AmerGen Energy Company, LLC
200 Exelon Way
Suite 345
Kennett Square, PA 19348

www.exeloncorp.com

AmerGen.[SM]
An Exelon/British Energy Company

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10 CFR 50.90

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Request for License Amendment Related to Revision of Instrument Channel Trip Setpoint Allowable Values

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit.", AmerGen Energy Company (AmerGen), LLC hereby requests an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for Clinton Power Station (CPS). The proposed changes are requested to make revisions to instrument channel trip setpoint Allowable Values. Specifically, the changes revise the Allowable Values for thirteen (13) TS defined functions. The affected functions are identified in Attachment 1 to this submittal.

The current Allowable Values were determined to require revision during a detailed review of all CPS instrumentation setpoints and allowable values. The revisions are needed to remedy a previously identified and corrected concern regarding the control and quality of calculations for instrumentation setpoints and Allowable Values. As a result of the concern, an extensive program was implemented to identify the full scope of the issue and to revise the associated calculations as required. Administrative controls consistent with the guidelines of NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," were established to assure that operability of systems and components would continue to be met. These administrative controls consisted of placing into effect conservative administrative limits for use in plant procedures as a short-term corrective action to ensure that the associated setpoint is maintained within the required Allowable Value. These administrative controls ensure that adequate margins to the design or accident or transient analyses assumptions are maintained until the final supporting calculations and changes to the TS are approved as the long-term corrective action. The necessary calculations required to support revisions to the affected Allowable Values were completed and therefore, the proposed changes to the TS are requested.

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The information supporting the proposed TS changes is subdivided as follows.

1. Attachment 1 provides our evaluation supporting the proposed changes.
2. Attachment 2 contains the copies of the marked up TS pages.

The proposed TS changes have been reviewed by the CPS Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the Quality Assurance Program. AmerGen is requesting approval of this change by October 31, 2004.

AmerGen is notifying the State of Illinois of this application for amendment by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions or require additional information, please contact Mr. Timothy A. Byam at (630) 657-2804.

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

11/11/03
Executed on

Keith R. Jury
Keith R. Jury
Director – Licensing and Regulatory Affairs
AmerGen Energy Company, LLC

Attachments:
1. Evaluation of Proposed Changes
2. Markup of Proposed Technical Specification Page Changes

Subject: Request for License Amendment Related to Revision of Instrument Channel Trip Setpoint Allowable Values

1.0 DESCRIPTION

2.0 PROPOSED CHANGES

3.0 BACKGROUND

4.0 TECHNICAL ANALYSIS

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

5.2 Applicable Regulatory Requirements/Criteria

6.0 ENVIRONMENTAL CONSIDERATION

7.0 REFERENCES

1.0 DESCRIPTION

This letter is a request to amend Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for Clinton Power Station (CPS). The proposed changes would revise several CPS TS instrument channel trip setpoint Allowable Values. Specifically, the changes revise the Allowable value for the following TS defined functions.

- Main Steam Isolation Valve – Closure
- Anticipated Transient Without Scram Recirculation Pump Trip Reactor Steam Dome Pressure – High
- Reactor Vessel Pressure – Low (Injection Permissive)
- Reactor Vessel Water Level – Low Low Low, Level 1
- Reactor Vessel Water Level – Low Low, Level 2
- High Pressure Core Spray (HPCS) System Reactor Vessel Water Level – High, Level 8
- Reactor Core Isolation Cooling (RCIC) Storage Tank Level – Low
- HPCS System Suppression Pool Water Level – High (Pump Suction Transfer)
- Automatic Depressurization System (ADS) Initiation Permissive, Low Pressure Core Spray (LPCS) Pump Discharge Pressure – High
- ADS Initiation Permissive, Low Pressure Coolant Injection (LPCI) Pumps Discharge Pressure – High
- RCIC System Suppression Pool Water Level – High (Pump Suction Transfer)
- Main Steam Line Pressure – Low, and
- Safety Relief Valve (SRV) Relief and Low-Low Set functions channel calibration surveillance requirement

2.0 PROPOSED CHANGES

2.1 TS Section 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Table 3.3.1.1-1, Function 6 specifies an Allowable Value for the Main Steam Isolation Valve (MSIV) – Closure of "≤ 12% closed." Function 6 is being revised to reflect an Allowable Value of "≤ 13% closed."

