90 days after its approval by the NRC.

2. Prior to implementation of the proposed TS change, the methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.