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Jerry C. Roberts
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RBG-47157

July 27, 2011

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: License Amendment Request
Changes to Technical Specification 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation"
River Bend Station, Unit 1
Docket No. 50-458
License No. NPF-47

Dear Sir or Madam:

In accordance with the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Entergy Operations, Inc. (Entergy) is submitting a request for a amendment to the Technical Specifications (TS) for River Bend Station (RBS), Unit 1. A change is proposed to Technical Specification (TS) 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation" to revise the allowable value setpoints for the Main Steam Tunnel Temperature functions 1.e, 3.f and 4.h.

This setpoint revision is based upon a revision to the analytical limit calculation. The change will provide additional margin for elevated temperatures in the Main Steam Tunnel - North during the summer reliability period.

In addition to the changes to the RBS Technical Specifications the enclosed Emergency Plan changes are also submitted for NRC staff review and approval as required by 10 CFR 50.54(q), 50.4, and 50.90. The proposed change modifies the station's Emergency Action Levels (EAL) in support of the proposed changes to TS 3.3.6.1.

The proposed change to the EALs was evaluated against the criteria of 10 CFR 50.47, 10 CFR 50, Appendix E and other NRC guidance documents. This EAL change has been evaluated in accordance with 10 CFR 50.54(q) and the guidance contained in NRC RIS 2005-02, and the evaluation has deemed NRC prior approval is required. The requested change is acceptable in that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

The proposed changes have been evaluated in accordance 10 CFR 50.92(c) and it has been determined that the changes involve no significant hazards considerations. Attachment 1 provides the No Significant Hazards Consideration for the change.

Attachment 1 provides a description of the proposed change. Attachment 2 provides the existing TS pages marked up to show the proposed change. Attachment 3 provides the existing TS Bases pages marked up to show the proposed change (for information only). Attachment 4 provides Justification of Emergency Plan Emergency Action Level change, a mark-up of the latest Emergency Action Levels and a revised copy of Emergency Action Levels. Attachment 5 provides a summary of the regulatory commitments made in this submittal.

This change has been reviewed and approved by the Onsite Safety Review Committee (OSRC).

Although this request is neither exigent nor emergency, your prompt review is requested. Once approved, the amendment shall be implemented within 60 days. If you have any questions or require additional information, please contact Mr. Joseph A. Clark at (225) 381-4177.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 27, 2011

Sincerely,

J C Robert

JCR/JAC/bmb

Attachments:
1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)
3. Changes to Technical Specification Bases Pages – For Information Only
4. Justification of Emergency Plan Emergency Action Level change
5. List of Regulatory Commitments

cc: Regional Administrator
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U. S. Nuclear Regulatory Commission
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Attachment 1

RBG-47157

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

The proposed amendment would revise River Bend Station (RBS), Unit 1 Technical Specification (TS) 3.3.6.1;" Primary Containment and Drywell Isolation Instrumentation," allowable value setpoints contained in TS 3.3.6.1-1 items; 1.e "Main Steam Tunnel Temperature–High," 3.f "Main Steam Line Tunnel Ambient Temperature–High," 4.h "Main Steam Line Tunnel Ambient Temperature–High."

In addition to the changes to the Technical Specification allowable value setpoints a corresponding change will be made to increase the nominal trip value in the Technical Requirements Manual (TRM) functions identified above.

This request is to change the Technical Specification allowable value and the associated change to the Emergency Action Level (EAL) under 10 CRF 50.90. The changes to the setpoint nominal values and the use of GOTHIC are addressed under 10 CFR 50.59.

2.0 PROPOSED CHANGE

The Technical Specifications (TS) functions which this proposed change addresses the new proposed allowable value of ≤ 183.0 °F are the following:

1. TS Table 3.3.6.1-1 Function 1.e – Main Steam Line Isolation -Main Steam Tunnel Temperature-High
2. TS Table 3.3.6.1-1 Function 3.f – Reactor Core Isolation Cooling (RCIC) System Isolation -Main Steam Line Tunnel Ambient Temperature-High
3. TS Table 3.3.6.1-1 Function 4.h – Reactor Water Cleanup (RWCU) System Isolation -Main Steam Line Tunnel Ambient Temperature-High

Attachment 2 contains a markup of the proposed Technical Specification pages.

In addition to the identified changes to the Technical Specifications above, the BASES will be revised upon implementation to include the following information based upon TSTF-493 BASES revision to NUREG-1434, SR 3.3.6.1.5, to include the following statement;

> For Functions 1.e, 3.f and 4.h there is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

Attachment 3 contains a markup of the proposed Technical Specification BASES page.

3.0 BACKGROUND

During the summer months the Main Steam Line Tunnel area ambient temperatures in conjunction with minor steam leaks in the MSL Tunnel have approached the trip setpoints. These setpoints are designed to detect a 25 gpm leak in the Main Steam

Tunnel – North area and initiates plant isolations. The proposed changes will maintain the design leak detection criteria while providing additional margin for plant operation.

Technical Specification 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation" contains the Allowable Values for isolation instrumentation that cause initiation of primary containment and reactor coolant pressure boundary (RCPB) isolation. Functional diversity is provided by monitoring a wide range of independent parameters. One of the parameters monitored to provide isolation of the main steam (MS) lines, reactor core isolation cooling (RCIC) lines and reactor water clean up (RWCU) lines is Main Steam Tunnel Temperature – High.

In order to provide additional margin for elevated temperatures in the Main Steam Tunnel - North during the summer reliability period, the analysis that establishes the analytical limit for the main steam tunnel temperature corresponding to a 25 gpm leak has been refined to remove unnecessary conservatism and raise the analytical temperature limit.

The Analytical Limit (AL) is the point where desired action is to be initiated to maintain the integrity of physical barriers or other plant characteristics which must remain intact or operational. The Allowable Value is chosen based on an Analytical Limit that is calculated to detect a leak equivalent to 25 gpm.

