TVA

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

MAR 30 1994

TVA-BFN-TS-318 10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

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 SPECIFICATION (TS) NO. 318 - ANALOG TRANSMITTER/TRIP SYSTEM, LEVEL : REACTOR WATER LEVEL SETPOINTS, AND VARIOUS CALIBRATION FREQUENCIES

In accordance with the provisions of 10 CFR 50.4 and 50.90, TVA is submitting a request for an amendment (TS-318) to licenses DPR-33, DPR-52, and DPR-68 to change the TSs for Units 1, 2, and 3. The proposed change:

1. Reflects the installation of an Analog Transmitter/Trip System (ATTS) on Unit 3, which is similar to the system previously installed on Unit 2.

2. Revises the Units 1 and 3 Reactor Vessel Water Level Safety Limit to reflect the analytical limit provided by General Electric. In addition, the Level 1 Low Reactor Vessel Water Level setpoint is being revised to provide a more conservative limit. These changes were previously approved to the Unit 2 TSs.

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3. Adds or corrects Unit 2 instrument identifiers to enhance the useability of the TSs.

4. Revises calibration frequencies and funct onal test descriptions for the Unit 2 Reactor High ater Level, Reactor Core Isolation Cooling and High Pressure Coolant Injection Turbine Steam Line High Flow, and Drywell Pressure instrument channels.

5. Revises the calibration frequency for the differential pressure instrumentation, which actuates the pressure suppression chamber-reactor building vacuum breakers, in the Units 1, 2, and 3 TSs to reflect current calculations. In addition, tables that specify the minimum number of instrument channels per trip system, function, trip level setting, actions required, remarks, functional test, and instrument check are being added.

6. Corrects the capitalization of terms used on the affected Units 1, 2, and 3 TS pages in order to conform with the current TS Definitions section. This part also corrects spelling and capitalization of other words on the same pages.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the change is exempt from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). The BFN Plant Operations Review Committee and the BFN Nuclear Safety Review Board have reviewed this proposed change and determined that operation of BFN Units 1, 2, and 3 in accordance with the proposed change will not endanger the health and safety of the public. 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 to this letter provides the description and evaluation of the proposed change. This includes TVA's evaluation that the proposed change does not involve a significant hazards consideration, and is exempt from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). Enclosure 2 contains copies of the appropriate TS pages from Units 1, 2, and 3 marked-up to show the proposed change. Enclosure 3 forwards the revised TS pages for Units 1, 2, and 3 which incorporate the proposed change. Enclosure 4 contains the commitment associated with the proposed TS change.

The Unit 3 ATTS modification, reactor vessel water level safety limit change, and Level 1 Low Reactor Vessel Water Level setpoint revision portions of this amendment are needed to support restart of BFN Unit 3. By letter dated December 23, 1993, TVA submitted the current schedule showing TS amendment need dates for support of Unit 3 restart. In order to support the restart schedule, TVA requests approval of the enclosed amendment by April 20, 1995. TVA also requests that the revised TS be made effective within 30 days of NRC approval. If you have any questions about this change, please telephone me at (205) 729-2636.

Sincerely,

Pedro Salas
Manager of Site Licensing

Enclosures
cc: See page 4

Subscribed and sworn to before me
on this 31st day of MARCH 1994.

Barbara A. Blanton
Notary Public

My Commission Expires 10-30-94

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-318
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

INDEX

I. DESCRIPTION OF THE PROPOSED CHANGE

In general, this proposed change to BFN Technical Specifications consists of six parts.

Part A: The Unit 3 mechanical pressure and differential pressure indicating switches in the Reactor Protection System (RPS) and Emergency Core Cooling System (ECCS) are being replaced with an Analog Transmitter/Trip System (ATTS).

The ATTS modification includes the replacement of power supplies and associated electrical cabling, breakers and fuses. As a result, instrument identifiers are being added and/or revised, the descriptions of the required functional testing are being updated, group designators are being corrected, notes pertaining to the minimum functional test frequency are being amended, the minimum calibration frequencies are being adjusted, and an indicator range is being changed to properly reflect the new equipment.

The ATTS provides the following system upgrades:

- Reduces the functional tests and calibration frequencies for the primary sensors.
- Decreases the duration and complexity of required testing and calibration of the inputs for safety related parameters.
- Reduces testing and maintenance related scrams.
- Reduces the number of reportable events related to setpoint drift.

In addition, the revised design implements the diversity requirements associated with the Anticipated Transient Without Scram (ATWS) system required by 10 CFR 50.62.

The RPS provides timely protection against the onset and consequences of conditions that threaten the integrities of the fuel barrier (uranium dioxide sealed in cladding) and the nuclear system process barrier. Excessive temperature threatens to perforate the cladding or melt the uranium dioxide. Excessive pressure threatens to rupture the nuclear system process barrier. The RPS limits the uncontrolled release of radioactive material by terminating excessive temperature and pressure increases through the initiation of an automatic scram.

The RPS includes the motor-generator power supplies with associated control and indicating equipment, sensors, relays, bypass circuitry, and switches that cause rapid insertion of control rods (scram) to shut down the reactor. It also includes outputs to the process computer system and annunciators. The Reactor Protection System is designed to meet the intent of the IEEE proposed criteria for nuclear power plant protection systems (IEEE-279-1971). A detailed description of the RPS is included in Section 7.2.3 of the Browns Ferry Final Safety Analysis Report (FSAR).

The controls and instrumentation for the ECCS initiate appropriate responses from the various cooling systems so that the fuel is adequately cooled under abnormal or accident conditions. The cooling provided by the systems restricts the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant is lost from the nuclear system.

Even after the reactor is shut down from power operation by the full insertion of all control rods, heat continues to be generated in the fuel as radioactive fission products decay. An excessive loss of reactor coolant allows the fuel temperature to rise, cladding to melt, and fission products in the fuel to be released. If the temperatures in the reactor rise to a sufficiently high value, a metal (zirconium) water reaction occurs which releases energy. Such a reaction increases the pressure inside the nuclear system and the primary containment. This threatens the integrity of the barriers, which are relied upon to prevent the uncontrolled release of radioactive materials. The controls and instrumentation for the ECCS prevent such a sequence of events by actuating core cooling systems in time to limit fuel temperatures to acceptable levels (less than 2200°F). A detailed description of the ECCS is included in Section 7.4.3 of the Browns Ferry FSAR.

The ATTS has been installed and successfully operated on BFN Unit 2. The specific differences between the Unit 3 and Unit 2 instrumentation are described in the Safety Analysis section. Similar Technical Specification changes were previously approved for Unit 2 (References 1 through 8).

Part B: The Units 1 and 3 reactor vessel water level safety limit is being revised to reflect the analytical limit provided by General Electric and the Level 1 Low Reactor Vessel Water Level setpoint is being revised to provide a more conservative limit.

The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit requires unit shutdown and review by the Nuclear Regulatory Commission before resumption of the unit operation. Operation beyond such a limit, the reactor vessel water level safety limit in this case, may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

The limiting safety system setting is a setting on instrumentation which initiates the automatic protective action at a level such that the safety limits will not be exceeded. The Level 1 Low Reactor Vessel Water Level setpoint is the limiting safety system setting which is being revised by this proposed Technical Specification amendment. The region between the safety limit and these settings represent margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

TVA committed, in the Browns Ferry Nuclear Performance Plan, to ensure that calculations exist to support the safe shutdown basis of Unit 2. During the process of generating these setpoint and accuracy calculations for plant parameters for which no calculational basis could be found, it was determined that the Unit 2 Level 1 Low Reactor Vessel Water Level trip setting was not conservative based on the new calculation methodology. The Reactor Vessel Water Level Safety Limit and the Level 1 Low Reactor Vessel Water Level setpoint have previously been changed in the BFN Unit 2 Technical Specifications (References 9 through 11). Similar revisions to the Units 1 and 3 Technical Specification are required to reflect the Unit 3 specific calculations and to make the Technical Specifications consistent for all three units.

Part C: For Unit 2, RPS and ECCS instrument identifiers are being added or corrected to enhance useability of the Technical Specifications. These changes do not reflect a change in equipment, operation of the associated system, or the safety function of that system.

Part D: For Unit 2, Reactor High Water Level, Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) Turbine Steam Line High Flow, and Drywell Pressure instrumentation calibration frequencies and functional test descriptions are being revised to reflect current calculations and test methods. These changes do not reflect a change in equipment, operation of the associated system, or the safety function of that system.

Part E: For Units 1, 2, and 3, the differential pressure instrumentation, which actuates the pressure suppression chamber-reactor building vacuum breakers, calibration frequency is being revised. In addition, tables that specify the minimum number of instrument channels per trip system, function, trip level setting, actions required, remarks, functional test, and instrument check are being added.

Automatic vacuum relief devices are used to prevent the primary containment from exceeding the external design pressure. The drywell vacuum relief valves draw air from the pressure suppression chamber. The pressure suppression chamber vacuum relief device draws air from the Reactor Building.

The pressure suppression chamber vacuum relief system consists of two vacuum breakers in series in each of two lines to atmosphere. One valve is air-operated and is actuated by a differential pressure signal. This valve fails open upon a loss of power. These are the valves being addressed by this Technical Specification change. The second valve is self-actuating.

Part F: Corrects the capitalization of terms used on the affected Units 1, 2, and 3 TS pages in order to conform with the current TS Definitions section. This part also corrects spelling and capitalization of other words on the same pages.

The specific proposed changes to the BFN Technical Specifications are delineated below. The applicability to which unit is specified for every change.

1. For Units 1 and 3. Proposed change to Safety Limit 1.1.B, Power Transient. Capitalize the words safety limit in two places as shown below:

   Existing Technical Specifications:

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

   Proposed Technical Specifications:

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

2. For Units 1 and 3. Proposed change to Safety Limit 1.1.C, Reactor Vessel Water Level. Change the Reactor Vessel Water Level Safety Limit from greater than or equal to 378 inches to greater than or equal to 372.5 inches as shown below:

   Existing Technical Specifications:

   "Whenever there is irradiated fuel in the reactor vessel, the water level shall be greater than or equal to 378 inches above vessel zero."

   Proposed Technical Specifications:

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

3. For Units 1 and 3. Proposed change to Limiting Safety System Setting 2.1.C.1, Water Level Trip Settings. Change the Core Spray and Low Pressure Coolant Injection (LPCI) setpoint from greater than or equal to 378 inches to greater than or equal to 398 inches as shown below:

   Existing Technical Specifications:

      "Core spray and LPCI actuation -- reactor low water level ≥ 378 in. above vessel zero"

   Proposed Technical Specifications:

      Core spray and LPCI actuation -- reactor low water level ≥ 398 in. above vessel zero

4. For Units 1 and 3. Proposed change to Limiting Safety System Setting 2.1.C.3, Water Level Trip Settings. Change the main steam isolation valve closure setpoint from greater than or equal to 378 inches to greater than or equal to 398 inches as shown below:

   Existing Technical Specifications:

      "Main steam isolation valve closure -- reactor low water level ≥ 378 in. above vessel zero"

   Proposed Technical Specifications:

      Main steam isolation valve closure -- reactor low water level ≥ 398 in. above vessel zero

5. For Units 1 and 3. Change to Bases 1.1. Revise the Reactor Vessel Water Level Safety Limit from 378 inches to 372.5 inches and delete the reference to the lower reactor low water level trip as shown below:

Existing Bases:

"The safety limit has been established at 378 inches above vessel zero to provide a point which can be monitored and also provide adequate margin to assure sufficient cooling. This point is the lower reactor low water level trip."

Revised Bases:

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

6. For Units 1 and 3. Addition to the references for Bases 1.1:

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

7. For Unit 3 only. Proposed additions to Table 3.1.A, Reactor Protection System (SCRAM) Instrumentation Requirements:

   a. For the High Reactor Pressure trip, insert instrument identifiers: (PIS-3-22AA,BB,C,D).

   b. For the High Drywell Pressure trip, insert instrument identifiers: (PIS-64-56 A-D).

   c. For the Reactor Low Water Level trip, insert instrument identifiers: (LIS-3-203 A-D).

   d. For the Turbine First Stage Pressure Permissive, insert instrument identifiers: (PIS-1-81A&B, PIS-1-91A&B).