2.2 TS Section 3.3.4.2, "Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation," Surveillance Requirement (SR) 3.3.4.2.4.b requires performance of a channel calibration on the Reactor Steam Dome Pressure – High function. This SR specifies an Allowable Value of "≤ 1150 psig." This Allowable Value is being revised to "≤ 1143 psig."

2.3 TS Section 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation," Table 3.3.5.1-1

2.3.1 Function 1.a specifies an Allowable Value for the Reactor Vessel Water Level – Low Low Low, Level 1 of "≥ -147.7 inches." Function 1.a, is revised to reflect an Allowable Value of "≥ -148.1 inches."

2.3.2 Function 1.d specifies an Allowable Value for the Reactor Vessel Pressure – Low (Injection Permissive) of "≥ 452 psig and ≤ 478 psig." Function 1.d is being revised to reflect an Allowable Value of "≥ 454 psig and ≤ 494 psig."

2.3.3 Function 2.a specifies an Allowable Value for the Reactor Vessel Water Level – Low Low Low, Level 1 of "≥ -147.7 inches." Function 2.a, is revised to reflect an Allowable Value of "≥ -148.1 inches."

2.3.4 Function 2.d specifies an Allowable Value for the Reactor Vessel Pressure – Low (Injection Permissive) of "≥ 452 psig and ≤ 478 psig." Function 2.d is being revised to reflect an Allowable Value of "≥ 454 psig and ≤ 494 psig."

2.3.5 Function 3.a specifies an Allowable Value for the Reactor Vessel Water Level – Low Low, Level 2 of "≥ -47.7 inches." Function 3.a is revised to reflect an Allowable Value of "≥ -48.1 inches."

2.3.6 Function 3.c specifies an Allowable Value for the Reactor Vessel Water Level – High, Level 8 of "≤ 54.2 inches." Function 3.c is revised to reflect an Allowable Value of "≤ 54.6 inches."

2.3.7 Function 3.d specifies an Allowable Value for the RCIC Storage Tank Level – Low of "≥ 2.5 inches." Function 3.d is revised to reflect an Allowable Value of "≥ 3.0 inches."

2.3.8 Function 3.e specifies an Allowable Value for the Suppression Pool Water Level – High of "≤ 12 inches." Function 3.e is revised to reflect an Allowable Value of "≤ 11 inches."

2.3.9 Function 4.a specifies an Allowable Value for the Reactor Vessel Water Level – Low Low Low, Level 1 of "≥ -147.7 inches." Function 4.a, is revised to reflect an Allowable Value of "≥ -148.1 inches."

2.3.10 Function 4.e specifies an Allowable Value for the LPCS Pump Discharge Pressure – High of "≥ 125 psig." Function 4.e is revised to reflect an Allowable Value of "≥ 125 psig and ≤ 176.3 psig."

2.3.11 Function 4.f specifies an Allowable Value for the LPCI Pump A Discharge Pressure – High of "≥ 115 psig." Function 4.f is revised to reflect an Allowable Value of "≥ 115 psig and ≤ 135 psig."

2.3.12 Function 5.a specifies an Allowable Value for the Reactor Vessel Water Level – Low Low Low, Level 1 of "≥ -147.7 inches." Function 5.a, is revised to reflect an Allowable Value of "≥ -148.1 inches."

2.3.13 Function 5.e specifies an Allowable Value for the LPCI Pumps B & C Discharge Pressure – High of "≥ 115 psig." Function 5.e is revised to reflect an Allowable Value of "≥ 115 psig and ≤ 135 psig."