A new Analytical Limit of 194.77 °F was determined based upon a refined analysis that established the analytical limit for the main steam tunnel temperature corresponding to the same 25 gpm leak, The previous Analytical Limit was 154 °F. The determination of the new analytical limit is discussed below in section 4.1.

The corresponding setpoint calculation determined a new limiting Allowable Value of ≤183 °F based on the new Analytical Limit of 194.77 °F. The Allowable Value of ≤183 °F was derived from the Analytical Limit by correcting for calibration, process and other instrument uncertainties as discussed in Technical Specification Bases 3.3.6.1 and USAR Section 7. The new Nominal Trip Setpoint of 173 °F was determined considering instrument drift, with the drift value for the 24 month fuel cycle, and loop uncertainty. The determinations of the setpoints are discussed in sections 4.3 and 4.4 below.

The Emergency Plan is affected by the change to the main steam tunnel temperature trip setpoint because Emergency Implementing Procedure EIP-2-001, "Classification of Emergencies" contains this setpoint in an Emergency Action Level and will require NRC approval. The discussion and justification is included in Attachment 4 of this submittal.

4.0 TECHNICAL ANALYSIS

4.1 Revised Analytical Limit

As part of this amendment request, the analytical limit for the main steam tunnel temperature – high is increased from 154°F to 194.77°F. The basis for the current main steam tunnel temperature – high setpoint analytical limit is an area temperature rise equivalent to a 25 gpm steam leakage rate as is described in RBS USAR Section 5.2.5.1.3, "Detection of Leakage External to Containment". The basis is unchanged by this request.

It should be noted that the main steam tunnel temperature – high isolation function is not credited in any analysis reported in the USAR.

The following summarizes the principle differences between the current analytical limit calculation and the proposed analytical limit calculation:

Computer Program Used:

The current calculation was performed using the THREED computer program, a Stone & Webster modified version of RELAP4 intended for sub-compartment analysis. The proposed calculation was performed using the Generation of Thermal-Hydraulic Information for Containments (GOTHIC) computer program. RBS has used GOTHIC for analysis of sub-compartments in containment as well as the analysis of high energy line breaks in the Auxiliary Building as described in submittals to the NRC dated May 14, 2002, supplemented by letters dated June 27 and July 9, 2003, April 7 and May 12, 2004, and approved in Amendment 139 to NPF-47 dated May 20, 2004."

Steam Tunnel Unit Cooler Simulation:

The current calculation simulates cooling in the steam tunnel by using two boundary conditions. One boundary condition represents the suction to the unit cooler providing cooling for the steam tunnel and the other represents the unit cooler return air. A fan is used to simulate the unit cooler flow, drawing air from the steam tunnel and to simulate the return air flow. This arrangement returns air to the leak volume at a constant temperature of 105°F and 32% relative humidity, regardless of the conditions of the air being drawn from the leak volume. This approach is highly conservative as it ignores the fact that as the suction temperature and humidity rise the return temperature will also rise. Thus the calculated temperature at one hour is artificially low, which is conservative for this application.

The proposed calculation explicitly simulates the unit cooler using the AIRCOOLER component in GOTHIC. The AIRCOOLER model in GOTHIC calculates both sensible and latent heat removal. Inputs for the AIRCOOLER were selected to represent design performance of the unit cooler.

This change in steam tunnel unit cooler modeling is the main contributor to the increase in the analytical limit.

Initial Conditions:

The current analytical limit is based on an initial temperature in the steam tunnel of 128°F. The proposed calculation uses a value of 110°F, which is based on ~ 2 years of plant data and represents a bounding low value during power operation.

The proposed calculation uses a value of 60°F for the service water temperature which is the cooling medium for the unit cooler. There is no similar input in the current calculation method.

Low values for initial area temperature or service water temperature will yield low values for the area temperature at one hour, which is conservative. Selecting bounding low values for the initial temperature and service water temperature will result in a conservative setpoint applicable to foreseeable temperatures.

Conclusion

As identified above, this function is not credited in current design basis analysis. Also, the proposed changes maintain the 25 gpm design criteria. As a result, there are no changes to other design basis evaluations, plant response or consequences to an event.

4.2 Current Licensing Basis Review for the Main Steam Tunnel Ambient Temperature Isolations.

USAR Section 5.2.5 discusses the reactor coolant pressure boundary (RCPB) and ECCS leakage detection system.

Section 5.2.5.1, "Leakage Detection Methods," identifies the system is designed to be in conformance with Regulatory Guide 1.45 and Reference Section IEEE 279. This proposed change to TS does not change the conformance to IEEE 279. Regulatory Guide 1.45 deals with leakage inside containment and is not applicable to the proposed change.

USAR section 5.2.5.1.3, "Detection of Leakage External to Containment", describes that main steam tunnel ambient temperature is used as an indication of RCPB leakage and that the temperature setpoints are predicated on an area temperature rise equivalent to reactor coolant leakage into the monitored areas of 25 gpm. The proposed change does not change the basis for these instruments. The setpoints are still predicated on an area temperature rise equivalent to reactor coolant leakage into the monitored areas of 25 gpm.

USAR section 5.2.5.2, "Leak Detection Instrumentation and Monitoring", states that high temperature alarms in the main control room provide a signal to close the main steam and drain line isolation valves, RCIC steam isolation valves and the RWCU system isolation valves. This description remains unchanged. USAR Tables 5.2-7 and 5.2-8 indicate that high main steam tunnel temperature provides both alarm and isolation for the Main Steam, RCIC and RWCU systems. The proposed change does not change the number of temperature elements employed or where the instruments are physically located. The proposed change does not delete or remove any high temperature alarms.