8. For Unit 3 only. Proposed changes to Table 4.1.A, Reactor Protection System (SCRAM) Instrumentation Functional Tests Minimum Functional Test Frequencies for Safety Instr. and Control Circuits:

a. For the High Reactor Pressure trip, insert instrument identifiers: (PIS-3-22AA, BB, C, D), insert a reference to Footnote 7 in the Functional Test Column, remove the reference to Footnote 1 in the Minimum Frequency column, and revise the group designator as shown below:

Existing Technical Specifications:

"<u>Group (2)</u>

A"

Proposed Technical Specifications:

<u>Group (2)</u>

B

b. For the High Drywell Pressure trip, insert instrument identifiers: (PIS-64-56 A-D), insert a reference to Footnote 7 in the Functional Test Column, remove the reference to Footnote 1 in the Minimum Frequency column, and revise the group designator as shown below:

Existing Technical Specifications:

"<u>Group (2)</u>

A"

Proposed Technical Specifications:

<u>Group (2)</u>

B

c. For the Reactor Low Water Level trip, insert instrument identifiers: (LIS-3-203 A-D), insert a reference to Footnote 7 in the Functional Test Column, remove the reference to Footnote 1 in the Minimum Frequency column, and revise the group designator as shown below:

Existing Technical Specifications:

"Group (2)

A"

Proposed Technical Specifications:

Group (2)

B

d. Proposed changes to Table 4.1.A. For the Turbine First Stage Pressure Permissive, insert instrument identifiers: (PIS-1-81A and B, PIS-1-91A and B), insert a reference to Footnote 7 in the Functional Test Column, and revise the group designator as shown below:

Existing Technical Specifications:

"Group (2)

A"

Proposed Technical Specifications:

Group (2)

B

9. For Unit 3 only. Proposed changes to Table 4.1.B, Reactor Protection System (SCRAM) Instrumentation Calibration Minimum Calibration Frequencies for Reactor Protection Instrument Channels:

a. For the High Reactor Pressure trip, insert instrument identifiers: (PIS-3-22AA, BB, C, D) and revise the group designator and minimum calibration frequency as shown below:

(1) Existing Technical Specifications:

"<u>Group (1)</u>

A"

Proposed Technical Specifications:

<u>Group (1)</u>

B

(2) Existing Technical Specifications:

"<u>Minimum Frequency (2)</u>

Every 3 Months"

Proposed Technical Specifications:

<u>Minimum Frequency (2)</u>

Once/6 Months (9)

b. For the High Drywell Pressure trip, insert instrument identifiers: (PIS-64-56 A-D) and revise the group designator and minimum calibration frequency as shown below:

(1) Existing Technical Specifications:

"Group (1)

A"

Proposed Technical Specifications:

Group (1)

B

(2) Existing Technical Specifications:

"Minimum Frequency (2)

Every 3 Months"

Proposed Technical Specifications:

Minimum Frequency (2)

Once/18 Months (9)

c. For the Reactor Low Water Level trip, insert instrument identifiers: (LIS-3-203 A-D) and revise the group designator and minimum calibration frequency as shown below:

(1) Existing Technical Specifications:

"<u>Group (1)</u>

A"

Proposed Technical Specifications:

<u>Group (1)</u>

B

(2) Existing Technical Specifications:

"<u>Minimum Frequency (2)</u>

Every 3 Months"

Proposed Technical Specifications:

<u>Minimum Frequency (2)</u>

Once/18 Months (9)

d. For the Turbine First Stage Pressure Permissive, insert instrument identifiers: (PIS-1-81 A&B, PIS-1-91 A&B) and revise the group designator and minimum calibration frequency as shown below:

(1) Existing Technical Specifications:

"<u>Group (1)</u>

A"

Proposed Technical Specifications:

<u>Group (1)</u>

B

(2) Existing Technical Specifications:

"<u>Minimum Frequency (2)</u>

Every 6 Months"

Proposed Technical Specifications:

<u>Minimum Frequency (2)</u>

Once/18 Months (9)

10. For Unit 3 only. Changes to Bases 3.1.

a. Capitalize the words reactor protection system in two places as shown below:

Existing Bases:

"The reactor protection system automatically initiates a reactor scram to: ... The reactor protection system is made up of two independent trip systems (refer to Section 7.2, FSAR)."

Revised Bases:

The Reactor Protection System automatically initiates a reactor scram to: ... The Reactor Protection System is made up of two independent trip systems (refer to Section 7.2, FSAR).

b. Capitalize the words limiting conditions for operation and decapitalize the word inoperable as shown below:

Existing Bases:

"This specification provides the limiting conditions for operation necessary to preserve ... When necessary, one channel may be made INOPERABLE for brief intervals to conduct required functional tests and calibrations."

Revised Bases:

This specification provides the LIMITING CONDITIONS FOR OPERATION necessary to preserve ... When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

11. For Unit 3 only. Changes to Bases 3.1. Insert the following paragraph as part of the description of the reactor protection system:

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

12. Proposed changes to Table 3.2.A, Primary Containment and Reactor Building Isolation Instrumentation.

   a. For Unit 3 only. For the Reactor Low Water Level Instrument Channel that has a trip level setting of ≥ 538" above vessel zero, insert instrument identifiers: (LIS-3-203 A-D) and delete the reference to the isolation groups as shown below:

      Existing Technical Specifications:

      "Remarks

      1. Below trip setting does the following:
         ...

         b. Initiates Primary Containment Isolation (Groups 2, 3, and 6)"

      Proposed Technical Specifications:

      Remarks

      1. Below trip setting does the following:
         ...

         b. Initiates Primary Containment Isolation

   b. For Units 1 and 3. For the Reactor Low Water Level Instrument Channel that currently has a Trip Level Setting of ≥ 378" above vessel zero, revise the trip level setting to ≥ 398" as shown below:

      Current Technical Specifications:

      Trip Level Setting

      "≥ 378" above vessel zero"

      Proposed Technical Specifications:

      Trip Level Setting

      ≥ 398" above vessel zero

c. For Unit 3 only. For the Reactor Low Water Level Instrument Channel that currently has a Trip Level Setting of ≥ 378" above vessel zero, revise the instrument identifiers as shown below:

Current Technical Specifications:

Function
"Instrument Channel - Reactor Low Water Level (LIS-3-56A-D, SW #1)"

Proposed Technical Specifications:

Function
Instrument Channel - Reactor Low Water Level (LIS-3-56A-D)

d. For Unit 3 only. For the High Drywell Pressure Instrument Channel, revise the instrument identifiers as shown below:

Current Technical Specifications:

Function
"Instrument Channel - High Drywell Pressure (6) (PS-64-56A-D)"

Proposed Technical Specifications:

Function
Instrument Channel - High Drywell Pressure (6) (PIS-64-56A-D)

e. For Unit 3 only. For the Low Pressure Main Steam Line Instrument Channel, insert instrument identifiers: (PIS-1-72, 76, 82, 86)

f. For Unit 3 only. For the High Flow Main Steam Line Instrument Channel, insert instrument identifiers: (PdIS-1-13A-D, 25A-D, 36A-D, 50A-D)

13. Proposed changes to Table 3.2.B, Instrumentation that Initiates or Controls the Core and Containment Cooling Systems.

   a. For Unit 3 only. For the two Reactor Low Water Level Instrument Channel entries that have a trip level setting of ≥ 470" above vessel zero, insert instrument identifiers: (LIS-3-58A-D).

   b. For Units 1 and 3. For the first listing of the Reactor Low Water Level Instrument Channel that has a trip level setting of ≥ 378" above vessel zero, revise the trip level setting as shown below:

      Current Technical Specifications:

      <u>Trip Level Setting</u>

      "≥ 378" above vessel zero."

      Proposed Technical Specifications:

      <u>Trip Level Setting</u>

      ≥ 398" above vessel zero.

   c. For Unit 3 only. For the first listing of the Reactor Low Water Level Instrument Channel that has a trip level setting of ≥ 378" above vessel zero, revise the instrument identifiers as shown below:

      Current Technical Specifications:

      <u>Function</u>

      "Instrument Channel -
      Reactor Low Water Level
      (LIS-3-58A-D, SW #1)"

      Proposed Technical Specifications:

      <u>Function</u>

      Instrument Channel -
      Reactor Low Water Level
      (LS-3-58A-D)

d. For Units 1 and 3. For the next listing of the Reactor Low Water Level Instrument Channel that has a trip level setting of ≥ 378" above vessel zero, revise the trip level setting as shown below:

Current Technical Specifications:

Trip Level Setting

"≥ 378" above vessel zero."

Proposed Technical Specifications:

Trip Level Setting

≥ 398" above vessel zero.

e. For Unit 3 only. For the next listing of the Reactor Low Water Level Instrument Channel that has a trip level setting of ≥ 378" above vessel zero, revise the instrument identifiers as shown below:

Current Technical Specifications:

Function

"Instrument Channel -
Reactor Low Water Level
(LIS-3-58A-D, SW #2)"

Proposed Technical Specifications:

Function

Instrument Channel -
Reactor Low Water Level
(LS-3-58A-D)

f. For Unit 3 only. For the Reactor Low Water Level Permissive Instrument Channel, revise the instrument identifiers as shown below:

Current Technical Specifications:

Function
"Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184 & 185, SW #1)"

Proposed Technical Specifications:

Function
Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184, 185)

g. For Unit 3 only. For the Reactor Low Water Level Instrument Channel that has a trip level setting of ≥ 312 5/16" above vessel zero, revise the instrument identifiers as shown below:

Current Technical Specifications:

Function
"Instrument Channel - Reactor Low Water Level (LITS-3-52 and 62, SW #1)"

Proposed Technical Specifications:

Function
Instrument Channel - Reactor Low Water Level (LIS-3-52 and LIS-3-62A)

h. For Unit 3 only. For the Drywell High Pressure Instrument Channel with a trip level setting of 1 $\leq$ p $\leq$ 2.5 psig, revise the instrument identifiers as shown below:

Current Technical Specifications:

Function
"Instrument Channel - Drywell High Pressure (PS-64-58 E-H)"

Proposed Technical Specifications:

Function
Instrument Channel - Drywell High Pressure (PIS-64-58 E-H)

i. For Unit 3 only. For the first Drywell High Pressure Instrument Channel with a trip level setting of $\leq$ 2.5 psig, revise the instrument identifiers as shown below:

Current Technical Specifications:

Function
"Instrument Channel - Drywell High Pressure (PS-64-58 A-D, SW #2)"

Proposed Technical Specifications:

Function
Instrument Channel - Drywell High Pressure (PIS-64-58 A-D)

j. For Unit 3 only. For the second Drywell High Pressure Instrument Channel with a trip level setting of ≤ 2.5 psig, revise the instrument identifiers as shown below:

Current Technical Specifications:

Function
"Instrument Channel - Drywell High Pressure (PS-64-58A-D, SW #1)"

Proposed Technical Specifications:

Function
Instrument Channel - Drywell High Pressure (PIS-64-58A-D)

k. For Unit 3 only. For the third Drywell High Pressure Instrument Channel with a trip level setting of ≤ 2.5 psig, revise the instrument identifiers as shown below:

Current Technical Specifications:

Function
"Instrument Channel - Drywell High Pressure (PS-64-57A-D)"

Proposed Technical Specifications:

Function
Instrument Channel - Drywell High Pressure (PIS-64-57A-D)

l. For Unit 3 only. For the Reactor Low Pressure Instrument Channel with a trip level setting of 450 psig ± 15, revise the instrument identifiers as shown below:

Current Technical Specifications:

Function

"Instrument Channel -
Reactor Low Pressure
(PS-3-74 A & B, SW #2)
(PS-68-95, SW #2)
(PS-68-96, SW #2)"

Proposed Technical Specifications:

Function

Instrument Channel -
Reactor Low Pressure
(PIS-3-74A & B)
(PIS-68-95, 96)

m. For Unit 3 only. For the Reactor Low Pressure Instrument Channel with a trip level setting of 230 psig ± 15, revise the instrument identifiers as shown below:

Current Technical Specifications:

Function

"Instrument Channel -
Reactor Low Pressure
(PS-3-74 A & B, SW #1)
(PS-68-95, SW #1)
(PS-68-96, SW #1)"

Proposed Technical Specifications:

Function

Instrument Channel -
Reactor Low Pressure
(PS-3-74A & B)
(PS-68-95, 96)

n. For Unit 3 only. For the Reactor High Water Level Instrument Channel that has a trip level setting of ≤ 583" above vessel zero and above the trip setting trips the RCIC turbine, insert instrument identifiers: (LIS-3-208A and LIS-3-208C).

o. For Unit 3 only. For the RCIC Turbine Steam Line High Flow Instrument Channel, insert instrument identifiers: (PDIS-71-1A and 1B).

p. For Unit 3 only. For the next Reactor High Water Level Instrument Channel that has a trip level setting of ≤ 583" above vessel zero, and above the trip setting trips the HPCI turbine, insert instrument identifiers: (LIS-3-208B and LIS-3-208D).

q. For Unit 3 only. For the HPCI Turbine Steam Line High Flow Instrument Channel, insert instrument identifiers: (PDIS-73-1A and 1B)

r. For Units 1 and 3. For Note 15 in Notes for Table 3.2.B section, revised the vessel low water level setpoint from ≥ 378" to ≥ 398" as shown below:

Current Technical Specifications:

15. "The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level (≥ 378" above vessel zero) originating in the core spray system trip system."

Proposed Technical Specifications:

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

14. For Unit 3 only. Proposed changes to Table 3.2.F, Surveillance Instrumentation.

a. For the Reactor Water Level Instrument, revise the instrument identifiers as shown below:

Current Technical Specifications:

<u>Instrument #</u>

"LI-3-46 A
LI-3-46 B"

Proposed Technical Specifications:

<u>Instrument #</u>

LI-3-58A
LI-3-58B

b. For the Reactor Pressure Instrument, revise the instrument identifiers and range as shown below:

(1) Current Technical Specifications:

<u>Instrument #</u>

"PI-3-54
PI-3-61"

Proposed Technical Specifications:

<u>Instrument #</u>

PI-3-74A
PI-3-74B

(2) Current Technical Specifications:

<u>Type Indication and Range</u>

"Indicator 0-1500 psig"

Proposed Technical Specifications:

<u>Type Indication and Range</u>

Indicator 0-1200 psig

15. For Units 2 and 3. Proposed correction to the spelling of the word instrumentation in Table 3.2.L:

Current Title:

"Anticipated Transient Without Scram (ATWS) - Recirculation Pump Test (RPT) Surveillance Instrumenation"

Proposed Title:

Anticipated Transient Without Scram (ATWS) - Recirculation Pump Test (RPT) Surveillance Instrumentation

16. For Units 2 and 3. Proposed addition to Table 3.2.L, Anticipated Transient Without Scram (ATWS) - Recirculation Pump Test (RPT) Surveillance Instrumentation.

   a. For the ATWS/RPT Logic Reactor Dome Pressure High Function, insert instrument identifiers: (PIS-3-204 A-D).

   b. For the Reactor Vessel Level Low Function, insert instrument identifiers: (LS-3-58 A1-D1).