2.4 TS Section 3.3.5.2, "Reactor Core Isolation Cooling (RCIC) System Instrumentation," Table 3.3.5.2-1

2.4.1 Function 1 specifies an Allowable Value for the Reactor Vessel Water Level – Low Low, Level 2 of "≥ -47.7 inches." Function 1 is revised to reflect an Allowable Value of "≥ -48.1 inches."

2.4.2 Function 3 specifies an Allowable Value for the RCIC Storage Tank Level – Low of "≥ 2.5 inches." Function 3 is revised to reflect an Allowable Value of "≥ 3.0 inches."

2.4.3 Function 4 specifies an Allowable Value for the Suppression Pool Water Level – High of "≤ -3 inches." Function 4 is revised to reflect an Allowable Value of "≤ -5 inches."

2.5 TS Section 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation," Table 3.3.6.1-1

2.5.1 Function 1.a specifies an Allowable Value for the Main Steam Line Isolation Reactor Vessel Water Level – Low Low Low, Level 1 of "≥ -147.7 inches." Function 1.a, is revised to reflect an Allowable Value of "≥ -148.1 inches."

2.5.2 Function 1.b specifies an Allowable Value for the Main Steam Line Pressure – Low of "≥ 837 psig." Function 1.b is being revised to reflect an Allowable Value of "≥ 840 psig."

2.5.3 Function 2.a specifies an Allowable Value for the Primary Containment and Drywell Isolation Reactor Vessel Water Level – Low Low, Level 2 of "≥ -47.7 inches." Function 2.a is revised to reflect an Allowable Value of "≥ -48.1 inches."

2.5.4 Function 2.e specifies an Allowable Value for the Reactor Vessel Water Level – Low Low, Level 2 (HPCS NSPS Div 3 and 4) of "≥ -47.7 inches." Function 2.e is revised to reflect an Allowable Value of "≥ -48.1 inches."

2.5.5 Function 2.j specifies an Allowable Value for the Primary Containment and Drywell Isolation Reactor Vessel Water Level – Low Low Low, Level 1 of "≥ -147.7 inches." Function 2.j, is revised to reflect an Allowable Value of "≥ -148.1 inches."

2.5.6 Function 3.h specifies an Allowable Value for the RCIC System Isolation Reactor Vessel Water Level – Low Low, Level 2 of "≥ -47.7 inches." Function 3.h is revised to reflect an Allowable Value of "≥ -48.1 inches."

2.5.7 Function 4.f specifies an Allowable Value for the Reactor Water Cleanup (RWCU) System Isolation Reactor Vessel Water Level – Low Low, Level 2 of "≥ -47.7 inches." Function 4.f is revised to reflect an Allowable Value of "≥ -48.1 inches."

2.5.8 Function 5.d specifies an Allowable Value for the Residual Heat Removal (RHR) System Isolation Reactor Vessel Water Level – Low Low Low, Level 1 of "≥ -147.7 inches." Function 5.d, is revised to reflect an Allowable Value of "≥ -148.1 inches."

2.6 TS Section 3.3.6.2, "Secondary Containment Isolation Instrumentation," Table 3.3.6.2-1, Function 1 specifies an Allowable Value for the Reactor Vessel Water Level – Low Low, Level 2 of "≥ -47.7 inches." Function 1 is revised to reflect an Allowable Value of "≥ -48.1 inches."

2.7 TS Section 3.3.6.3, "Residual Heat Removal (RHR) Containment Spray System Instrumentation," Table 3.3.6.3-1, Function 3 specifies an Allowable Value for the Reactor Vessel Water Level – Low Low Low, Level 1 of "≥ -147.7 inches." Function 3, is revised to reflect an Allowable Value of "≥ -148.1 inches."

2.8 TS Section 3.3.6.4, "Suppression Pool Makeup (SPMU) System Instrumentation," Table 3.3.6.4-1, Function 2 specifies an Allowable Value for the Reactor Vessel Water Level – Low Low Low, Level 1 of "≥ -147.7 inches." Function 2, is revised to reflect an Allowable Value of "≥ -148.1 inches."