USAR Section 5.2.5.1.3 "Detection of Leakage External to Containment" discusses the areas outside the containment which are monitored for primary coolant leakage equipment areas in the auxiliary building, the main steam tunnel, and the turbine building. These monitors have temperature setpoints which are predicated on an area temperature rise equivalent to reactor coolant leakage into the monitored areas of 25 gpm. The proposed change does not change the basis for these instruments. The setpoints are still predicated on an area temperature rise equivalent to reactor coolant leakage into the monitored areas of 25 gpm. The proposed change dictates a change in

the Allowable Value for the main steam tunnel temperature isolation instrumentation due to a change in the engineering model being used. As stated above, the GOTHIC model has been previously used for River Bend Station calculations and is being applied for determining the Analytical Limit for these instruments. The Allowable Value is then determined from the Analytical Limit using the existing plant setpoint methodology.

USAR section 7.1.2.5, "NSSS and BOP Safety-Related Set Point Methodology", states that the methodology used for determination of NSSS safety related setpoints is documented in General Electric document NEDC-31336 that was approved by the NRC in 1993. The change to the setpoints is done in accordance with this methodology; therefore, no change to the USAR or other licensing basis is needed. Further discussion of the changes to the setpoint using the approved method are included below.

USAR section 7.6.1.2, "Leak Detection System" describes that the safety related portion of the leak detection system includes main steam, RCIC and RWCU system leak detection and that one of the signals used to initiate alarms and isolation is main steam tunnel area high ambient temperature. As previously stated the proposed change does not affect the design or operation of the main steam tunnel ambient temperature instrumentation. In addition, the setpoint will continue to be based on present licensing basis criteria.

USAR 7.3.1.1.2, "Containment and Reactor Vessel Isolation Control System (CRVICS)," discusses the instrument channels, trip logics, and actuation circuits that automatically initiate valve closure providing isolation of the containment and/or reactor vessel, and initiation of systems to limit the release of radioactive materials. This USAR section also discusses variables which provide inputs to the CRVICS logics for initiation of reactor vessel and containment isolation, as well as the initiation or trip of other plant functions when predetermined limits are exceeded. As previously stated the proposed change does not affect the design or operation of the main steam tunnel ambient temperature instrumentation. In addition, the setpoint will continue to be based on present licensing basis criteria.

4.3 Allowable Value Determination – River Bend Station (RBS) Setpoint Methodology

The RBS setpoint control program is implemented utilizing engineering process controls, plant procedural controls, and the corrective action process. This program ensures the associated instrument channel is capable of performing its specified safety functions.

USAR Section 7.1.2.5, "NSSS and BOP Safety-Related Set Point Methodology," states that the methodology used in determining BOP safety system set points is in accordance with Regulatory Guide 1.105, Revision 1 and IEEE 279-1971. It references USAR Table 1.8-1 for the River Bend Station position on Regulatory Guide 1.105. It also states that the methodology used in determining NSSS safety system setpoints is similar to that described above for BOP safety system setpoints. This methodology is documented in NEDC-31336, General Electric Instrument Setpoint Methodology, dated October 1986. This document was developed by the Instrument Setpoint Methodology Owners Group (ISMG) and approved by the staff on February 9, 1993. The proposed revision does not change any requirement for setpoint methodology.

This methodology was recently discussed with the NRC staff during the staff review of the RBS 24 month cycle submittal and approval. In a letter dated August 17, 2010 (ML102350155) Entergy responded to an NRC question (RAI) concerning compliance with RIS 2006-17. As stated in this response the original Nominal Trip Setpoint, Allowable Value, and Analytical Limit design bases were established from the supplier design requirements. Calculations were developed approximately 10 to 15 years ago to confirm that the plant conditions (measurement and test equipment, device accuracies, drift, etc,) were conservative relative to the assumptions in the design basis.

These original calculations, subsequent revisions and the methods of calibration continue to confirm that the existing Technical Specification Allowable Values and Technical Requirement Manual Nominal Trip Setpoints are conservative with respect to the original design basis. Figures and Tables supplied with the response summarized the important values created by or relative to sample setpoint calculations for the 24 month submittal.

The conclusion to the RAI response was that the tables and charts demonstrated that the setpoint calculations and associated surveillance procedures affected by the change to a 24 month fuel cycle at River Bend Station met the intent of NRC Regulatory Issue Summary 2006-17. Amendment 168 to the River Bend operating license for the 24 month cycle was issued August 31, 2010 (ML 102350266).

The methodology described in the August 17, 2010 letter has not changed as a result of this request. The only change is the computer code being used to determine the analytical limit and, as a result, the corresponding setpoint values using the approved methodology. Therefore the proposed change is not a change to the setpoint methodology.

4.4 Nominal Trip Setpoint

The RBS setpoint control program also determines the nominal trip setpoint. This program contains the same engineering process controls, plant procedural controls, and the corrective action process used for determining and controlling the allowable value.

The proceeding discussion of license basis and setpoint method described above also applies to the nominal trip setpoint. In addition the as-found and as-left values are identified in the setpoint calibration procedures. Setpoints found outside of the established as-found limits are identified in the corrective action process.

The nominal trip setpoints are identified in the Technical Requirements Manual (TRM). Changes to the Nominal Trip Setpoint are controlled under 10 CFR 50.59.

4.5 Equipment Qualification

Equipment qualifications have been reviewed within the affect area and determined the equipment will perform as required with the revised setpoints.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

Section 50.36 to Title 10 to the Code of Federal Regulations (10 CFR) Part 50, requires that TS include limiting conditions for operation (LCOs) for any structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. Section 50.36 to 10 CFR, also requires that TS Surveillance Requirements (SRs) be requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCO will be met. When an LCO for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

Appendix A to 10 CFR 50, General Design Criteria for Nuclear Power Plants, Criterion 13 Instrumentation and Control states that:

> Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

USAR section 3.1.2.13 discusses the RBS conformance to this General Design Criteria. The proposed changes do not affect the conformance as presented in this USAR section.