17. For Unit 3 only. Proposed changes to Table 4.2.A, Surveillance Requirements for Primary Containment and Reactor Building Isolation Instrumentation.

a. For the first listing of the Reactor Low Water Level Instrument Channel, revise the instrument identifiers, insert a reference to Footnote 28 to the Functional Test column and revise the calibration frequency as shown below:

(1) Current Technical Specifications:

<u>Function</u>

"Instrument Channel -
Reactor Low Water Level
(LIS-3-203A-D, SW 2-3)"

Proposed Technical Specifications:

<u>Function</u>

Instrument Channel -
Reactor Low Water Level
(LIS-3-203A-D)

(2) Current Technical Specifications:

<u>Calibration Frequency</u>

"(5)"

Proposed Technical Specifications:

<u>Calibration Frequency</u>

once/18 Months (29)

b. For the second listing of the Reactor Low Water Level Instrument Channel, revise the instrument identifiers, insert a reference to Footnote 28 to the Functional Test column and revise the calibration frequency as shown below:

(1) Current Technical Specifications:

<u>Function</u>

"Instrument Channel -
Reactor Low Water Level
(LIS-3-56A-D, SW #1)"

Proposed Technical Specifications:

<u>Function</u>

Instrument Channel -
Reactor Low Water Level
(LIS-3-56A-D)

(2) Current Technical Specifications:

<u>Calibration Frequency</u>

"once/3 month"

Proposed Technical Specifications:

<u>Calibration Frequency</u>

once/18 months (29)

c. For the High Drywell Pressure Instrument Channel, revise the instrument identifiers, insert a reference to Footnote 28 to the Functional Test column and revise the calibration frequency as shown below:

(1) Current Technical Specifications:

<u>Function</u>

"Instrument Channel -
High Drywell Pressure
(PS-64-56A-D)"

Proposed Technical Specifications:

<u>Function</u>

Instrument Channel -
High Drywell Pressure
(PIS-64-56A-D)

(2) Current Technical Specifications:

<u>Calibration Frequency</u>

"(5)"

Proposed Technical Specifications:

<u>Calibration Frequency</u>

once/18 Months (29)

d. For the Low Pressure Main Steam Line Instrument Channel, insert instrument identifiers: (PIS-1-72, 76, 82, 86) and revise the functional test and calibration frequency as shown below:

(1) Current Technical Specifications:

Functional Test

"once/3 months (27)"

Proposed Technical Specifications:

Functional Test

(28) (27)

(2) Current Technical Specifications:

Calibration Frequency

"once/3 months"

Proposed Technical Specifications:

Calibration Frequency

once/18 Months (29)

e. For the High Flow Main Steam Line Instrument Channel, insert instrument identifiers: (PdIS-1-13A-D, 25A-D, 36A-D, 50A-D) and revise the functional test and calibration frequency as shown below:

(1) Current Technical Specifications:

<u>Functional Test</u>

"once/3 months (27)"

Proposed Technical Specifications:

<u>Functional Test</u>

(28) (27)

(2) Current Technical Specifications:

<u>Calibration Frequency</u>

"once/3 months"

Proposed Technical Specifications:

<u>Calibration Frequency</u>

once/18 Months (29)

18. Proposed changes to Table 4.2.B, Surveillance Requirements for Instrumentation that Initiate or Control the CSCS.

    a. For Units 2 and 3. For the first listing of the Reactor Low Water Level Instrument Channel, revise the instrument identifiers as shown below:

       Current Technical Specifications:

       <u>Function</u>

       "Instrument Channel -
       Reactor Low Water Level
       (LIS-3-58A-D)"

       Proposed Technical Specifications:

       <u>Function</u>

       Instrument Channel -
       Reactor Low Water Level
       (LS-3-58A-D, LIS-3-58A-D)

    b. For Unit 3 only. For the first listing of the Reactor Low Water Level Instrument Channel, insert a reference to Footnote 28 to the Functional Test column and revise the calibration frequency as shown below:

       Current Technical Specifications:

       <u>Calibration Frequency</u>

       "once/3 months"

       Proposed Technical Specifications:

       <u>Calibration Frequency</u>

       once/18 months (29)

c. For Unit 3 only. For the second listing of the Reactor Low Water Level Instrument Channel (LIS-3-184 & 185), insert a reference to Footnote 28 to the Functional Test column and revise the calibration frequency as shown below:

Current Technical Specifications:

Calibration Frequency

"once/3 months"

Proposed Technical Specifications:

Calibration Frequency

once/18 months (29)

d. For Unit 3 only. For the third listing of the Reactor Low Water Level Instrument Channel, revise the instrument identifiers, insert a reference to Footnote 28 to the Functional Test column and revise the calibration frequency as shown below:

(1) Current Technical Specifications:

<u>Function</u>

"Instrument Channel - Reactor Low Water Level (LITS-3-52 & 62)"

Proposed Technical Specifications:

<u>Function</u>

Instrument Channel - Reactor Low Water Level (LIS-3-52 & 62A)

(2) Current Technical Specifications:

<u>Calibration Frequency</u>

"once/3 months"

Proposed Technical Specifications:

<u>Calibration Frequency</u>

once/18 months (29)

e. For Unit 3 only. For the first Drywell High Pressure Instrument Channel, revise the instrument identifiers, insert a reference to Footnote 28 to the Functional Test column and revise the calibration frequency as shown below:

(1) Current Technical Specifications:

Function

"Instrument Channel -
Drywell High Pressure
(PS-64-58E-H)"

Proposed Technical Specifications:

Function

Instrument Channel -
Drywell High Pressure
(PIS-64-58E-H)

(2) Current Technical Specifications:

Calibration Frequency

"once/3 months"

Proposed Technical Specifications:

Calibration Frequency

once/18 months (29)

f. For Unit 3 only. For the second Drywell High Pressure Instrument Channel, revise the instrument identifiers, insert a reference to Footnote 28 to the Functional Test column and revise the calibration frequency as shown below:

(1) Current Technical Specifications:

<u>Function</u>

"Instrument Channel -
Drywell High Pressure
(PS-64-58A-D)"

Proposed Technical Specifications:

<u>Function</u>

Instrument Channel -
Drywell High Pressure
(PIS-64-58A-D)

(2) Current Technical Specifications:

<u>Calibration Frequency</u>

"once/3 months"

Proposed Technical Specifications:

<u>Calibration Frequency</u>

once/18 months (29)

g. For Unit 3 only. For the third Drywell High Pressure Instrument Channel, revise the instrument identifiers, insert a reference to Footnote 28 to the Functional Test column and revise the calibration frequency as shown below:

(1) Current Technical Specifications:

<u>Function</u>

"Instrument Channel -
Drywell High Pressure
(PS-64-57A-D)"

Proposed Technical Specifications:

<u>Function</u>

Instrument Channel -
Drywell High Pressure
(PIS-64-57A-D)

(2) Current Technical Specifications:

<u>Calibration Frequency</u>

"once/3 months"

Proposed Technical Specifications:

<u>Calibration Frequency</u>

once/18 months (29)

h. For Unit 3 only. For the Reactor Low Pressure Instrument Channel, revise the instrument identifiers, insert a reference to Footnote 28 to the Functional Test column and revise the calibration frequency as shown below:

(1) Current Technical Specifications:

<u>Function</u>

"Instrument Channel -
Reactor Low Pressure
(PS-3-74A & B)
(PS-68-95)
(PS-68-96)"

Proposed Technical Specifications:

<u>Function</u>

Instrument Channel -
Reactor Low Pressure
(PIS-3-74A&B, PS-3-74A&B)
(PIS-68-95, PS-68-95)
(PIS-68-96, PS-68-96)

(2) Current Technical Specifications:

<u>Calibration Frequency</u>

"once/3 months"

Proposed Technical Specifications:

<u>Calibration Frequency</u>

once/6 months (29)

i. For Units 2 and 3. For the Reactor High Water Level Instrument Channel, insert instrument identifiers: (LIS-3-208A-D).

j. For Unit 2 only. For the Reactor High Water Level Instrument Channel, insert a reference to Footnote 27 to the Functional Test column and revise the calibration frequency as shown below:

Current Technical Specifications:

<u>Calibration Frequency</u>

"Once/3 months"

Proposed Technical Specifications:

<u>Calibration Frequency</u>

Once/18 months (28)

k. For Unit 3 only. For the Reactor High Water Level Instrument Channel, insert a reference to Footnote 28 to the Functional Test column and revise the calibration frequency as shown below:

Current Technical Specifications:

<u>Calibration Frequency</u>

"once/3 months"

Proposed Technical Specifications:

<u>Calibration Frequency</u>

once/18 months (29)

l. For Unit 2 only. For the RCIC Turbine Steam Line High Flow Instrument Channel, insert a reference to Footnote 27 to the Functional Test column and revise the calibration frequency as shown below:

Current Technical Specifications:

Calibration Frequency

"Once/3 months"

Proposed Technical Specifications:

Calibration Frequency

Once/18 months (28)

m. For Unit 3 only. For the RCIC Turbine Steam Line High Flow Instrument Channel, insert a reference to Footnote 28 to the Functional Test column and revise the calibration frequency as shown below:

Current Technical Specifications:

Calibration Frequency

"once/3 months"

Proposed Technical Specifications:

Calibration Frequency

once/18 months (29)

n. For Unit 2 only. For the HPCI Turbine Steam Line High Flow Instrument Channel, insert a reference to Footnote 27 to the Functional Test column and revise the calibration frequency as shown below:

Current Technical Specifications:

Calibration Frequency

"Once/3 months"

Proposed Technical Specifications:

Calibration Frequency

Once/18 months (28)

o. For Unit 3 only. For the HPCI Turbine Steam Line High Flow Instrument Channel, insert a reference to Footnote 28 to the Functional Test column and revise the calibration frequency as shown below:

Current Technical Specifications:

Calibration Frequency

"once/3 months"

Proposed Technical Specifications:

Calibration Frequency

once/18 months (29)

19. Proposed changes to Table 4.2.F, Minimum Test and Calibration Frequency for Surveillance Instrumentation.

a. For Unit 3 only. For the Reactor Water Level Instrument Channel, insert instrument identifiers: (LI-3-58A&B).

b. For Units 2 and 3. For the Reactor Water Level Instrument Channel, revise the calibration frequency as shown below:

Current Technical Specifications:

Calibration Frequency

"Once/6 months"

Proposed Technical Specifications:

Calibration Frequency

Once/18 months

c. For Unit 3 only. For the Reactor Pressure Instrument Channel, insert instrument identifiers: (PI-3-74A&B).

d. For Unit 3 only. For the third Drywell Pressure Instrument Channel, revise the instrument identifiers as shown below:

Current Technical Specifications:

Instrument Channel

"11) Drywell Pressure (PS-64-58B)"

Proposed Technical Specifications:

Instrument Channel

11) Drywell Pressure (PIS-64-58A)

e. For Units 2 and 3. For the third Drywell Pressure Instrument Channel, revise the calibration frequency as shown below:

Current Technical Specifications:

Calibration Frequency

"Once/6 months"

Proposed Technical Specifications:

Calibration Frequency

Once/18 months

20. For Units 1 and 3. Changes to Bases 3.2.

a. Capitalize the words primary containment integrity as shown below:

Existing Bases:

"Such instrumentation must be available whenever primary containment integrity is required."

Revised Bases:

Such instrumentation must be available whenever PRIMARY CONTAINMENT INTEGRITY is required.

b. Revise the reactor vessel water low level trip that closes the Main Steam Isolation Valves, the Main Steam Line Drain Valves, and the Reactor Water Sample Valves from 378 inches to greater than or equal to 398 inches as shown below:

Existing Bases:

"The low water level instrumentation set to trip at 378 inches above vessel zero (Table 3.2.B) closes the Main Steam Isolation Valves, the Main Steam Line Drain Valves, and the Reactor Water Sample Valves (Group 1)."

Revised Bases:

The low water level instrumentation set to trip at ≥ 398 inches above vessel zero (Table 3.2.B) closes the Main Steam Isolation Valves, the Main Steam Line Drain Valves, and the Reactor Water Sample Valves (Group 1).

c. Revise the reactor vessel water low level trip that Initiates the LPCI, Cores Spray Pumps, contributes to ADS initiation, and starts the diesel generators from 378 inches to greater than or equal to 398 inches as shown below:

Existing Bases:

"The low reactor water level instrumentation that is set to trip when reactor water level is 378 inches above vessel zero (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation, and starts the diesel generators."

Revised Bases:

The low reactor water level instrumentation that is set to trip when reactor water level is ≥ 398 inches above vessel zero (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation, and starts the diesel generators.

21. For Units 1, 2, and 3. Proposed change to Limiting Conditions For Operation (LCO) 3.7.A.3.a, Primary Containment - Pressure Suppression Chamber - Reactor Building Vacuum Breakers. Capitalize the words primary containment integrity and revise the LCO to refer to Table 3.7.A for the pressure suppression chamber-reactor building vacuum breakers actuation setpoint as shown below:

Existing LCO:

"Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when primary containment integrity is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be 0.5 psid."

Revised LCO:

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

22. For Units 1, 2, and 3. Proposed change to Surveillance Requirement 4.7.A.3.a, Primary Containment - Pressure Suppression Chamber - Reactor Building Vacuum Breakers. Revise the Surveillance Requirement to refer to Table 4.7.A for the pressure suppression chamber-reactor building vacuum breakers calibration frequency as shown below:

Existing Surveillance Requirement:

"The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation each three months."

Revised Surveillance Requirement:

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

23. For Units 1, 2, and 3, Insert a new Table 3.7.A, Instrumentation for Containment System, as shown below:

TABLE 3.7.A

INSTRUMENTATION FOR CONTAINMENT SYSTEMS

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

Footnote:

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

24. For Units 1, 2, and 3, Insert a new Table 4.7.A, Containment System Instrumentation Surveillance Requirements, as shown below:

TABLE 4.7.A

CONTAINMENT SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

Footnotes:

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

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

25. Change to LCO 3.7.A.3.b, Primary Containment - Pressure Suppression Chamber - Reactor Building Vacuum Breakers.

a. For Units 1 and 3. Decapitalize the word inoperable and capitalize the words primary containment integrity as shown below:

Existing LCO:

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

Revised LCO:

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

b. For Unit 2. Capitalize the words primary containment integrity as shown below:

Existing LCO:

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

Revised LCO:

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

26. Change to Surveillance Requirement 4.7.A.4.b, Primary Containment - Drywell-Pressure Suppression Chamber Vacuum Breakers.

   a. For Units 1 and 3. Decapitalize the word inoperable in two places and capitalize the word operability as shown below:

      Existing Surveillance Requirement:

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

      Revised Surveillance Requirement:

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

b. For Unit 2. Capitalize the word operability as shown below:

Existing Surveillance Requirement:

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

Revised Surveillance Requirement:

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

27. Changes to Bases 3.7.D/4.7.D, Primary Containment Isolation Valves.

a. For Unit 1. Correct the spelling of the word specifications as shown below:

Existing Bases:

"The procedures are subject to the change control provisions for plant procedures in the administrative controls section of the Technical Specifictions."

Revised Bases:

The procedures are subject to the change control provisions for plant procedures in the administrative controls section of the Technical Specifications.

b. For Units 1 and 3. Revise the reactor vessel water low level process line isolation trip from 378 inches to greater than or equal to 398 inches in two places as shown below:

Existing Bases:

"Group 1 - Process lines are isolated by reactor vessel low water level (378") in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. ... The reactor water sample line valves isolate only on reactor low water level at 378" or main steam line high radiation."