2.9 TS Section 3.3.6.5, "Relief and Low-Low Set (LLS) Instrumentation," Surveillance requirement (SR) 3.3.6.5.3 requires performance of a channel calibration of the safety relief valve (SRV) Relief and LLS functions. Items a. and b. provide the Allowable Values for these functions. The proposed change revises the Relief and LLS Allowable Values as follows.

<u>From:</u>

a. Relief Function

Low:	1103 ± 15 psig
Medium:	1113 ± 15 psig
High:	1123 ± 15 psig

b. LLS Function

Low	open:	1033 ± 15 psig
	close:	926 ± 15 psig
Medium	open:	1073 ± 15 psig
	close	936 ± 15 psig
High	open:	1113 ± 15 psig
	close:	946 ± 15 psig

To:

a. Relief Function

Low:	≤ 1118 psig
Medium:	≤ 1128 psig
High:	≤ 1138 psig

b. LLS Function

Low	open:	≤ 1044 psig
	close:	≤ 937 psig
Medium	open:	≤ 1084 psig
	close:	≤ 947 psig
High	open:	≤ 1124 psig
	close:	≤ 957 psig

In summary, AmerGen proposes to revise a number of CPS TS instrument channel trip setpoint allowable values. Specifically, the changes revise the Allowable Values for the following TS defined functions.

- Main Steam Isolation Valve – Closure
- Anticipated Transient Without Scram Recirculation Pump Trip Reactor Steam Dome Pressure – High
- Reactor Vessel Pressure - Low (Injection Permissive)
- Reactor Vessel Water Level - Low Low Low, Level 1
- Reactor Vessel Water Level – Low Low, Level 2
- HPCS System Reactor Vessel Water Level – High, Level 8
- RCIC Storage Tank Level – Low
- HPCS System Suppression Pool Water Level – High (Pump Suction Transfer)
- ADS Initiation Permissive, LPCS Pump Discharge Pressure – High
- ADS Initiation Permissive, LPCI Pumps Discharge Pressure – High
- RCIC System Suppression Pool Water Level – High (Pump Suction Transfer)
- Main Steam Line Pressure – Low, and
- SRV Relief and Low-Low Set functions channel calibration surveillance requirement.

3.0 BACKGROUND

The current Allowable Values have been determined to require revision during a detailed review of all CPS instrumentation setpoint and Allowable Values. The revisions are corrective measures needed to remedy a previously identified and corrected concern regarding control and quality of calculations for instrumentation setpoints and Allowable Values. As a result of the concern, an extensive program was implemented to identify the full scope of the issue and to revise the associated calculations as required. The proposed Allowable Values included in this amendment request represent the full scope of TS changes required as a result of that program.

Administrative controls consistent with the guidelines of NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," were established by AmerGen Energy Company (AmerGen), LLC to assure that operability of systems and components would continue to be met while the required changes were completed. These administrative controls consisted of placing into effect conservative administrative limits for use in plant procedures as a short-term corrective action to ensure that the associated setpoint is maintained within the required Allowable Value. These administrative controls ensure that adequate margins to the design or accident or transient analyses assumptions are maintained until the final supporting calculations and changes to the TS are approved as the long-term corrective action.

New Allowable Values were calculated in accordance with the guidance provided in Regulatory Guide 1.105, "Instrument Setpoints," (Reference 1) as implemented by the CPS Setpoint Methodology (Reference 2) and the Instrument Society of America (ISA) Standard S67.04, "Setpoints for Nuclear Safety Related Instrumentation," Parts I and II, dated September 1994 (Reference 3). These calculations determined the instrument uncertainties, setpoint, and Allowable Value for the affected function. The Allowable Values were determined in a manner that has been reviewed and approved as suitable to establish limits for its application. As such, the revised Allowable Values ensure that sufficient margins exist in the applicable safety analyses to confirm the affected instruments are capable of performing their intended design function.