Appendix A to 10 CFR 50, Criterion 24-Separation of Protection and Control Systems states that:

> The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

USAR Section 3.1.2.13 discusses the RBS conformance to this General Design Criteria. The proposed changes do not affect the conformance as presented in this USAR section.

Regulatory Guide (RG) 1.105, Revision 1, Setpoints for Safety-Related Instrumentations, describes a method acceptable to the NRC staff for complying with the NRC's regulations for ensuring that instrumentation setpoints are initially within and remain within the TS limits. USAR Table 1.8-1 and Section 7.1.2 discuss the RBS conformance to this Regulatory Guide. The proposed changes do not affect the conformance as presented in this USAR section.

Regulatory Issue Summary (RIS) 2006-17, "NRC Regulatory Issue Summary 2006-17 NRC Staff Position On The Requirements Of 10 CFR 50.36, "Technical Specifications," Regarding Limiting Safety System Settings During Periodic Testing And Calibration Of Instrument Channels,",dated August 24, 2006, discusses the requirements of 10 CFR 50.36 related to Limiting Safety System Settings (LSSS) and provides an approach acceptable to the NRC to address LSSS issues. The LSSSs are settings for automatic protective devices related to those variables having significant safety functions.

As discussed above in Section 4.3, compliance with this RIS has previously been discussed in response to a Request for Additional Information (RAI) during the review of the RBS 24 month cycle submittals. This resulted in part in the approval of Amendment 168 to the RBS facility operating license on August 31, 2010 (ML 102350266).

5.2 <u>No Significant Hazards Consideration</u>

Entergy has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," discussed below.

The requested change would affect certain Technical Specification (TS) Allowable Values for Main Steam Tunnel Ambient Temperature isolation instrumentation and the Emergency Action Levels supporting the Emergency Plan.

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

    Response: No.

    The proposed change increases the Technical Specification allowable value for the main steam tunnel ambient temperature isolation instrumentation for the main steam line isolation, Reactor Core Isolation Cooling System isolation and the Reactor Water Cleanup System isolation. This TS change does not introduce the possibility of an increase in the probability or consequences of an accident because the basis for the instrument setpoint is not being changed as a result of this request. The proposed TS change involves no physical alteration of the plant. The proposed TS change does not degrade the performance of, or increase the challenges to, any safety systems assumed to function in the accident analysis. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The consequences of a previously evaluated accident are not significantly increased. The proposed change does not affect the performance of any equipment credited to mitigate the radiological consequences of an accident. The basis for the main steam tunnel ambient temperature isolation instrumentation has not changed as a result of this proposed Allowable value change.

The proposed change to the Emergency Action Level (EAL) does not increase the probability of an accident. The change only impacts the initial condition for entry into the Emergency Plan and thus has no impact on the probability of an event. The proposed change to the Emergency Action Level (EAL) does not increase the consequences of an accident. As described in the Technical Analysis the revised setpoint continues to support the current licensing basis and event analysis.

Because the process, personnel, and equipment involved in implementing the Emergency Plan would complete the same functions as those completed under the existing Emergency Plan, the plan would continue to ensure adequate protection of public health and safety.

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

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

   Response: No.

   As discussed above, the proposed change involves increasing the TS allowable value for the for the main steam tunnel ambient temperature isolation instrumentation for the main steam line isolation, Reactor Core Isolation Cooling System isolation and the Reactor Water Cleanup System isolation. The proposed TS change does not introduce any failure mechanisms of a different type than those previously evaluated, since there are no physical changes being made to the facility. No new or different equipment is being installed. No installed equipment is being operated in a different manner. The computer program being used has been previously used and reviewed. As a result, no new failure modes are being introduced. There are no new types of failures or new or different kinds of accidents or transients that could be created by these changes.

   The change affects the implementation of the Emergency Plan by changing the EALs temperature value for entry into the Emergency Plan; however, the basis for the temperature value is not changed. The change to the EAL does not impact any plant equipment or systems needed to respond to an accident, nor does it change the results of an analysis of plant accident consequences.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

As discussed above, the proposed change involves increasing the TS allowable value for the for the main steam tunnel ambient temperature isolation instrumentation, the main steam line isolation, the Reactor Core Isolation Cooling System isolation and the Reactor Water Cleanup System isolation. The effect of this change on system availability is not significant, based on the determination that the basis for the allowable values is not being revised. The proposed change does not adversely affect the condition or performance of structures, systems, and components relied upon for accident mitigation. The proposed change does not result in any hardware changes. Existing operating margin between plant conditions and actual plant setpoints is not significantly reduced due to these changes. The proposed change does not significantly impact any safety analysis assumptions or results.

The change to the Emergency Plan does not reduce the margin of safety currently provided by the plan. As discussed in this submittal the change does not revise the design criteria of detecting a 25 gpm leak. Also the methods used to determine the revised analytical limit and setpoint values are currently accepted. The proposed change does not impact other design basis evaluations or consequences. Therefore the changes do not affect a margin of safety identified in the plant accident analysis.

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

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

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 Precedence

Similar request to revise instrument setpoints has been identified as follows:

1 Pilgrim Nuclear Power Station," Request for Technical Specification Change Concerning Change of Trip Level Settings, Calibration Frequencies, and Editorial Changes, Revision 1," Dated October 10 , 2002

7.0 References

1 River Bend Station, "License Amendment Request (LAR) 2001-43, "High Energy Line Break Analysis Method," Dated May 14, 2002
2 General Electric Instrument Setpoint Methodology, NEDC-31336, October 1986.
3 River Bend Station, "Response to Request for Additional Information on License Amendment Request 2009-05, 24-month Fuel Cycles," Dated August 17, 2010
4 NRC, "River Bend Station, Unit 1 – Issuance of Amendment RE: Revise Technical Specification Surveillance Requirement Frequencies from 18- to 24-Month Fuel Cycle Interval (TAC No. ME1872)," Dated August 31, 2010

Attachment 2

RBG-47157

Technical Specification Markup

Table 3.3.6.1-1 (page 1 of 5)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level—Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ -147 inches
b. Main Steam Line Pressure—Low	1	2	E	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 837 psig
c. Main Steam Line Flow—High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 190 psid, Line A ≤ 194 psid, Line B ≤ 194 psid, Line C ≤ 194 psid, Line D
d. Condenser Vacuum—Low	1,2[(a)], 3[(a)]	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 7.6 inches Hg vacuum
e. Main Steam Tunnel Temperature—High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 18348.6°F
f. deleted					
g. deleted					
h. deleted					

(continued)

(a) With any turbine stop valve not closed.