Revised Bases:

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

II. REASON FOR THE PROPOSED CHANGE

These changes are proposed for the following reasons:

Part A: The Unit 3 mechanical pressure and differential pressure indicating switches in the Reactor Protection System (RPS) and Emergency Core Cooling System (ECCS) are being replaced with an Analog Transmitter/Trip System (ATTS) similar to that previously installed on Unit 2. This modification also includes the replacement of power supplies and associated electrical cabling, breakers and fuses. The differences between the Unit 3 and Unit 2 instrumentation are described in the Safety Analysis section. The ATTS modification provides for continuous monitoring of critical parameters in additi to performing basic logic trip operations. The ATTS modification provides the following system upgrades:

- Reduces the functional tests and calibration frequencies for the primary sensors.
- Decreases the duration and complexity of required testing and calibration of the inputs for safety related parameters.
- Reduces testing and maintenance related scrams.
- Reduces the number of reportable events related to setpoint drift.

In addition, the revised design implements the diversity requirements associated with the Anticipated Transient Without Scram (ATWS) system required by 10 CFR 50.62.

These changes are described in the Description of the Proposed Change section and included as part of Items 7a-d, 8a-d, 9a-d, 11, 12a, 12c-f, 13a, 13c, 13e-r, 14a-b, 16a-b, 17a-e, 18a-i, 18k, 18m, 18o, and 19a-d.

Part B: The Units 1 and 3 Reactor Vessel Water Level Safety Limit is being revised to reflect the analytical limit provided by General Electric and the Level 1 Low Reactor Vessel Water Level setpoint is being revised to provide a more conservative limit.

The Reactor Vessel Water Level Safety Limit is provided by the Nuclear Steam System Supply vendor, General Electric, and is the design basis limit that should not be exceeded. A limiting safety system setting, such as the Level 1 low reactor vessel water level, is a setting on instrumentation which initiates the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represent margin. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

These changes are described in the Description of the Proposed Change section and included as part of Items 2, 3, 4, 5, 6, 12b, 13b, 13d, 13r, 20b-c, and 27b.

Part C: For Unit 2, instrument identifiers are being added or corrected to enhance the useability of the Technical Specifications. These changes are described in the Description of the Proposed Change section and included as part of Items 16a-b, 18a, and 18i.

Part D: For Unit 2, reactor high water level, Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) turbine steam line high flow, and drywell pressure instrumentation calibration frequencies and functional test descriptions are being revised to reflect current calculations and test methods. This instrumentation was upgraded as part of the installation of the ATTS, which was installed prior to Cycle 6 operation. These changes are described in the Description of the Proposed Change section and included as part of Items 18j, 18l, 18n, and 19d.

Part E: For Units 1, 2, and 3, the differential pressure instrumentation, which actuates the pressure suppression chamber-reactor building vacuum breakers, calibration frequency is being revised to reflect current Unit 2 and 3 calculations. The Unit 1 change is based on the similarity of this system and equipment between the three units. In addition, tables that specify the minimum number of instrument channels per trip system, function, trip level setting, actions required, remarks, functional test, and instrument check are being added in order to be consistent with the treatment of other electronic trip circuitry in the Technical Specifications. These changes are described in the Description of the Proposed Change section and included as part of Items 21, 22, 23, and 24.

Part F: The capitalization of terms used on the affected Units 1, 2, and 3 Technical Specification pages is being corrected in order to conform with the current TS Definitions section. Spelling and capitalization of other words is also being corrected on the same pages. These changes are described in the Description of the Proposed Change section and included as part of Items 1, 10a-b, 15, 16, 20a, 22, 25a-b, 26a-b, and 27a.

The Unit 3 ATTS modification, reactor vessel water level safety limit change, and Level 1 Low Reactor Vessel Water Level setpoint revision portions of this amendment are needed to support restart of BFN Unit 3. By letter dated December 23, 1993, TVA submitted the current schedule showing TS amendment need dates for support of Unit 3 restart. In order to support the restart schedule, TVA requests approval of the enclosed amendment by April 20, 1995.

III. SAFETY ANALYSIS

These changes are justified for the following reasons:

Part A: The Unit 3 Barton, Barksdale, Static-O-Ring, and Yarway instruments in the Reactor Protection System (RPS) and Emergency Core Cooling System (ECCS) are being replaced with an environmentally qualified Analog Transmitter/Trip System (ATTS). The ATTS modifications provide for continuous monitoring of critical parameters in addition to performing basic logic trip operations. This system, including the new instrumentation, was designed to meet or exceed the requirements in General Electric NEDO-21617-A, Analog Transmitter/Trip Unit System for Engineered Safeguard Sensor Trip Units. Overall, the ATTS provides the following system upgrades:

- Reduces the functional tests and calibration frequencies for the primary sensors.
- Decreases the duration and complexity or required testing and calibration of the inputs for safety related parameters.
- Reduces testing and maintenance related scrams.
- Reduces the number of reportable events related to setpoint drift.

The generic NRC approval of the ATTS is included in General Electric NEDO-21617-A, Analog Transmitter/Trip Unit System for Engineered Safeguard Sensor Trip Units. This Licensing Topical Report was approved by NRC letter, dated June 27, 1978, to the General Electric Company. This letter states that: "The staff does not intend to repeat its review of this topical report when it appears as a reference in specific license applications, except to assure that the report is applicable to the specific plants involved." In Section 5.4 of NEDO-21617-A, each applicant that uses this topical report as licensing basis was required to provide the following plant specific information to NRC:

Plant Specific Information Required

Section 5.4.1 - Specific Instrument Loops

Supply information for each instrument loop that will be converted to the analog sensor system as identified below:

1. Variable name
2. Part number of device being deleted
3. System involved
4. The engineered safeguards division
5. Model number and vendor of the transmitter or RTD

TVA Response

The requested information is provided below.

SECTION 5.4.1 - SPECIFIC INSTRUMENT LOOPS

VARIABLE NAME	DELETED INSTRUMENT VENDOR / NO. (LOCATION)	NEW TRANSMITTER TVA LOOP NO. VENDOR / NO. (LOCATION)	NEW ATU TVA LOOP NO. VENDOR/NO. (LOCATION)	DIVISION	SYSTEM INPUTS (LOCATION)
Main Steam Line High Flow	PDIS-1-13A Barton 278 (25-56A)	PDT-1-13A Rosemount 1153 (25-56A)	PDIS-1-13A Rosemount 710DU (9-83)	IA	PCIS (9-15)
	PDIS-1-13B Barton 278 (25-56A)	PDT-1-13B Rosemount 1153 (25-56A)	PDIS-1-13B Rosemount 710DU (9-84)	IB	PCIS (9-17)
	PDIS-1-13C Barton 278 (25-56A)	PDT-1-13C Rosemount 1153 (25-56A)	PDIS-1-13C Rosemount 710DU (9-85)	IIA	PCIS (9-15)
	PDIS-1-13D Barton 278 (25-56A)	PDT-1-13D Rosemount 1153 (25-56A)	PDIS-1-13D Rosemount 710DU (9-86)	IIB	PCIS (9-17)
Main Steam Line High Flow	PDIS-1-25A Barton 278 (25-56A)	PDT-1-25A Rosemount 1153 (25-56A)	PDIS-1-25A Rosemount 710DU 9-83)	IA	PCIS (9-15)
	PDIS-1-25B Barton 278 (25-56A)	PDT-1-25B Rosemount 1153 (25-56A)	PDIS-1-25B Rosemount 710DU (9-84)	IB	PCIS (9-17)
	PDIS-1-25C Barton 278 (25-56A)	PDT-1-25C Rosemount 1153 (25-56A)	PDIS-1-25C Rosemount 710DU (9-85)	IIA	PCIS (9-15)
	PDIS-1-25D Barton 278 (25-56A)	PDT-1-25D Rosemount 1153 (25-56A)	PDIS-1-25D Rosemount 710DU (9-86)	IB	PCIS (9-17)
Main Steam Line High Flow	PDIS-1-36A Barton 278 (25-56B)	PDT-1-36A Rosemount 1153 (25-56B)	PDIS-1-36A Rosemount 710DU (9-83)	IA	PCIS (9-15)
	PDIS-1-36B Barton 278 (25-56B)	PDT-1-36B Rosemount 1153 (25-56B)	PDIS-1-36B Rosemount 710DU (9-84)	IB	PCIS (9-17)
	PDIS-1-36C Barton 278 (25-56B)	PDT-1-36C Rosemount 1153 (25-56B)	PDIS-1-36C Rosemount 710DU (9-85)	IIA	PCIS (9-15)
	PDIS-1-36D Barton 278 (25-56B)	PDT-1-36D Rosemount 1153 (25-56B)	PDIS-1-36D Rosemount 710DU (9-86)	IIB	PCIS (9-17)

VARIABLE NAME	DELETED INSTRUMENT VENDOR / NO. (LOCATION)	NEW TRANSMITTER TVA LOOP NO. VENDOR / NO. (LOCATION)	NEW ATU TVA LOOP NO. VENDOR/NO. (LOCATION)	DIVISION	SYSTEM INPUTS (LOCATION)
Main Steam Line High Flow	PDIS-1-50A Barton 278 (25-56B)	PDT-1-50A Rosemount 1153 (25-56B)	PDIS-1-50A Rosemount 710DU (9-83)	IA	PCIS (9-15)
	PDIS-1-50B Barton 278 (25-56B)	PDT-1-50B Rosemount 1153 (25-56B)	PDIS-1-50B Rosemount 710DU (9-84)	IB	PCIS (9-17)
	PDIS-1-50C Barton 278 (25-56B)	PDT-1-50C Rosemount 1153 (25-56B)	PDIS-1-50C Rosemount 710DU (9-85)	IIA	PCIS (9-15)
	PDIS-1-50D Barton 278 (25-56B)	PDT-1-50D Rosemount 1153 (25-56B)	PDIS-1-50D Rosemount 710DU (9-86)	IIB	PCIS (9-17)
Main Steam Line Low Pressure	PS-1-72 Barksdale B2T-A12SS (25-112)	PT-1-72 Rosemount 1153 (25-112)	PIS-1-72 Rosemount 710DU (9-83)	IA	PCIS (9-15)
	PS-1-76 Barksdale B2T-A12SS (25-112)	PT-1-76 Rosemount 1153 (25-112)	PIS-1-76 Rosemount 710DU (9-84)	IB	PCIS (9-17)
Turbine First Stage Pressure Permissive	PS-1-81A Barksdale B2T-A12SS (25-111)	PT-1-81A Rosemount 1153 (25-111)	PIS-1-81A Rosemount 710DU (9-86)	IIB	RPS/RPT (9-17)
	PS-1-81B Barksdale B2T-A12SS (25-111)	PT-1-81B Rosemount 1153 (25-111)	PIS-1-81B Rosemount 710DU (9-85)	IIA	RPS/RPT (9-15)
Main Steam Line Low Pressure	PS-1-82 Barksdale B2T-A12SS (25-113C)	PT-1-82 Rosemount 1153 (25-113C)	PIS-1-82 Rosemount 710DU (9-85)	IIA	PCIS (9-15)
	PS-1-86 Barksdale B2T-A12SS (25-113C)	PT-1-86 Rosemount 1153 (25-113C)	PIS-1-86 Rosemount 710DU (9-86)	IIB	PCIS (9-17)

VARIABLE NAME	DELETED INSTRUMENT VENDOR / NO. (LOCATION)	NEW TRANSMITTER TVA LOOP NO. VENDOR / NO. (LOCATION)	NEW ATU TVA LOOP NO. VENDOR/NO. (LOCATION)	DIVISION	SYSTEM INPUTS (LOCATION)
Turbine First Stage Pressure Permissive	PS-1-91A Barksdale B2T-A12SS (25-110)	PT-1-91A Rosemount 1153 (25-110)	PIS-1-91A Rosemount 710DU (9-84)	IB	RPS/RPT (9-17)
	PS-1-91B Barksdale B2T-A12SS (25-110)	PT-1-91B Rosemount 1153 (25-110)	PIS-1-91B Rosemount 710DU (9-83)	IA	RPS/RPT (9-15)
Reactor High Pressure	PS-3-22A Barksdale B2T-A12SS (25-5A)	PT-3-22AA Tobar 32PA2212 (25-5A)	PIS-3-22AA Rosemount 710DU (9-83)	IA	RPS (9-15)
	PS-3-22B Barksdale B2T-A12SS (25-5A)	PT-3-22BB Tobar 32PA2212 (25-5A)	PIS-3-22BB Rosemount 710DU (9-84)	IB	RPS (9-17)
	PS-3-22C Barksdale B2T-A12SS (25-6A)	PT-3-22C Tobar 32PA2212 (25-6A)	PIS-3-22C Rosemount 710DU (9-85)	IIA	RPS (9-15)
	PS-3-22D Barksdale B2T-A12SS (25-6A)	PT-3-22D Tobar 32PA2212 (25-6A)	PIS-3-22D Rosemount 710DU (9-86)	IIB	RPS (9-17)
Reactor Low Water Level (Level 0)	LITS-3-52 Yarway 4418CE (25-51B)	LT-3-52 Rosemount 1153 (25-51B)	LIS-3-52 Rosemount 710DU (9-81)	I	Cont. Spray (9-32)
Reactor Low Low Water Level (Level 1)	LIS-3-56A Yarway 4418C (25-5A)	LT-3-56A Rosemount 1153 (25-5D)	LIS-3-56A Rosemount 710DU (9-83)	IA	PCIS (9-15)
	LIS-3-56B Yarway 4418C (25-5A)	LT-3-56B Rosemount 1153 (25-5D)	LIS-3-56B Rosemount 710DU (9-84)	IB	PCIS (9-17)
	LIS-3-56C Yarway 4418C (25-6A)	LT-3-56C Rosemount 1153 (25-6D)	LIS-3-56C Rosemount 710DU (9-85)	IIA	PCIS (9-15)
	LIS-3-56D Yarway 4418C (25-6A)	LT-3-56D Rosemount 1153 (25-6D)	LIS-3-56D Rosemount 710DU (9-86)	IIB	PCIS (9-17)