4.0 TECHNICAL ANALYSIS

4.1. Main Steam Isolation Valve – Closure

The setpoint for the Main Steam Isolation Valve – Closure function was initially calculated for CPS by General Electric (GE) in the absence of a plant-specific calculation. AmerGen subsequently prepared a calculation to provide a site-specific basis for the Reactor Protection System Instrumentation Main Steam Isolation Valve – Closure function. The objective of this calculation is to determine the instrument uncertainty, setpoint, and Allowable Value for the Main Steam Isolation Valve – Closure trip function.

The new Allowable Value has been calculated in accordance with the guidance provided in Reference 1, as implemented in References 2 and 3. These calculations determine the instrument uncertainties, setpoint, and Allowable Value for the MSIV trips. The Allowable Value is derived from an analytic limit corrected for process and instrument errors. As such, the Allowable Value has been determined in a manner that has been reviewed and approved as suitable to establish limits for establishing emergency core cooling.

MSIV closure results in loss of the main turbine and the condenser as a heat sink for the Nuclear Steam Supply System and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on a MSIV closure signal before the MSIVs are completely closed in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. The MSIV – Closure Allowable Value is specified to ensure that a

scram occurs prior to a significant reduction in steam flow, thereby reducing the severity of the subsequent pressure transient. Implementing the proposed Allowable Value has no effect on the design basis of the Reactor Protection System MSIV – Closure trip. The Allowable Value is calculated to ensure that the associated safety analysis limit is not exceeded due to instrumentation uncertainties.

4.2. Anticipated Transient Without Scram Recirculation Pump Trip Reactor Steam Dome Pressure – High

AmerGen prepared a calculation to provide a plant-specific basis for the Reactor Steam Dome Pressure High – Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Allowable Value. The objective of the calculation was to determine the instrument uncertainty, Allowable Value, and setpoint for the ATWS-RPT System Reactor Pressure Vessel (RPV) Steam Dome Pressure.

The new Allowable Value has been calculated in accordance with the guidance provided in Reference 1 as implemented in References 2 and 3. The Allowable Value is derived from an analytic limit corrected for process and instrument errors. As such, the Allowable Value has been determined in a manner that has been reviewed and approved as suitable to establish limits for establishing a diverse trip to mitigate the consequences of a postulated ATWS event.

To mitigate the potential consequences of a postulated ATWS event, a non-safety related recirculation pump trip subsystem is provided as part of the ATWS system. This subsystem functions independently of the safety related recirculation pump trip system by providing an alternate means of tripping the recirculation pump motors and the low frequency motor/generator (LFMG) sets. The ATWS-RPT function is initiated by either RPV high pressure or RPV low level. An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and thermal power transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the reactor coolant pressure boundary. Implementing the proposed allowable value has no effect on the design basis of the Reactor Steam Dome Pressure – High, ATWS-RPT. The Allowable Value is calculated to ensure that the associated safety analysis limit is not exceeded due to instrumentation uncertainties.

4.3. Reactor Vessel Water Level – Low Low Low, Level 1

The setpoint for the Reactor Vessel Water Level – Low Low Low, Level 1 function was initially calculated by General Electric in the absence of a formal CPS calculation. AmerGen has subsequently completed calculations to provide a plant-specific basis for the required Reactor Vessel Water Level – Low Low Low, Level 1 trips. These trips are associated with the Low Pressure Coolant Injection (LPCI) A, B, and C; Low Pressure Core Spray (LPCS); Automatic Depressurization System (ADS) Trip Systems 1 and 2; Main Steam Line Isolation; Primary Containment and Drywell Isolation; Residual Heat Removal

(RHR) System Isolation; RHR Containment Spray; and Suppression Pool Makeup subsystems.