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Table 3.3.6.1-1 (page 3 of 5)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. RCIC System Isolation (continued)					
d. RCIC Turbine Exhaust Diaphragm Pressure–High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig
e. RCIC Equipment Room Ambient Temperature–High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 186.4°F
f. Main Steam Line Tunnel Ambient Temperature–High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 183~~43.6~~°F
g. Main Steam Line Tunnel Temperature Timer	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	NA
h. RHR Equipment Room Ambient Temperature–High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 121.1°F
i. RCIC/RHR Steam Line Flow–High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 64.2 inches water
j. Drywell Pressure–High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 1.88 psid
k. Manual Initiation	1,2,3	1	G	SR 3.3.6.1.6	NA

(continued)

Table 3.3.6.1-1 (page 4 of 5)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Reactor Water Cleanup (RWCU) System Isolation					
a. Differential Flow—High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 62.1 gpm
b. Differential Flow—Timer	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 47 seconds
c. RWCU Heat Exchanger Equipment Room Temperature—High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 107.5°F
d. RWCU Pump Rooms Temperature—High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 169.5°F
e. RWCU Valve Nest Room Temperature—High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 114.5°F
f. RWCU Demineralizer Rooms Temperature—High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 114.5°F
g. RWCU Receiving Tank Room Temperature—High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 114.5°F
h. Main Steam Line Tunnel Ambient Temperature—High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 18348.6°F
i. Reactor Vessel Water Level—Low Low, Level 2	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -47 inches
j. Standby Liquid Control System Initiation	1,2	1	I	SR 3.3.6.1.6	NA
k. Manual Initiation	1,2,3	2	G	SR 3.3.6.1.6	NA

(continued)

Attachment 3

RBG-47157

Technical Specification BASES Mark-up

(For Information Only)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.3 (continued)

Table 3.3.6.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of References 5 and 6.

SR 3.3.6.1.4 and SR 3.3.6.1.5

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. For Functions 1.e, 3.f and 4.h there is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of SR 3.3.6.1.4 and SR 3.3.6.1.5 is based on the assumption of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.6.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on PCIVs in LCO 3.6.1.3 and on drywell isolation valves in LCO 3.6.5.3 overlaps this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

Attachment 4

RBG-47157

Emergency Planning Evaluation

Emergency Planning Evaluation

The "Main Steam Line Tunnel Ambient Temperature – High" trip setpoint value in the Technical Requirements Manual (TRM) Nominal Trip Setpoint (NTSP) specification is being changed from 144°F to 173°F. When temperature sensors reach this setpoint a signal is sent to actuate the Control Room annunciator "Main Steam Line Tunnel Ambient Temperature - High" and a signal is sent to close the Main Steam Isolation Valves (MSIVs), Reactor Core Isolation Cooling (RCIC) steam isolation valves, and the Reactor Water Cleanup (RWCU) system isolation valves on the piping penetrations running outside the containment barrier into the Main Steam Tunnel.

The "Main Steam Line Tunnel Ambient Temperature – High" setpoint is based on an area temperature rise equivalent to steam leakage into the steam tunnel of 25 gpm. As discussed in Attachment 1, an updated calculation determined a revised Analytical Limit (AL) of 194.77 °F for the 25 gpm steam tunnel leakage by using more realistic assumptions and removing excessive conservatisms. Also, as discussed in Attachment 1, an updated TRM NTSP calculation determined a limit value of 173 °F for the "Main Steam Line Tunnel Ambient Temperature - High" trip function and a Technical Specifications Allowable Value (AV) limit of ≤183 °F.

This plant change affects the Main Steam Tunnel NTSP temperature value (144°F to 173°F) listed for Fission Product Barrier IC-EALs, Reactor Coolant System (RC) Barrier RC3 (RCS Leak Rate) "Loss" and "Potential Loss" determination. The RC3 Bases uses the Main Steam Tunnel isolation setpoint as the bases for determining a reactor coolant leak outside containment in determining the EAL entry condition.

<u>**RC3 Bases (Excerpts, See Enclosure A for entire bases)**</u>

LOSS – An unisolable Main Steam Line break represents a loss of the reactor coolant system barrier. This EAL is included for consistency with the ALERT emergency classification.

The leak is NOT isolable from the Main Control Room **OR** an attempt for isolation from the Main Control Room panels has been made and was <u>not</u> successful. An attempt for isolation should be made prior to the accident classification. If isolable upon identification, this INITIATING CONDITION is not applicable. Dispatch of operators outside the Control Room for manual attempts to close the valve is not considered.

POTENTIAL LOSS - The alarms for the high area ambient temperature are associated with the TS 3.3.6.1 allowable values for primary containment isolation. The T.S. allowable high ambient temperature setpoints are set low enough to detect a leak equivalent to 25 gpm. The use of an isolation alarm and EOP condition provides a method of rapid identification of the EAL condition without the task of obtaining values from a back panel indication. [Note: the TS 3.3.6.1 new temperature value is 183°F and the TRM NTSP value with design margin is being changed from 144°F to 173°F.]