VARIABLE NAME	DELETED INSTRUMENT VENDOR / NO. (LOCATION)	NEW TRANSMITTER TVA LOOP NO. VENDOR / NO. (LOCATION)	NEW ATU TVA LOOP NO. VENDOR/NO. (LOCATION)	DIVISION	SYSTEM INPUTS (LOCATION)
Reactor Low Water Level (Level 2)	LIS-3-58A Yarway 4418C (25-5A)	LT-3-58A Rosemount 1153 (25-5D)	LIS-3-58A GE 184C5988G (9-81)	I	HPCI,RCIC (9-32)
			LS-3-58A GE 184C5988G (9-81)	I	CSS, LPCI (9-32) ADS (9-30)
			LS-3-58A1 GE 184C5988G (9-81)	I	ATWS (ARI/RPT) (25-416)
	LITS-3-58B Yarway 4418C (25-5A)	LT-3-58B Rosemount 1153 (25-5D)	LIS-3-58B GE 184C5988G (9-81)	I	HPCI, RCIC (9-32)
			LS-3-58B GE 184C5988G (9-81)	I	CSS, LPCI (9-32) ADS (9-30)
			LS-3-58B1 GE 184C5988G (9-81)	I	ATWS (ARI/RPT) (25-416)
	LIS-3-58C Yarway 4418C (25-6A)	LT-3-58C Rosemount 1153 (25-6D)	LIS-3-58C GE 184C5988G (9-82)	II	HPCI (9-39) RCIC (9-33)
			LS-3-58C GE 184C5988G (9-82)	II	CSS, LPCI (9-33) ADS (9-33)
			LS-3-58C1 GE 184C5988G (9-82)	II	ATWS (ARI/RPT) (25-613)
	LITS-3-58D Yarway 4418C (25-6A)	LT-3-58D Rosemount 1153 (25-6D)	LIS-3-58D GE 184C5988G (9-82)	II	HPCI (9-39) RCIC (9-33)
			LS-3-58D GE 184C5988G (9-82)	II	CSS, LPCI (9-33) ADS (9-33)
			LS-3-58D1 GE 184C5988G (9-82)	II	ATWS (ARI/RPT) (25-613)

VARIABLE NAME	DELETED INSTRUMENT VENDOR / NO. (LOCATION)	NEW TRANSMITTER TVA LOOP NO. VENDOR / NO. (LOCATION)	NEW ATU TVA LOOP NO. VENDOR/NO. (LOCATION)	DIVISION	SYSTEM INPUTS (LOCATION)
Reactor Low Water Level (Level 0)	LITS-3-62 Yarway 4418CE (25-52B)	LT-3-62A Rosemount 1153 (25-52B)	LIS-3-62A Rosemount 710DU (9-82)	II	Cont. Spray (9-33)
Reactor Pressure	PS-3-74A Barksdale B2T-M12SS (25-5A)	PT-3-74A Tobar 32PA1212 (25-5A)	PIS-3-74A Rosemount 710DU (9-81)	I	CSS, LPCI (9-32)
			PS-3-74A Rosemount 710DU (9-81)	I	LPCI (9-32)
	PS-3-74B Barton 288 (25-6A)	PT-3-74B Tobar 32PA1212 (25-6A)	PIS-3-74B Rosemount 710DU (9-82)	II	CSS, LPCI (9-33)
			PS-3-74B Rosemount 710DU (9-82)	II	LPCI (9-33)
Reactor Low Water Level (Level 3)	LIS-3-184 Yarway 4418C (25-5B)	LT-3-184 Rosemount 1153 (25-5D)	LIS-3-184 Rosemount 710DU (9-81)	I	ADS (9-30)
	LIS-3-185 Yarway 4418C (25-6B)	LT-3-185 Rosemount 1153 (25-6D)	LIS-3-185 Rosemount 710DU (9-82)	II	ADS (9-33)
Reactor Low Water Level (Level 3)	LIS-3-203A Barton 288A (25-5-1)	LT-3-203A Rosemount 1153 (25-5C)	LIS-3-203A Rosemount 710DU (9-83)	IA	RPS, PCIS (9-15)
	LIS-3-203B Barton 288A (25-5-1)	LT-3-203B Rosemount 1153 (25-5C)	LIS-3-203B Rosemount 710DU (9-84)	IB	RPS, PCIS (9-17)
	LIS-3-203C Barton 288A (25-6-1)	LT-3-203C Rosemount 1153 (25-6C)	LIS-3-203C Rosemount 710DU (9-85)	IIA	RPS, PCIS (9-15)
	LIS-3-203D Barton 288A (25-6-1)	LT-3-203D Rosemount 1153 (25-6C)	LIS-3-203D Rosemount 710DU (9-86)	IIB	RPS, PCIS (9-17)

VARIABLE NAME	DELETED INSTRUMENT VENDOR / NO. (LOCATION)	NEW TRANSMITTER TVA LOOP NO. VENDOR / NO. (LOCATION)	NEW ATU TVA LOOP NO. VENDOR/NO. (LOCATION)	DIVISION	SYSTEM INPUTS (LOCATION)
Reactor High Pressure	PS-3-204A Static-O-Ring 9N-AA45 (25-5-1)	PT-3-204A Rosemount 1153 (25-5C)	PIS-3-204A GE 184C5988G (9-81)	I	ATWS (ARI/RPT) (25-416)
	PS-3-204B Static-O-Ring 9N-AA45 (25-5-1)	PT-3-204B Rosemount 1153 (25-5C)	PIS-3-204B GE 184C5988G (9-81)	I	ATWS (ARI/RPT) (25-416)
	PS-3-204C Static-O-Ring 9N-AA45 (25-6-1)	PT-3-204C Rosemount 1153 (25-6C)	PIS-3-204C GE 184C5988G (9-82)	II	ATWS (ARI/RPT) (25-613)
	PS-3-204D Static-O-Ring 9N-AA45 (25-6-1)	PT-3-204D Rosemount 1153 (25-6C)	PIS-3-204D GE 184C5988G (9-82)	II	ATWS (ARI/RPT) (25-613)
Reactor High water Level (Level 8)	LIS-3-208A Barton 288A (25-5-1)	LT-3-208A Rosemount 1153 (25-5C)	LIS-3-208A Rosemount 710DU (9-81)	I	RCIC (25-31)
	LIS-3-208B Barton 288A (25-5-1)	LT-3-208B Rosemount 1153 (25-5C)	LIS-3-208B Rosemount 710DU (9-82)	II	HPCI (9-39)
	LIS-3-208C Barton 288A (25-6-1)	LT-3-208C Rosemount 1153 (25-6C)	LIS-3-208C Rosemount 710DU (9-81)	I	RCIC (25-31)
	LIS-3-208D Barton 288A (25-6-1)	LT-3-208D Rosemount 1153 (25-6C)	LIS-3-208D Rosemount 710DU (9-82)	II	HPCI (9-39)
Primary Containment High Pressure	PS-64-56A Static-O-Ring 12N-AA4 (25-5B)	PT-64-56A Rosemount 1153 (25-5A)	PIS-64-56A Rosemount 710DU (9-83)	IA	RPS/PCIS (9-15)
	PS-64-56B Static-O-Ring 12N-AA4 (25-5B)	PT-64-56B Rosemount 1153 (25-5A)	PIS-64-56B Rosemount 710DU (9-84)	IB	RPS/PCIS (9-17)
	PS-64-56C Static-O-Ring 12N-AA4 (25-6B)	PT-64-56C Rosemount 1153 (25-6B)	PIS-64-56C Rosemount 710DU (9-85)	IIA	RPS/PCIS (9-15)
	PS-64-56D Static-O-Ring 12N-AA4 (25-6B)	PT-64-56D Rosemount 1153 (25-6B)	PIS-64-56D Rosemount 710DU (9-86)	IIB	RPS/PCIS (9-17)

VARIABLE NAME	DELETED INSTRUMENT VENDOR / NO. (LOCATION)	NEW TRANSMITTER TVA LOOP NO. VENDOR / NO. (LOCATION)	NEW ATU TVA LOOP NO. VENDOR/NO. (LOCATION)	DIVISION	SYSTEM INPUTS (LOCATION)
Primary Containment High Pressure	PS-64-57A Static-O-Ring 12N-AA4 (25-6B)	PT-64-57A Rosemount 1153 (25-6B)	PIS-64-57A Rosemount 710DU (9-82)	II	ADS (9-33)
	PS-64-57B Static-O-Ring 12N-AA4 (25-5B)	PT-64-57B Rosemount 1153 (25-5A)	PIS-64-57B Rosemount 710DU (9-81)	I	ADS (9-30)
	PS-64-57C Static-O-Ring 12N-AA4 (25-6B)	PT-64-57C Rosemount 1153 (25-6A)	PIS-64-57C Rosemount 710DU (9-82)	II	ADS (9-33)
	PS-64-57D Static-O-Ring 12N-AA4 (25-5B)	PT-64-57D Rosemount 1153 (25-5A)	PIS-64-57D Rosemount 710DU (9-81)	I	ADS (9-30)
Primary Containment Low Pressure	PS-64-58A Static-O-Ring 12N-AA4 (25-6B)	PT-64-58A Rosemount 1153 (25-6B)	PIS-64-58A Rosemount 710DU (9-82)	II	CSS, HPCI, LPCI (9-33)
	PS-64-58B Static-O-Ring 12N-AA4 (25-5B)	PT-64-58B Rosemount 1153 (25-5B)	PIS-64-58B Rosemount 710DU (9-81)	I	CSS, HPCI, LPCI (9-32)
	PS-64-58C Static-O-Ring 12N-AA4 (25-6B)	PT-64-58C Rosemount 1153 (25-6B)	PIS-64-58C Rosemount 710DU (9-82)	II	CSS, HPCI, LPCI (9-33)
	PS-64-58D Static-O-Ring 12N-AA4 (25-5B)	PT-64-58D Rosemount 1153 (25-5B)	PIS-64-58D Rosemount 710DU (9-81)	I	CSS, HPCI, LPCI (9-32)
Primary Containment Low Pressure	PS-64-58E Static-O-Ring 12N-AA4 (25-5B)	PT-64-58E Rosemount 1153 (25-5B)	PIS-64-58E Rosemount 710DU (9-81)	I	Cont. Spray (9-32)
	PS-64-58F Static-O-Ring 12N-AA4 (25-6B)	PT-64-58F Rosemount 1153 (25-6A)	PIS-64-58F Rosemount 710DU (9-82)	II	Cont. Spray (9-33)
	PS-64-58G Static-O-Ring 12N-AA4 (25-5B)	PT-64-58G Rosemount 1153 (25-5B)	PIS-64-58G Rosemount 710DU (9-81)	I	Cont. Spray (9-32)
	PS-64-58H Static-O-Ring 12N-AA4 (25-6B)	PT-64-58H Rosemount 1153 (25-6A)	PIS-64-58H Rosemount 710DU (9-82)	II	Cont. Spray (9-33)

VARIABLE NAME	DELETED INSTRUMENT VENDOR / NO. (LOCATION)	NEW TRANSMITTER TVA LOOP NO. VENDOR / NO. (LOCATION)	NEW ATU TVA LOOP NO. VENDOR/NO. (LOCATION)	DIVISION	SYSTEM INPUTS (LOCATION)
Reactor Low Pressure	PS-68-95 Barksdale B2T-M12AA (25-51B)	PT-68-95 Tobar 32PA1212 (25-51B)	PIS-68-95 Rosemount 710DU (9-81)	I	CSS, LPCI (9-32)
			PS-68-95 Rosemount 710DU (9-81)	I	LPCI (9-32)
	PS-68-96 Barksdale B2T-M12AA (25-52B)	PT-68-96 Tobar 32PA1212 (25-52B)	PIS-68-96 Rosemount 710DU (9-82)	II	CSS, LPCI (9-33)
			PS-68-96 Rosemount 710DU (9-82)	II	LPCI (9-33)
RCIC Steam Line High Flow	PDIS-71-1A Barton 288 (25-7A)	PDT-71-1A Rosemount 1153 (25-7A)	PDIS-71-1A Rosemount 710DU (9-81)	I	RCIC (25-31)
	PDIS-71-1B Barton 288 (25-7A)	PDT-71-1B Rosemount 1153 (25-7A)	PDIS-71-1B Rosemount 710DU (9-82)	II	RCIC (9-33)
HPCI Steam Line High Flow	PDIS-73-1A Barton 288A (25-7B)	PDT-73-1A Rosemount 1153 (25-7B)	PDIS-73-1A Rosemount 710DU (9-81)	I	HPCI (9-33)
	PDIS-73-1B Barton 288A (25-7B)	PDT-73-1B Rosemount 1153 (25-7B)	PDIS-73-1B Rosemount 710DU (9-82)	II	HPCI (9-39)

Plant Specific Information Required

Section 5.4.2 - Trip Unit Cabinet

Supply information for each trip unit cabinet as identified below:

1. Cabinet layout showing location areas of the power supplies, trip relays, and trip units.

2. Division of which the cabinet is assigned.

3. Layout of each card file in the trip unit cabinet showing the trip variable for each card file slot.

TVA Response

1. The cabinet layout and the location of trip relays and units is shown in Figure 1 and is listed in the Section 5.4.1 table. Panels that begin with the prefix 25-?? are local (Physically located near the instrument). Panels that begin with the prefix 9-?? are located in the auxiliary instrument room (No pipe from the nuclear system or the primary containment penetrates the Auxiliary Instrument Room).

2. The division assigned to each cabinet is shown in Figure 2.

3. The layout of each card file in the trip unit cabinet and the trip variable for each card file slot is shown in Figure 2.