The new Allowable Value has been calculated in accordance with the guidance provided in Reference 1 as implemented by References 2 and 3. These calculations determine the instrument uncertainties, setpoint, and Allowable Value for the Reactor Vessel Water Level – Low Low Low, Level 1 trips. The Allowable Value is derived from an analytic limit corrected for calibration, process and instrument errors. As such, the Allowable Value has been determined in a manner that has been reviewed and approved as suitable to determine limits for establishing emergency core cooling.

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The low pressure ECCS are initiated at Level 1 to ensure core spray and flooding functions are available to prevent or minimize fuel damage. ADS receives one of the signals necessary for initiation from this function and isolation of MSIVs, other RPV interfaces, and primary containment occurs at Level 1 to prevent offsite dose limits from being exceeded. The RHR Containment Spray System is initiated at Level 1 to mitigate the consequences of steam leaking from the drywell directly into the containment airspace, thus bypassing the suppression pool. In addition, the Level 1 signal is used to dump water from the upper containment pool into the suppression pool to ensure the LOCA vents remain covered. The Reactor Vessel Water Level – Low Low Low, Level 1 Allowable Value is high enough to allow for the low-pressure core flooding systems to initiate and provide adequate cooling. The Level 1 Allowable Values for the other functions are chosen to be the same as the ECCS Level 1 Allowable Value to ensure that these functions (i.e., containment isolation, containment spray initiation and suppression pool makeup) are completed in conjunction with the ECCS functions. Implementation of the proposed Allowable Value has no effect on the design basis of the Reactor Vessel Water Level – Low Low Low, Level 1 trips. The Allowable value has been calculated to ensure that the associated safety limits are not exceeded due to instrumentation uncertainties.

4.4. Reactor Vessel Pressure – Low

General Electric initially calculated the Reactor Vessel Pressure – Low function setpoint in the absence of a formal CPS calculation. AmerGen has subsequently completed calculations to provide a plant-specific basis for the required Reactor Vessel Pressure – Low injection permissive signal. These permissive signals are associated with the injection permissive for the LPCS and LPCI A, B, and C subsystems. The objective of this calculation is to determine the instrument uncertainty, setpoint, and Allowable Value for the permissive.

The new Allowable Value was calculated in accordance with the guidance provided in Reference 1 as implemented in References 2 and 3. The Allowable Value is derived from an analytic limit corrected for process and instrument errors. As such, the Allowable Value has been determined in a manner that has

been reviewed and approved as suitable to establish limits for establishing emergency core cooling.

Low reactor vessel pressure signals are used as a permissive for the low-pressure ECCS subsystems. This ensures that, prior to opening the injection valves of the low-pressure ECCS subsystems, the reactor pressure has fallen to a value below the maximum design pressure for these subsystems. The Reactor Vessel Pressure – Low function is one of the functions assumed to be operable and capable of permitting initiation of ECCS during the transients analyzed in CPS Updated Safety Analysis Report (USAR) Section 5.2.2 and USAR Chapter 15, "Safety Analyses." Implementing the proposed allowable values has no effect on the design bases of the LPCS and LPCI A, B, and C subsystems Reactor Vessel Pressure – Low injection permissive. The Allowable Value has been calculated to ensure that the associated safety analysis limit is not exceeded due to instrumentation uncertainties.

4.5. Reactor Vessel Water Level – Low Low, Level 2

The setpoint for the Reactor Vessel Water Level – Low Low, Level 2 function was initially calculated by General Electric in the absence of a formal CPS calculation. AmerGen has subsequently completed calculations to provide a plant-specific basis for the required Reactor Vessel Water Level – Low Low, Level 2 trips. These trips are associated with the High Pressure Core Spray (HPCS) System; Reactor Core Isolation Cooling (RCIC) System; Primary Containment and Drywell Isolation; Primary Containment and Drywell Isolation (HPCS NSPS Division 3 and 4); RCIC System Isolation; Reactor Water Cleanup (RWCU) System Isolation; and Secondary Containment Isolation subsystems.