RC3 Loss bases specify that an attempt for isolation from the Main Control Room should be made prior to the accident classification. This action would be performed after receiving the high temperature isolation signal. The temperature isolation trip setpoint

will now occur at 173°F. RC3 Potential Loss bases specify the setpoint is set low enough to detect a 25 gpm leak and use of the isolation alarm (will be set at 173°F) provides a rapid identification of the EAL entry condition. The new higher setpoint still provides conservative indication of the 25 gpm leak. The new higher setpoint could slightly delay the declaration of the event due to the temperature change from 144 to 173°F. However, it does not delay declaration beyond the 25 gpm limit that is in the RC3 bases. The temperature in the EAL IC was chosen because they match the isolation and alarm setpoints which provide positive indication of the entry condition. This change preserves that condition. The RC3 Main Steam Tunnel EAL determination temperature value should be the same value as the isolation and alarm setpoint value.

Changing the RC3 Loss and Potential Loss Main Steam Tunnel temperature from 144°F to 173°F does NOT decrease the effectiveness of the EALs since the new temperature value will be the isolation and alarm setpoints for a 25 gpm leak in the main steam tunnel as described in the EAL bases. The RC3 EAL Main Steam Tunnel numerical temperature value should be equal to the Main Steam Tunnel isolation setpoint value to ensure the value will detect a 25 gpm leak, EAL temperature matches isolation setpoint and Operators will receive an isolation alarm to readily identify potential EAL entry conditions.

The increase in the "Main Steam Line Tunnel Ambient Temperature-High" trip function temperature from 144 °F to 173 °F will help minimize the possibility of a spurious instrument trip to isolate yet maintain the ability to detect and isolate based on a leak of 25 gpm in the Main Steam Line Tunnel.

This increase in the setpoint for "Main Steam Line Tunnel Ambient Temperature-High" trip function and alarm will still maintain the plant within current design basis for the detection of 25 gpm leaks in the Main Steam Line Tunnel.

The temperature value change does not impact the requirement for recognition and classification of an emergency within approximately 15 minutes of the time indications are available. The temperature change still maintains the capability to identify conditions and does not degrade the availability of the approved EAL threshold indications. The temperature value change does not affect the time requirement or capability to perform in a timely manner the functions for notification, protective action recommendation, and alert and notification of the public.

REFERENCE USE

Field Code Changed

Deleted: 2

FISSION PRODUCT BARRIER

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOUE
FPB Loss / Potential Loss	FG1 (1 2 3 4 5 D) Loss of any two barriers and loss or potential loss of third barrier **Emergency Action Level(s):** (1) 1. Loss of any two barriers **AND** Loss or potential loss of third barrier	FS1 (1 2 3 4 5 D) Loss or potential loss of any two barriers **Emergency Action Level(s):** (1) 1. Loss or potential loss of any two barriers	FA1 (1 2 3 4 5 D) Any loss or any potential loss of either fuel clad or RCS **Emergency Action Level(s):** (1) 1. Any loss or any potential loss of fuel clad **OR** Any loss or any potential loss of RCS	FU1 (1 2 3 4 5 D) Any loss or any potential loss of containment **Emergency Action Level(s):** (1) 1. Any loss or any potential loss of containment

FUEL CLAD (FC) Barrier			REACTOR COOLANT SYSTEM (RC) Barrier			PRIMARY CONTAINMENT (PC) Barrier		
Parameter	Loss	Potential Loss	Parameter	Loss	Potential Loss	Parameter	Loss	Potential Loss
FC1 Primary coolant activity level	Coolant activity greater than 300 µCi/gm dose equivalent I-131	None	RC1 Drywell pressure	Drywell pressure greater than 1.68 psid with indications of reactor coolant leak in drywell.	None	PC1 Primary containment pressure	Rapid unexplained loss of pressure following initial pressure rise **OR** PC pressure response not consistent with LOCA conditions	PC pressure 15 psig and rising **OR** PC hydrogen in the unsafe zone of HDOL curve **OR** DW hydrogen concentration greater than 9%
FC2 Reactor vessel water level	RPV water level less than -186 inches	RPV water level less than -162 inches	RC2 Reactor vessel water level	RPV water level less than -162 inches with indication of reactor coolant leak in drywell	None	PC2 Reactor vessel water level	None	Entry into PC flooding procedures SAP-1 and SAP-2
FC3 Primary Containment radiation monitors	Containment radiation monitor RMS-RE16 reading greater than 3,000 R/hr	None	RC3 RCS Leak Rate	Unisolable main steam line break as indicated by the failure of both MSIVs in any one line to close **AND** High MSL flow annunciator (P601-19A-A2) **AND** Main Steam Tunnel Temperature greater than 173°F	RCS leakage greater than 50 gpm inside the drywell **OR** Unisolable primary system leak outside PC as indicated by any area temperature alarm or area radiation level alarm in Table F2	PC3 Primary containment isolation failure or bypassed	Failure of both valves in any one line to close and downstream pathway to the environment exists **OR** Intentional venting per EOPs or SAPs **OR** Unisolable RCS leakage outside PC as indicated by any area temperature or area radiation level in Table F1	None
			RC4 Drywell radiation	Drywell radiation monitor RMS-RE20 reading greater than 100 R/hr due to reactor coolant leakage	None	PC4 Significant radioactive inventory in primary containment	None	Containment radiation monitor RMS-RE16 reading greater than 10,000 R/hr
FC4 ED judgment	Any condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier	Any condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier	RC5 ED judgment	Any condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	Any condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	PC5 ED judgment	Any condition in the opinion of the Emergency Director that indicates loss of the primary containment barrier	Any condition in the opinion of the Emergency Director that indicates potential loss of the primary containment barrier

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Plant Modes (white boxes indicate applicable modes) 1 Power Operations 2 Startup 3 Hot Shutdown 4 Cold Shutdown 5 Refuel D Defueled