FIGURE 1
TRIP CABINET ASSEMBLY

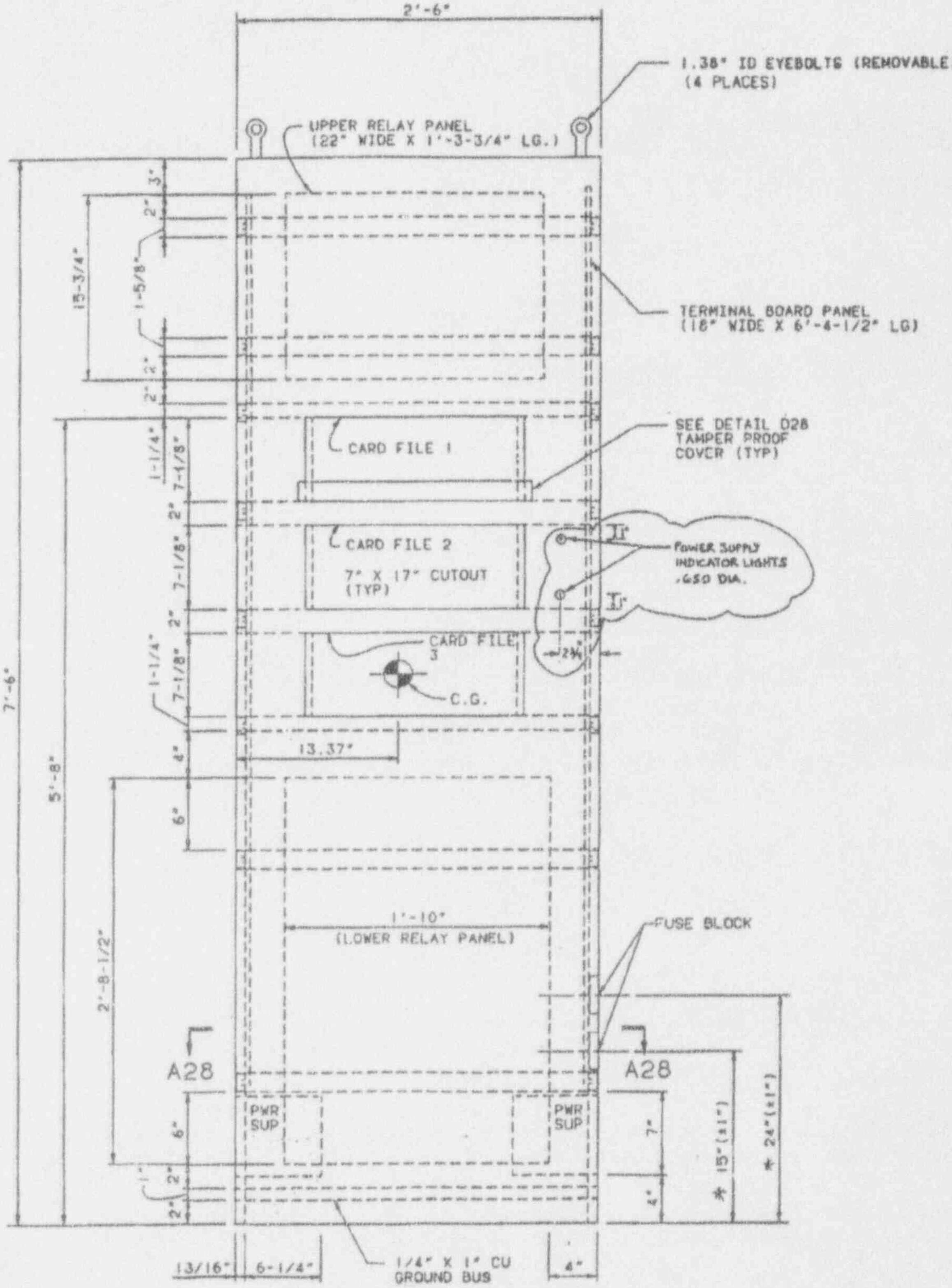


FIGURE 2
TRIP CABINET ASSEMBLY

		ECCS DIV I PANEL 9-81	ECCS DIV II PANEL 9-82	RPS A1 PANEL 9-83	RPS B1 PANEL 9-84	RPS A2 PANEL 9-85	RPS B2 PANEL 9-86
24V DC PWR SUPPLY		PX-71-60-1 & 1A	PX-71-60-2 & 2A	PX-99-A1 & A1A	PX-99-B1 & B1A	PX-99-A2 & A2A	PX-99-B2 & B2A
PWR SUPPLY IND LIGHTS		IL-71-60-1 IL-71-60-1A	IL-71-60-2 IL-71-60-2A	IL-99-A1 IL-99-A1A	IL-99-B1 IL-99-B1A	IL-99-A2 IL-99-A2A	IL-99-B2 IL-99-B2A
ANNUNCIATOR		XA-71-60 (SEE NOTE 12)		XA-99-1 (SEE NOTE 12)			
CARD FILE 1	1	LIS-3-52	LIS-3-62A	PDIS-1-13A	PDIS-1-13B	PDIS-1-13C	PDIS-1-13D
	2	LIS-3-58B	LIS-3-58D	FUTURE	FUTURE	FUTURE	FUTURE
	3	LS-3-58B	LS-3-58D	PDIS-1-25A	PDIS-1-25B	PDIS-1-25C	PDIS-1-25D
	4	LS-3-58B1	LS-3-58D1	FUTURE	FUTURE	FUTURE	FUTURE
	5	FUTURE	FUTURE	PDIS-1-36A	PDIS-1-36B	PDIS-1-36C	PDIS-1-36D
	6	LIS-3-58A	LIS-3-58C	FUTURE	FUTURE*	FUTURE	FUTURE
	7	LS-3-58A	LS-3-58C	PDIS-1-50A	PDIS-1-50B	PDIS-1-50C	PDIS-1-50D
	8	LS-3-58A1	LS-3-58C1	FUTURE	FUTURE	FUTURE	FUTURE
	9	FUTURE	FUTURE	PIS-1-72	PIS-1-76	PIS-1-82	PIS-1-86
	10	LIS-3-184	LIS-3-185	FUTURE	FUTURE	FUTURE	FUTURE
	11	PIS-64-57B	PIS-64-57A	PIS-1-91B	PIS-1-91A	PIS-1-81B	PIS-1-81A
	12	PIS 64-57D	PIS-64-57C	FUTURE	FUTURE	FUTURE	FUTURE
	13 14	XIS-71-60-1	XIS-71-60-2	XIS-99-1	XIS-99-1B	XIS-99-2	XIS-99-2B
CARD FILE 2	1	PIS-64-58B	PIS-64-58A	PIS-3-22AA	PIS-3-22BB	PIS-3-22C	PIS-3-22D
	2	FUTURE	FUTURE	LIS-3-56A	LIS-3-56B	LIS-3-56C	LIS-3-56D
	3	PIS-64-58D	PIS-64-58C	LIS-3-203A	LIS-3-203B	LIS-3-203C	LIS-3-203D
	4	FUTURE	FUTURE	FUTURE	FUTURE	FUTURE	FUTURE
	5	PIS-64-58E	PIS-64-58F				
	6	FUTURE	FUTURE				
	7	PIS-64-58G	PIS-64-58H	PIS-64-56A	PIS-64-56B	PIS-64-56C	PIS-64-56D
	8	FUTURE	FUTURE	FUTURE	FUTURE	FUTURE	FUTURE
	9	PIS-68-95	PIS-68-96				
	10	PS-68-95	PS-68-96				
	11	PDIS-71-1A	PDIS-71-1B				
	12	PDIS-73-1A	PDIS-73-1B				
	13 14	XIS-71-60-1A	XIS-71-60-2A	XIS-99-1A	XIS-99-1BB	XIS-99-2A	XIS-99-2BB
CARD FILE 3	1	LIS-3-208A	LIS-3-208B	FUTURE	FUTURE	FUTURE	FUTURE
	2	LIS-3-208C	LIS-3-208D				
	3	FUTURE	FUTURE				
	4	FUTURE	FUTURE				
	5	FUTURE	FUTURE				
	6	PIS-3-204A	PIS-3-204C				
	7	PIS-3-204B	PIS-3-204D				
	8	FUTURE	FUTURE				
	9	FUTURE	FUTURE				
	10	PIS-3-74A	PIS-3-74B				
	11	PS-3-74A	PS-3-74B				
	12	FUTURE	FUTURE				
	13 14	XIS-73-91	XIS-73-92				

Plant Specific Information Required

Section 5.4.3 - Environmental Interface

The environment at each location where the retrofit hardware will be located must be compared to the maximum environment as stated in the topical report for the following factors:

1. Normal operation and post-accident temperature and humidity.

TVA Response

1. The response to Item 1, regarding the normal operation and post-accident temperature and humidity, is provided in the attached table.

SECTION 5.4.3, ITEM 1 - ENVIRONMENTAL INTERFACE TEMPERATURE AND HUMIDITY

Transmitter Number[1]	Maximum Normal Temperature	Maximum Normal Humidity	Maximum Post-Accident Temperature	Maximum Post-Accident Humidity	Maximum Qualified Temperature	Maximum Qualified Humidity
PDT-1-13 A, B, C, D	95°F	80%	110°F	100%	415°F	100%
PDT-1-25 A, B, C, D	95°F	80%	110°F	100%	415°F	100%
PDT-1-36 A, B, C, D	95°F	80%	110°F	100%	415°F	100%
PDT-1-50 A, B, C, D	95°F	80%	110°F	100%	415°F	100%
PT-1-72, 76, 82, 86[2]	105°F	80%	105°F	80%	318°F	100%
PT-1-81A&B PT-1-91A&B[2]	105°F	80%	105°F	80%	318°F	100%
PT-3-22[2] A, B, C, D	90°F	80%	100°F	90%	420°F	100%
LT-3-52 LT-3-62A	90°F	80%	195°F	100%	415°F	100%

Transmitter Number[1]	Maximum Normal Temperature	Maximum Normal Humidity	Maximum Post-Accident Temperature	Maximum Post-Accident Humidity	Maximum Qualified Temperature	Maximum Qualified Humidity
LT-3-56 A, B, C, D	90°F	80%	185°F	100%	318°F	100%
LT-3-58 A, B, C, D	90°F	80%	185°F	100%	318°F	100%
PT-3-74A&B	90°F	80%	185°F	100%	420°F	100%
LT-3-184 LT-3-185	90°F	80%	125°F	80%	318°F	100%
LT-3-203 A, B, C, D	90°F	80%	185°F	100%	318°F	100%
PT-3-204[2] A, B, C, D	90°F	80%	100°F	90%	318°F	100%
LT-3-208 A, B, C, D	90°F	80%	195°F	100%	318°F	100%
PT-64-20 PT-64-21	90°F	80%	126°F	90%	318°F	100%
PT-64-56 A, B, C, D	90°F	80%	125°F	100%	318°F	100%
PT-64-57 A, B, C, D	90°F	80%	125°F	90%	318°F	100%

Transmitter Number[1]	Maximum Normal Temperature	Maximum Normal Humidity	Maximum Post-Accident Temperature	Maximum Post-Accident Humidity	Maximum Qualified Temperature	Maximum Qualified Humidity
PT-64-58 A, B, C, D	90°F	80%	125°F	90%	318°F	100%
PT-64-58 E, F, G, H	90°F	80%	125°F	90%	318°F	100%
PT-68-95 PT-68-96	90°F	80%	185°F	100%	420°F	100%
PdT-71-1A PdT-71-1B	95°F	80%	136°F	90%	415°F	100%
PdT-73-1A PdT-73-1b	95°F	80%	165°F	100%	415°F	100%

Footnote: [1] - Worst case radiation exposure (40 year plus 100 day post-accident) is 1.37×10^7 rads. Minimum qualification for the transmitter models to be installed is 2.62×10^7 rads.

[2] - This equipment is not within the scope of the 10 CFR 50.49 program. The specified qualified temperature/humidity values are from the test report of this manufacturer/model.

Plant Specific Information Required

Section 5.4.3 - Environmental Interface

The environment at each location where the retrofit hardware will be located must be compared to the maximum environment as stated in the topical report for the following factors:

2. Comparison of the floor seismic response spectra of the cabinet mounting location for the specific plant to seismic test envelope that the cabinet was tested to.

TVA Response

2. Figure 3 provides the comparison of the floor seismic response spectra of the cabinet mounting location to the seismic test envelope that the cabinet was tested to.

Plant Specific Information Required

Section 5.4.3 - Environmental Interface

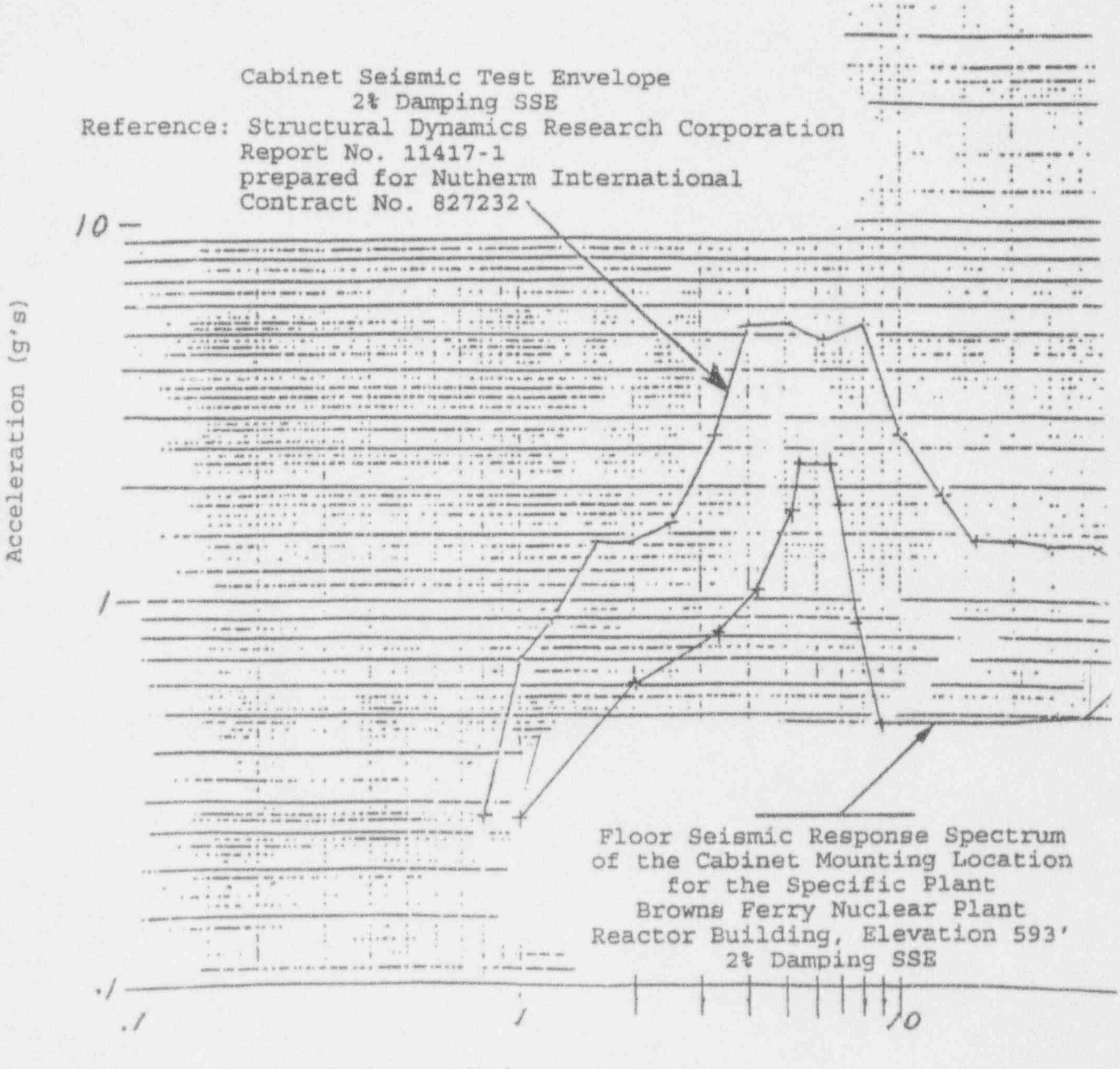
The environment at each location where the retrofit hardware will be located must be compared to the maximum environment as stated in the topical report for the following factors:

3. If the trip unit cabinets are not located in the preferred location as per paragraph 5.1.4, provide justification for the alternate selected location.