The new Allowable Value has been calculated in accordance with the guidance provided in Reference 1 as implemented by References 2 and 3. These calculations determine the instrument uncertainties, setpoint, and Allowable Value for the Reactor Vessel Water Level – Low Low, Level 2 trips. The Allowable Value is derived from an analytic limit corrected for calibration, process and instrument errors. As such, the Allowable Value has been determined in a manner that has been reviewed and approved as suitable to determine limits for establishing emergency core cooling.

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the HPCS and RCIC Systems are initiated at Level 2 to maintain level above the top of active fuel. The isolation of some reactor vessel interfaces (i.e., the RWCU System) occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 2 supports actions to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." In addition, the isolation of primary containment, isolation of the secondary containment, and actuation of the SGTS on Level 2 are initiated to minimize the potential of an offsite dose release. The Reactor Vessel Water Level – Low Low, Level 2 Allowable Value is chosen such that the

HPCS and RCIC Systems flows will be sufficient to avoid initiation of the low pressure ECCS at Reactor Vessel Water Level – Low Low Low, Level 1. The Level 2 Allowable Values for the other functions were chosen to be the same as the ECCS Reactor Vessel Water Level – Low Low, Level 2. Implementation of the proposed Allowable Value has no effect on the design basis of the Reactor Vessel Water Level – Low Low, Level 2 trips. The Allowable Value has been calculated to ensure that the associated safety limits are not exceeded due to instrumentation uncertainties.

4.6. High Pressure Core Spray (HPCS) System Reactor Vessel Water Level – High, Level 8

The setpoint for the Reactor Vessel Water Level – High, Level 8 trip was originally calculated by GE in the absence of a formal CPS calculation. AmerGen has subsequently completed a calculation to provide a plant-specific basis for the ECCS instrumentation HPCS Reactor Vessel Water Level – High, Level 8 trip.

The new Allowable Value has been calculated in accordance with the guidance provided in Reference 1 as implemented in References 2 and 3. The Allowable Value has been derived from an analytic limit corrected for calibration, process and instrument errors. As such, the Allowable Value has been determined in a manner that has been reviewed and approved as suitable to establish limits for establishing emergency core cooling.

The HPCS System provides water to the reactor until the reactor vessel water level reaches the high water level (i.e., Level 8) trip, at which time the HPCS injection valve closes. The Reactor Vessel Water Level – High, Level 8 Allowable Value is chosen to isolate flow from the HPCS system prior to water overflowing into the main steam lines. Implementing the proposed Allowable Value has no effect on the design basis of the HPCS Reactor Vessel Water Level – High, Level 8 trip. The Allowable Value has been calculated to ensure that the associated safety analysis limit is not exceeded due to instrumentation uncertainties.

4.7. RCIC Storage Tank Level - Low

General Electric initially calculated the setpoint for the RCIC Storage Tank Level – Low function in the absence of a formal CPS calculation. AmerGen subsequently completed calculations to provide a plant-specific basis for the required RCIC Storage Tank Level – Low function. This function is associated with the HPCS and RCIC systems.

The new Allowable Value has been calculated in accordance with the guidance provided in Reference 1 as implemented in References 2 and 3. The Allowable Value has been derived from an analytic limit corrected for process and instrument errors. As such, the Allowable Value has been determined in a manner that has been reviewed and approved as suitable to establish limits for establishing emergency core cooling.

Low level in the RCIC storage tank indicates the unavailability of an adequate supply of makeup water from this normal source. Normally, the suction valves between the RCIC storage tank and the RCIC and HPCS systems are open and, upon receiving an initiation signal, water for HPCS or RCIC injection would be taken from the RCIC storage tank. However, if the level in the tank falls below a preselected level, the suppression pool suction valve opens and then the RCIC storage tank suction valve closes. This ensures that an adequate supply of makeup water is available to the RCIC and HPCS pumps. The RCIC Storage Tank Level – Low function Allowable Value is high enough to ensure adequate pump suction head while water is being taken from the RCIC storage tank. Implementing the proposed Allowable Value has no effect on the design basis of the RCIC Storage Tank Level – Low function. The Allowable Value has been calculated to ensure that the associated safety analysis limit is not exceeded due to instrumentation uncertainties.