Field Code Changed

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FISSION PRODUCT BARRIER

TABLE F1			
PC 3 Loss of Primary Containment			
Parameter	Area Temperature Max Safe Operating Value	Area Radiation Level DRMS Grid 2	Max Safe Operating Value
RHR A equipment area	200° F	1213	9.5E+03 mR/hr
RHR B equipment area	200° F	1214	9.5E+03 mR/hr
RHR C equipment area	N/A	1215	9.5E+03 mR/hr
RCIC room	200° F	1219	9.5E+03 mR/hr
MSL Tunnel	200° F	N/A	
RWCU pump room 1 (A) / 2 (B)	200° F	N/A	

TABLE F2			
RC 3 Potential Loss of RCS			
Parameter	Area Temperature (isolation temperature alarm)	Area Radiation Level DRMS Grid 2	Max Normal Operating Value
RHR A equipment area	117° F (P601-20A-B4)	1213	8.2E+01 mR/hr
RHR B equipment area	117° F (P601-20A-B4)	1214	8.2E+01 mR/hr
RHR C equipment area	N/A	1215	8.2E+01 mR/hr
RCIC room	182° F (P601-31A-B6)	1219	1.20E+02 mR/hr
MSL Tunnel	173° F (P601-19A-A1/A3/B1/B3)	N/A	
RWCU pump room 1 (A) / 2 (B)	165° F (P680-1A-A2/B2)	N/A	

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Field Code Changed
Deleted: 118

EAL BASES

RC3

REACTOR COOLANT SYSTEM
Emergency Action Level:

RCS leak rate

EAL Threshold:

LOSS: Unisolable main steam line break as indicated by the failure of both MSIVs in any one line to close

AND

High MSL flow annunciator (P601-19A-A2)

AND

Main Steam Tunnel Temperature greater than 173° F

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POTENTIAL LOSS: RCS leakage greater than 50 gpm inside the drywell

OR

Unisolable primary system leak outside PC as indicated by any area temperature alarm or area radiation level alarm in Table F2

TABLE F2			
RC 3 Potential Loss of RCS			
Parameter	Area Temperature (Isolation temperature alarm)	Area Radiation Level DRMS Grid 2	Max Normal Operating Value
RHR A equipment area	117° F (P601/20A/B04)	1213	8.2 E+01 mR/hr
RHR B equipment area	117° F (P601/20A/B04)	1214	8.2 E+01 mR/hr
RHR C equipment area	N/A	1215	8.2 E+01 mR/hr
RCIC room	182° F (P601/21A/B06)	1239	1.20 E+02 mR/hr
MSL Tunnel	173° F (P601/19A/A01, A03, B01, B03)	N/A	
RWCU pump room 1 (A) / 2 (B)	165° F (P680/1A/A02 and B02)	N/A	

Deleted: 144

Bases:

EAL BASES

LOSS – An unisolable Main Steam Line break represents a loss of the reactor coolant system barrier. This EAL is included for consistency with the ALERT emergency classification.

The leak is NOT isolable from the Main Control Room OR an attempt for isolation from the Main Control Room panels has been made and was not successful. An attempt for isolation should be made prior to the accident classification. If isolable upon identification, this INITIATING CONDITION is not applicable. Dispatch of operators outside the Control Room for manual attempts to close the valve is not considered.

POTENTIAL LOSS - A reactor coolant system leak rate of greater than 50 gallons per minute is at a level indicative of a small breach of the RCS but which is well within makeup capability of normal and emergency high pressure systems. Core uncovery is not a significant concern for a 50 gpm leak; however, break propagation leading to a significantly larger loss of inventory is possible. A leak of this size is a precursor of the loss of the reactor coolant system integrity and is therefore considered to be a potential loss of this barrier.

If the leak detection system leak rate information is unavailable (i.e., LOCA isolation, loss of power), other indicators of RCS leakage should be used. Other indications include a rise in drywell temperature and pressure and a rise in the drywell radiation monitors. If the leakage computer is unavailable, sump level and pump status may help determine if greater than 50 gpm.

If the DFR discharge line containment isolation valves have not isolated and a pump is running continuously without lowering sump level, the leakage may be assumed to exceed 50 gpm. The second pump can be started to verify that the first pump is not degraded. It is not intended to conclude a potential loss of the RCS barrier if both pumps are degraded and the observed leak rate as noted by rate of rise of level in the sump or calculated by the computer is such that it clearly confirms leakage below 50 gpm.

A VALID indication of area temperature(s) greater than or equal to the system MOV Tech Spec isolation value or area radiation level(s) greater than or equal to the monitor high alarm resulting from a primary system discharging into the Auxiliary Building is indicative of conditions in which significant RCS inventory is being lost. This is therefore considered to be a potential loss of the reactor coolant system boundary. The area radiation values are consistent with the EOP maximum normal operating values. The alarms for the high area ambient temperature are associated with the TS 3.3.6.1 allowable values for primary containment isolation. The T.S allowable high ambient temperature setpoints are set low enough to detect a leak equivalent to 25 gpm. The use of an isolation alarm and EOP condition provides a method of rapid identification of the EAL condition without the task of obtaining values from a back panel indication.