TVA Response

3. The preferred location recommended by NEDO-21617-A, for the trip unit cabinets is the auxiliary room or control room. The Unit 3 equipment is located in the preferred location.

FIGURE 3
ENVIRONMENTAL INTERFACE - SEISMIC RESPONSE



Plant Specific Information Required

Section 5.4.4 - Specific Plant Interconnections

An interconnection diagram which shows the interconnections between the existing logic cabinets and instrument cabinets and the new trip unit cabinets is to be provided the NRC. The content of the information is to be similar to the information shown of Figures 5-3, 5-4 and 5-5 as applicable. The detail of interconnection shown of the retrofit elementary and interconnection block diagram should be sufficient.

TVA Response

The BFN plant specific interconnection diagram is represented by NEDO-21617-A, Figure 5-5 .

Plant Specific Information Required

Section 5.4.5 - Field Calibration Rack

The design and operational information on the "Field Calibration Rack" is to be supplied to the NRC if such a device is purchased and used for transmitter calibration.

TVA Response

BFN did not purchase and does not use a "Field Calibration Rack."

In general, the instrumentation that will be installed as part of the ATTS on Unit 3 is the same instrumentation previously reviewed by NRC and installed on Unit 2, with the following exception:

The transmitters for the Reactor High Water Level instrument channel (Equipment identifiers LT-3-208 A-D, which include the level indicating switches LIS-3-208 A-D that appear in the Technical Specifications) are being replaced with environmentally qualified Rosemount 1153 transmitters. Gould transmitters were used in this application on Unit 2 due to the unavailability of the Rosemount transmitters at the time of the Unit 2 modifications.

As previously committed in TVA's March 15, 1993, response to NRC Bulletin 90-01, Supplement 1, TVA will replace or refurbish the Rosemount Model 1153 Series B and D and Model 1154 transmitters in safety related or ATWS applications prior to the restart of Unit 3.

The specific Technical Specification changes associated with the installation of the ATTS on Unit 3 (unless otherwise annotated) are discussed below. Also, unless otherwise noted, these changes make the Unit 3 Technical Specifications consistent with changes previously approved to the Unit 2 Technical Specifications.

1. Table 3.1.A, Reactor Protection System (SCRAM) Instrumentation Requirements, is being revised to add instrument identifiers for the equipment installed as part of the ATTS modifications. These instrument identifiers enhance the useability of the Technical Specifications.

2. Table 4.1.A, Reactor Protection System (Scram) Instrumentation Functional Tests Minimum Functional Test Frequencies for Safety Instr. and Control Circuits, is being revised to add or revise instrument identifiers, change the functional testing group designator for the instruments, and update the footnotes associated with the functional test type and minimum test frequency.

   The addition of, or revision to, the instrument identifiers reflects the equipment installed as part of the ATTS modifications. These instrument identifiers enhance the useability of the Technical Specifications.

   The functional testing group designator for the instrumentation being installed as part of the ATTS modifications is being changed to "B". As described in the Bases for this section, the Group B designator is for analog devices, coupled with bistable trips, that provide a scram function.

Note 7 is being added to the Functional Test column to define the new functional test requirements for the instrumentation being installed as part of the ATTS modifications. Note 7 states that the functional test consists of the injection of a simulated signal into the electronic trip circuitry, in place of the sensor signal, to verify the operability of the trip and alarm functions. The inclusion of Note 7 is consistent with the definition for the Channel Functional Test of Analog/Digital Channel, which is contained in Technical Specification Definition 1.0.V.12a.

Note 1 is being deleted from the Minimum Frequency column in order to clarify the Technical Specifications. Note 1 states that the minimum frequency for the indicated tests shall initially be once per month. Since the functional test frequency for the instrumentation being installed as part of the ATTS modifications is once per month, the inclusion of Note 1 is redundant.

3. Table 4.1.B, Reactor Protection System (Scram) Instrumentation Calibration Minimum Calibration Frequencies for Reactor Protection Instrument Channels, is being revised to add or revise instrument identifiers, change the functional testing group designator for the instruments, and change the minimum calibration frequencies.

   The addition of, or revision to, the instrument identifiers reflects the equipment installed as part of the ATTS modifications. These instrument identifiers enhance the useability of the Technical Specifications.

   The functional testing group designator for the instrumentation being installed as part of the ATTS modifications is being changed to "B". As described in the Bases for this section, the Group B designator is for analog devices, coupled with bistable trips, that provide a scram function.

In general, surveillance frequencies are based on industry accepted practices and engineering judgement. Consideration is given to the conditions required to perform a given test, the ease of performing the test, and the likelihood of a change in the system/component status during the performance of the test. Instrumentation calibration frequencies consist of an optimum selection of time versus drift. Setpoint scaling calculations are performed to provide assurance that there is adequate margin between the required trip setpoint and the limiting safety system settings to account for inaccuracies in the instrument loop.

The minimum calibration frequency for the High Drywell Pressure, Reactor Low Water Level, and Turbine First Stage Pressure Permissive are being changed to once per 18 months. The extension of the minimum calibration frequency to once per 18 months reflects the high reliability of the analog instrumentation systems.

The minimum calibration frequency for the High Reactor Pressure instrument channel is being revised to once per 6 months. This reflects the installation of instrument loops that contain transmitters manufactured by Tobar, Incorporated. These instruments only permit the extension to a six month calibration frequency.

These calibration frequencies are in accordance with the Unit 3 specific setpoint and scaling calculations for the ATTS instrumentation. Note 9 is being added to the Minimum Frequency column to specify the calibration methods required for this new instrumentation.

4. A paragraph is being added to Unit 3 Bases 3.1 in order to describe the Reactor Protection System (RPS) power supply. This change describes the ability of the RPS to tolerate a single failure of a non-class 1E power supply and makes the Unit 3 Technical Specifications consistent with changes previously performed on the Units 1 and 2 Technical Specifications.

5. Table 3.2.A, Primary Containment and Reactor Building Isolation Instrumentation, is being revised to add or revise instrument identifiers and to correct notes in the Remarks column.

   The addition of, or revision to, the instrument identifiers reflects the equipment installed as part of the ATTS modifications. These instrument identifiers enhance the useability of the Technical Specifications.

   The listing of the Primary Containment Isolation System valve groups, which are initiated by the reactor low water level, is being deleted. This change makes this entry consistent with the other entries in this table, which do not list the specific valve groups initiated by a trip function.

6. Table 3.2.B, Instrumentation that Initiates or Controls the Core and Containment Cooling Systems, is being revised to add or revise instrument identifiers. The addition of, or revision to, the instrument identifiers reflects the equipment installed as part of the ATTS modifications. These instrument identifiers enhance the useability of the Technical Specifications.

7. Table 3.2.F, Surveillance Instrumentation, is being revised to correct instrument identifiers and to change the indicated range of the reactor pressure instruments.

   The revision to the instrument identifiers reflects the equipment installed as part of the ATTS modifications. These instrument identifiers enhance the useability of the Technical Specifications.

   The revision to the reactor pressure indication range reflects the newly installed equipment. The newly installed range, 0 - 1200 psig, includes the full range of pressures for which operator actions would be initiated during accident conditions. This range was approved for BFN in the NRC's May 10, 1991 Safety Evaluation of Emergency Response Capability and Conformance to Regulatory Guide 1.97, Revision 3.

8. Table 3.2.L, Anticipated Transient Without Scram (ATWS) - Recirculation Pump Test (RPT) Surveillance Instrumentation, is being revised to add instrument identifiers.

   The addition of the instrument identifiers reflects the equipment installed as part of the ATTS modifications. These instrument identifiers enhance the useability of the Technical Specifications (NOTE: Similar changes are also being proposed for the Unit 2 Technical Specifications).

9. Table 4.2.A, Surveillance Requirements for Primary Containment and Reactor Building Isolation Instrumentation, is being revised to correct instrument identifiers, change the functional testing description for the instruments, and change the minimum calibration frequencies.

   The correction of the instrument identifiers reflects the equipment installed as part of the ATTS modifications. These instrument identifiers enhance the useability of the Technical Specifications.

   The addition of Note 28 in the Functional Test column describes the type of function test performed on the ATTS instrumentation.

   The minimum calibration frequencies for the Reactor Low Water Level, High Drywell Pressure, Low Pressure Main Steam Line, and High Flow Main Steam Line instrument channels are being revised in accordance with the Unit 3 specific setpoint and scaling calculations for the ATTS instrumentation. The addition of Note 29 in the Calibration Frequency column describes the type of calibration performed on the ATTS instrumentation.

10. Table 4.2.B, Surveillance Requirements for Instrumentation that Initiate or Control the CSCS, is being revised to add or correct instrument identifiers, change the functional testing description for the instruments, and change the minimum calibration frequencies.

The addition or correction of the instrument identifiers reflects the equipment installed as part of the ATTS modifications. These instrument identifiers enhance the useability of the Technical Specifications (NOTE: Instrument identifiers are also being added or corrected in the Unit 2 Technical Specifications for a Reactor Low Water Level and Reactor High Water Level instrument channel).

The addition of Note 28 in the Functional Test column describes the type of function test performed on the ATTS instrumentation (NOTE: A similar note is also being proposed for the Unit 2 Technical Specifications for the Reactor High Water Level and RCIC and HPCI Turbine Steam Line High Flow instrument channels).

The minimum calibration frequencies for the Reactor Low Water Level, Drywell High Pressure, Reactor Low Pressure, Reactor High Water Level, and RCIC and HPCI Turbine Steam Line High Flow instrument channels are being revised in accordance with the Unit 3 specific setpoint and scaling calculations for the ATTS instrumentation (NOTE: A similar note is also being proposed for the Unit 2 Technical Specifications for the Reactor High Water Level and RCIC and HPCI Turbine Steam Line High Flow instrument channels).

The addition of Note 29 in the Calibration Frequency column describes the type of calibration performed on the ATTS instrumentation (NOTE: A similar note is also being proposed for the Unit 2 Technical Specifications for the Reactor High Water Level and RCIC and HPCI Turbine Steam Line High Flow instrument channels).

The minimum calibration frequency for the Reactor Low Pressure instrument channel is being revised to once per 6 months in accordance with the Unit 3 specific setpoint and scaling calculations for the ATTS instrumentation. This reflects the installation of instrument loops that contain transmitters manufactured by Tobar, Incorporated. These instruments only permit the extension to a six month calibration frequency.

11. Table 4.2.F, Minimum Test and Calibration Frequency for Surveillance Instrumentation, is being revised to add or correct instrument identifiers and change the minimum calibration frequencies.

    The addition or correction of the instrument identifiers reflects the equipment installed as part of the ATTS modifications. These instrument identifiers enhance the useability of the Technical Specifications.

    The minimum calibration frequencies for the Reactor Water Level and Drywell Pressure instrument channels are being revised in accordance with the Unit 3 specific setpoint and scaling calculations for the ATTS instrumentation (NOTE: A similar change is also being proposed for the Unit 2 Technical Specifications).

<u>Part B</u>: The Units 1 and 3 Reactor Vessel Water Level Safety Limit and the Level 1 Low Reactor Vessel Water Level setpoint are being revised.

1. The analytical reactor vessel water level safety limit determined by General Electric (the NSSS supplier) calculations has always been greater than or equal to 372.5 inches above vessel zero. Lowering the current safety limit (378 inches) to match the analytical limit is supported by calculation and will make the Units 1 and 3 value consistent with the current Unit 2 setting.

2. The Level 1 low reactor vessel water level instruments actuate the Core Spray and Low Pressure Coolant Injections systems in order to mitigate the consequences of a loss of coolant accident. They also isolate the main steam lines to reduce inventory loss.

During the process of generating setpoint and accuracy calculations for plant parameters in support of Unit 2, a determination was made that the Level 1 Reactor Vessel Level 1 Low Water Level setpoint was not conservative based on the current calculation methodology, which is based on Regulatory Guide 1.105, Instrument Setpoints for Safety Related Systems. Regulatory Guide 1.105 endorses Instrument Society of America (ISA) Standard ISA-S67.04 - 1982, Setpoints for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants, as an acceptable method for ensuring that setpoints stay within technical specification limits.

As discussed above, the analytical safety limit for the Reactor Vessel Water Level Safety Limit is 372.5 inches above vessel zero. This limit was used as a design input to a scaling and setpoint calculation that determined the nominal trip setpoint and trip level setting based on inaccuracies associated with the instrument loops. The allowance for instrument inaccuracies in the determination of the actual trip setpoint provides conservative assurance that the trip function will be performed at or before reaching the analytical limit. This scaling and setpoint calculation is in accordance with the guidance contained in Regulatory Guide 1.105.