References:

FISSION PRODUCT BARRIER

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOUE
FPB Loss / Potential Loss	**FG1** 1 2 3 4 5 D Loss of any two barriers and loss or potential loss of third barrier **Emergency Action Level(s):** (1) 1. Loss of any two barriers **AND** Loss or potential loss of third barrier	**FS1** 1 2 3 4 5 D Loss or potential loss of any two barriers **Emergency Action Level(s):** (1) 1. Loss or potential loss of any two barriers	**FA1** 1 2 3 4 5 D Any loss or any potential loss of either fuel clad or RCS **Emergency Action Level(s):** (1) 1. Any loss or any potential loss of fuel clad **OR** Any loss or any potential loss of RCS	**FU1** 1 2 3 4 5 D Any loss or any potential loss of containment **Emergency Action Level(s):** (1) 1. Any loss or any potential loss of containment

FUEL CLAD (FC) Barrier			REACTOR COOLANT SYSTEM (RC) Barrier			PRIMARY CONTAINMENT (PC) Barrier		
Parameter	Loss	Potential Loss	Parameter	Loss	Potential Loss	Parameter	Loss	Potential Loss
FC1 Primary coolant activity level	Coolant activity greater than 300 μCi/gm dose equivalent I-131	None	RC1 Drywell pressure	Drywell pressure greater than 1.68 psid with indications of reactor coolant leak in drywell.	None	PC1 Primary containment pressure	Rapid unexplained loss of pressure following initial pressure rise **OR** PC pressure response not consistent with LOCA conditions	PC pressure 15 psig and rising **OR** PC hydrogen in the unsafe zone of HDOL curve **OR** DW hydrogen concentration greater than 9%
FC2 Reactor vessel water level	RPV water level less than -186 inches	RPV water level less than -162 inches	RC2 Reactor vessel water level	RPV water level less than -162 inches with indication of reactor coolant leak in drywell	None	PC2 Reactor vessel water level	None	Entry into PC flooding procedures SAP-1 and SAP-2
FC3 Primary Containment radiation monitors	Containment radiation monitor RMS-RE16 reading greater than 3,000 R/hr	None	RC3 RCS Leak Rate	Unisolable main steam line break as indicated by the failure of both MSIVs in any one line to close **AND** High MSL flow annunciator (P601-19A-A2) **AND** Main Steam Tunnel Temperature greater than 173°F	RCS leakage greater than 50 gpm inside the drywell **OR** Unisolable primary system leak outside PC as indicated by any area temperature alarm or area radiation level alarm in Table F2	PC3 Primary containment isolation failure or bypassed	Failure of both valves in any one line to close and downstream pathway to the environment exists **OR** Intentional venting per EOPs or SAPs **OR** Unisolable RCS leakage outside PC as indicated by any area temperature or area radiation level in Table F1	None
			RC4 Drywell radiation	Drywell radiation monitor RMS-RE20 reading greater than 100 R/hr due to reactor coolant leakage	None	PC4 Significant radioactive inventory in primary containment	None	Containment radiation monitor RMS-RE16 reading greater than 10,000 R/hr
FC4 ED judgment	Any condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier	Any condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier	RC5 ED judgment	Any condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	Any condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	PC5 ED judgment	Any condition in the opinion of the Emergency Director that indicates loss of the primary containment barrier	Any condition in the opinion of the Emergency Director that indicates potential loss of the primary containment barrier

Plant Modes (white boxes indicate applicable modes) 1 Power Operations 2 Startup 3 Hot Shutdown 4 Cold Shutdown 5 Refuel D Defueled

FISSION PRODUCT BARRIER

TABLE F1

PC 3 Loss of Primary Containment

Parameter	Area Temperature Max Safe Operating Value	Area Radiation Level DRMS Grid 2	Area Radiation Level Max Safe Operating Value
RHR A equipment area	200° F	1213	9.5E+03 mR/hr
RHR B equipment area	200° F	1214	9.5E+03 mR/hr
RHR C equipment area	N/A	1215	9.5E+03 mR/hr
RCIC room	200° F	1219	9.5E+03 mR/hr
MSL Tunnel	200° F	N/A	
RWCU pump room 1 (A) / 2 (B)	200° F	N/A	

TABLE F2

RC 3 Potential Loss of RCS

Parameter	Area Temperature (isolation temperature alarm)	Area Radiation Level DRMS Grid 2	Area Radiation Level Max Normal Operating Value
RHR A equipment area	117° F (P601-20A-B4)	1213	8.2E+01 mR/hr
RHR B equipment area	117° F (P601-20A-B4)	1214	8.2E+01 mR/hr
RHR C equipment area	N/A	1215	8.2E+01 mR/hr
RCIC room	182° F (P601-21A-B6)	1219	1.20E+02 mR/hr
MSL Tunnel	173° F (P601-19A-A1/A3/B1/B3)	N/A	
RWCU pump room 1 (A) / 2 (B)	165° F (P680-1A-A2/B2)	N/A	

EAL BASES

RC3

REACTOR COOLANT SYSTEM
Emergency Action Level:

RCS leak rate

EAL Threshold:

LOSS: Unisolable main steam line break as indicated by the failure of both MSIVs in any one line to close

AND

High MSL flow annunciator (P601-19A-A2)

AND

Main Steam Tunnel Temperature greater than 173° F

POTENTIAL LOSS: RCS leakage greater than 50 gpm inside the drywell

OR

Unisolable primary system leak outside PC as indicated by any area temperature alarm or area radiation level alarm in Table F2

TABLE F2			
RC 3 Potential Loss of RCS			
Parameter	Area Temperature (isolation temperature alarm)	Area Radiation Level DRMS Grid 2	Max Normal Operating Value
RHR A equipment area	117° F (P601/20A/B04)	1213	8.2 E+01 mR/hr
RHR B equipment area	117° F (P601/20A/B04)	1214	8.2 E+01 mR/hr
RHR C equipment area	N/A	1215	8.2 E+01 mR/hr
RCIC room	182° F (P601/21A/B06)	1219	1.20 E+02 mR/hr
MSL Tunnel	173° F (P601/19A/A01, A03, B01, B03)	N/A	
RWCU pump room 1 (A) / 2 (B)	165° F (P680/1A/A02 and B02)	N/A	

Bases:

Attachment 5

RBG-47157

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
In addition to the identified changes to the Technical Specifications above, the BASES will be revised upon implementation to include the following information based upon TSTF-493 BASES revision to NUREG-1434 , SR 3.3.6.1.5 to include the following statement; There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.	X		Implementation