The proposed change to the Level 1 Low Reactor Vessel Water Level setpoint guarantees that core cooling is maintained and fission product loss minimized during a design basis event by ensuring that the associated trips occur within the process parameter value (analytical limit) utilized to confirm the design bases of the plant.

The specific Technical Specification changes associated with the change to the Level 1 Low Reactor Vessel Water Level setpoint makes the Units 1 and 3 Technical Specifications consistent with changes previously approved to the Unit 2 Technical Specifications.

Part C: For Unit 2, instrument identifiers are being added or corrected to reflect the equipment previously installed as part of the ATTS modifications. These changes are administrative in nature and do not involve a design change or other physical change to the plant. These instrument identifiers enhance the useability of the Technical Specifications.

Part D: For Unit 2, Reactor High Water Level, RCIC and HPCI turbine steam line high flow, and drywell pressure instrumentation calibration frequencies and functional test descriptions are being revised to reflect current calculations and test methods. These changes do not reflect a change in equipment, operation of the associated system, or the safety function of that system. The revised scaling and setpoint calculations are in accordance with the guidance contained in Regulatory Guide 1.105. The allowance for instrument inaccuracies in the determination of the actual trip setpoint provides conservative assurance that the trip function will be performed at or before reaching the analytical limit.

In addition, the Rosemount Model 1153 transmitters, which are used for the RCIC and HPCI turbine steam line high flow, and drywell pressure instrument loops, are used in other safety related application at BFN with the new calibration frequency (18 months). These instruments have performed acceptably over the last operating cycle with this longer calibration frequency. The Gould transmitters, which are used for the Reactor High Water Level instrument loops, are not used in other safety related applications at BFN. However, vendor derived data supports the determination that these instruments will perform acceptably with the longer calibration frequency.

The instrumentation in the affected loops was upgraded as part of the installation of the ATTS, which was installed prior to Cycle 6 operation. The new calibration requirements, together with the new instrumentation, are expected to provide a more reliable instrumentation system.

Part E: For Units 1, 2, and 3, the differential pressure instrumentation, which actuates the pressure suppression chamber-reactor building vacuum breakers (Rosemount Model 1153 transmitters), calibration frequency is being revised to reflect current Unit 2 and 3 calculations. The Unit 1 change is based on the similarity of this system and equipment between the three units. A Unit 1 specific calculation will be performed to confirm the calibration frequency prior to Unit 1 restart.

The allowance for instrument inaccuracies in the determination of the actual trip setpoint provides conservative assurance that the trip function will be performed at or before reaching the analytical limit. The scaling and setpoint calculations are in accordance with the guidance contained in Regulatory Guide 1.105. In addition, the specified minimum number of instrument channels per trip system, function, trip level setting, actions required, remarks, functional test, and instrument check reflect current operational requirements are being added in order to be consistent with the treatment of other electronic trip circuitry in the Technical Specifications.

Part F: The capitalization of terms used on the affected Units 1, 2, and 3 Technical Specification pages is an administrative change. This change conforms with the current TS Definitions section. The correction of spelling and capitalization of other words on the same pages is also administrative in nature.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

DESCRIPTION OF THE PROPOSED TECHNICAL SPECIFICATION CHANGE

This proposed change to BFN Technical Specifications consists of six parts.

Part A: The Unit 3 mechanical pressure and differential pressure indicating switches in the Reactor Protection System (RPS) and Emergency Core Cooling System (ECCS) are being replaced with an Analog Transmitter/Trip System (ATTS).

Part B: The Units 1 and 3 reactor vessel water level safety limit is being revised to reflect the analytical limit provided by General Electric and the Level 1 Low Reactor Vessel Water Level setpoint is being revised to provide a more conservative limit.

Part C: For Unit 2, RPS and ECCS instrument identifiers are being added or corrected to enhance useability of the Technical Specifications. These changes do not reflect a change in equipment, operation of the associated system, or the safety function of that system.

Part D: For Unit 2, Reactor High Water Level, Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) Turbine Steam Line High Flow, and Drywell Pressure instrumentation calibration frequencies and functional test descriptions are being revised to reflect current calculations and test methods. These changes do not reflect a change in equipment, operation of the associated system, or the safety function of that system.

Part E: For Units 1, 2, and 3, the differential pressure instrumentation, which actuates the pressure suppression chamber-reactor building vacuum breakers, calibration frequency is being revised. In addition, tables that specify the minimum number of instrument channels per trip system, function, trip level setting, actions required, remarks, functional test, and instrument check are being added.

Part F: Corrects the capitalization of terms used on the affected Units 1, 2, and 3 TS pages in order to conform with the current TS Definitions section. This part also corrects spelling and capitalization of other words on the same pages.

TVA has concluded that operation of Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 in accordance with the proposed change to the technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

A. **The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

   Part A: The Unit 3 modification, which involves the installation of an Analog Transmitter/Trip System (ATTS), replaces older devices with devices of more modern design that perform the same function.

The initiation of control rod insertion to mitigate a design basis accident is contained in Chapter 14 of the BFN Final Safety Analysis Report (FSAR). There is no change in design bases, protective function (initiation of control rod insertion), redundancy, setpoints, or logic associated with the installation of the ATTS. The consequences of a failure of this equipment are no different than that of the original equipment. Since there is no change in any protective functions, nor the creation of any new operational conditions, the proposed amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Part B: The revision to the Units 1 and 3 reactor vessel water level safety limit and the Level 1 low reactor vessel water level setpoint do not reflect any change in plant equipment. The safety limit is being changed to reflect the actual analytical safety limit calculated by General Electric.

The Level 1 low reactor vessel water level trip initiates the Core Spray and Low Pressure Coolant Injection Systems and isolates the Main Steam lines. These actions are taken to mitigate the consequences of a Loss of Coolant Accident. The change in the setpoint affects the timing of the operation of equipment necessary to mitigate the consequences of an accident. A setpoint calculation has been generated which ensures these safety functions are initiated in accordance with the design basis accident analysis presented in Chapter 14 of the Browns Ferry FSAR. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Part C: The addition or correction of Unit 2 instrument identifiers is administrative in nature and does not reflect any modification to plant equipment. These administrative changes do not reflect any change to any precursor for the design basis events or operational transients analyzed in the Browns Ferry FSAR. There is also no change to any protective function or mitigating action for the design basis events or operational transients analyzed in the Browns Ferry FSAR. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

Part D: The change in Unit 2 reactor high water level and Reactor Core Isolation Cooling (RCIC) instrumentation functional test descriptions reflects the equipment currently installed and the functional tests currently being performed.

The changes in calibration frequencies are being made to reflect current setpoint calculations. There are no modifications to plant equipment or changes in instrument setpoints associated with these changes. The calibration frequencies specified by the current setpoint calculations ensure that the associated safety functions are initiated in accordance with the design basis accident analysis presented in Chapter 14 of the Browns Ferry Final Safety Analysis Report (FSAR). Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

Part E: The changes in Units 1, 2, and 3 calibration frequency for the differential pressure instrumentation, which actuates the pressure suppression chamber-reactor building vacuum breakers, is being made to reflect current setpoint calculations. The specified minimum number of instrument channels per trip system, function, trip level setting, actions required, remarks, functional test, and instrument check reflect current operational requirements. There are no modifications to plant equipment or changes in instrument setpoints associated with these changes. The calibration frequencies specified by the current setpoint calculations ensure that the associated safety functions are initiated in accordance with the design basis accident analysis presented in Chapter 14 of the Browns Ferry Final Safety Analysis Report (FSAR). Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

Part F: The proposed correction of the capitalization of terms in order to conform with the current TS Definitions section is administrative in nature and does not reflect any modification to plant equipment. The correction of spelling and capitalization of other words on the same pages is also administrative in nature and does not reflect any modification to plant equipment. These administrative changes do not reflect any change to any precursor for the design basis events or operational transients analyzed in the Browns Ferry FSAR. There is also no change to any protective function or mitigating action for the design basis events or operational transients analyzed in the Browns Ferry FSAR. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

B. **The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

Part A: The installation of the ATTS replaces older devices with devices of more modern design that perform the same function. No new control functions are added. No credible equipment failure modes or single failure are introduced which could result in the inability of redundant safety components or systems to perform their safety functions in accordance with the design basis accident analysis presented in Chapter 14 of the Browns Ferry FSAR. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Part B: The revision to the Units 1 and 3 reactor vessel water level safety limit and the Level 1 low reactor vessel water level setpoint do not reflect any change in plant equipment. The safety limit is being changed to reflect the actual analytical safety limit calculated by General Electric.

The change in the Level 1 low reactor vessel water level setpoint affects the timing of the operation of equipment necessary to mitigate the consequences of an accident. No new failure modes or system interactions are introduced. The same protection functions will still occur at the Level 1 low reactor water level setpoint. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Part C: The addition or correction of Unit 2 instrument identifiers is administrative in nature and does not reflect any modification to plant equipment. The correction of instrument identifiers does not require new system alignments, modifications, or changes in operating procedures. Therefore, no new external threats, system interactions, release pathways, equipment failure modes, or types of operator errors are created. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Part D: The change in Unit 2 reactor high water level and RCIC instrumentation functional test descriptions reflects the equipment currently installed and the functional tests currently being performed. The changes in calibration frequencies are being made to reflect current setpoint calculations. There are no modifications to plant equipment or changes in instrument setpoints associated with these changes. No new failure modes or system interactions are introduced. The same protection functions will still occur at the same setpoints. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Part E: The changes in Units 1, 2, and 3 calibration frequency for the differential pressure instrumentation, which actuates the pressure suppression chamber-reactor building vacuum breakers, is being made to reflect current setpoint calculations. The specified minimum number of instrument channels per trip system, function, trip level setting, actions required, remarks, functional test, and instrument check reflect current operational requirements. There are no modifications to plant equipment or changes in instrument setpoints associated with these changes. No new failure modes or system interactions are introduced. The same protection functions will still occur at the same setpoints. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Part F: The proposed correction of the capitalization of terms in order to conform with the current TS Definitions section is administrative in nature and does not reflect any modification to plant equipment. The correction of spelling and capitalization of other words on the same pages is also administrative in nature and does not reflect any modification to plant equipment. The correction of spelling and capitalization does not require new system alignments, modifications, or changes in operating procedures. Therefore, no new external threats, system interactions, release pathways, equipment failure modes, or types of operator errors are created. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. **The proposed amendment does not involve a significant reduction in a margin of safety.**

Part A: The installation of the ATTS replaces older devices with devices of more modern design that perform the same function. The replacement equipment will improve reliability, accuracy and response times. There are no changes in the systems' design basis, protective function, or logic arrangement. Instrument setpoints and calibration frequencies are supported by Unit 3 specific calculations. Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

Part B: The revision to the Units 1 and 3 reactor vessel water level safety limit and the Level 1 low reactor vessel water level setpoint do not reflect any change in plant equipment. The safety limit is being changed to reflect the actual analytical safety limit calculated by General Electric.

The change in the Level 1 low reactor vessel water level setpoint is supported by a Unit 3 specific setpoint calculation that has been performed in accordance with the methodology endorsed by Regulatory Guide 1.105, Instrument Setpoints for Safety Related Systems.

Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

Part C: The addition or correction of Unit 2 instrument identifiers is administrative in nature and does not reflect any modification to plant equipment. Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

Part D: The change in Unit 2 reactor high water level and RCIC instrumentation functional test descriptions reflects the equipment currently installed and the functional tests currently being performed. The changes in calibration frequencies are being made to reflect Unit 2 specific setpoint calculations. These calculations have been performed in accordance with the methodology endorsed by Regulatory Guide 1.105. There are no modifications to plant equipment or changes in instrument setpoints associated with these changes. Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

Part E: The changes in Units 1, 2, and 3 calibration frequency for the differential pressure instrumentation, which actuates the pressure suppression chamber-reactor building vacuum breakers, is being made to reflect current setpoint calculations. The specified minimum number of instrument channels per trip system, function, trip level setting, actions required, remarks, functional test, and instrument check reflect current operational requirements. The setpoint calculations have been performed in accordance with the methodology endorsed by Regulatory Guide 1.105. There are no modifications to plant equipment or changes in instrument setpoints associated with these changes. Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

Part F: The proposed correction of the capitalization of terms in order to conform with the current TS Definitions section is administrative in nature and does not reflect any modification to plant equipment. The correction of spelling and capitalization of other words on the same pages is also administrative in nature and does not reflect any modification to plant equipment. Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

V. ENVIRONMENTAL IMPACT CONSIDERATION

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

VI. REFERENCES

BFN Unit 2 Analog Transmitter/Trip System Technical Specification Approval -

1. TVA letter to NRC, dated August 23, 1984, in regards to Unit 2 Technical Specification No. 199
2. TVA letter to NRC, dated May 8, 1985, in regards to additional information on the analog trip system
3. TVA letter to NRC, dated November 20, 1985, in regards to additional information on the analog trip system
4. TVA letter to NRC, dated December 30, 1985, in regards to Unit 2 Technical Specification No. 199, Supplement 2
5. TVA letter to NRC, dated April 29, 1986, in regards to Unit 2 Technical Specification No. 199, Supplement 3
6. NRC letter to TVA, dated August 19, 1986, in regards to Amendment No. 125

BFN Unit 2 Tobar Transmitters Technical Specification Approval -

7. TVA letter to NRC, dated February 24, 1989, Technical Specification No. 263 - Tobar Transmitters
8. NRC letter to TVA, dated July 7, 1989, Technical Specification for Tobar, Inc. Transmitters (TS 263)

BFN Unit 2 Reactor Pressure Vessel Low Water Level Trip Technical Specification Approval -

9. TVA letter to NRC, dated August 6, 1990, TVA BFN Technical Specification (TS) No. 291 - Revision to Level 1 Low Reactor Pressure Vessel (RPV) Water Level
10. TVA letter to NRC, dated October 9, 1990, TVA BFN Technical Specification (TS) No. 291 - Revision to Level 1 Low Level Reactor Pressure Vessel (RPV) Water Level
11. NRC letter to TVA, dated January 2, 1991, Issuance of Amendment (TS 291)