4.8. High Pressure Core Spray (HPCS) System Suppression Pool Water Level – High

The setpoint for the HPCS System Suppression Pool Water Level – High was initially calculated by General Electric in the absence of a formal CPS calculation. AmerGen has subsequently completed a calculation to provide a plant-specific basis for the required HPCS System Suppression Pool Water Level – High. The objective of this calculation is to determine the instrument uncertainty, setpoint, and Allowable Value for the Suppression Pool Water Level – High.

The new Allowable Value has been calculated in accordance with the guidance provided in Reference 1 as implemented by References 2 and 3. The Allowable Value is derived from an analytic limit corrected for process and instrument errors. As such, the Allowable Value has been determined in a manner that has been reviewed and approved as suitable to determine limits for establishing emergency core cooling.

The HPCS System provides makeup water to the reactor until the high water level (Level 8) trip, at which time the HPCS injection valve closes. The injection valve automatically reopens if a low, low water level is subsequently received. The HPCS System monitors the water levels in the RCIC storage tank and the suppression pool, since these are the two sources of water for HPCS operation. Reactor grade water in the RCIC storage tank is the normal and preferred source. However, excessively high suppression pool water level could result in loads on the suppression pool exceeding design values should there be a blowdown of the reactor vessel pressure through the SRVs. Therefore, signals indicating high suppression pool water level are used to transfer the suction source of HPCS from the RCIC storage tank to the suppression pool to eliminate the possibility of HPCS continuing to provide additional water from a source outside containment. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valve must be open before the RCIC storage tank suction valve automatically closes. The Allowable Value for the Suppression Pool Water Level – High function is chosen to ensure that the HPCS system will be aligned for suction from the suppression pool before the

water level reaches the point at which suppression pool design loads would be exceeded. Implementation of the proposed Allowable Value has no effect on the design basis of the HPCS System Suppression Pool Water Level – High. The Allowable Value has been calculated to ensure that the associated safety limits are not exceeded due to instrumentation uncertainties.

4.9. Automatic Depressurization System (ADS) Trip System 1 (Logic A and E) – LPCS Pump Discharge Pressure – High

The setpoint for the ADS Trip System 1 (Logic A and E) – LPCS Pump Discharge Pressure - High was initially calculated by General Electric in the absence of a formal CPS calculation. AmerGen has subsequently completed a calculation to provide a plant-specific basis for the required ADS Trip System 1 – LPCS Pump Discharge Pressure - High function. The objective of this calculation is to determine the instrument uncertainty, setpoint and Allowable Values for the permissive.

The new Allowable Value has been calculated in accordance with the guidance provided in Reference 1 as implemented by References 2 and 3. The Allowable Value is derived from an analytic limit corrected for process and instrument errors. As such, the Allowable Value has been determined in a manner that has been reviewed and approved as suitable to determine limits for establishing emergency core cooling.

The ADS utilizes a number of reactor SRVs to reduce reactor pressure in the event that the HPCS System fails. This allows the low pressure systems (LPCI and LPCS) to inject water into the vessel and protect the fuel barrier. The Pump Discharge Pressure – High signals from the LPCS and LPCI pumps are used as permissive signals for ADS initiation, indicating that there is a source of low pressure ECCS available once ADS has depressurized the reactor vessel. This function is assumed to be operable and capable of permitting ADS initiation during the events analyzed in the CPS USAR. The low pressure core cooling function, in addition to the scram function of the Reactor Protection System, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

The LPCS Pump Discharge Pressure – High Allowable Value is less than the pump discharge pressure when the pump is operating in a full flow mode, and high enough to avoid any condition that results in a discharge pressure permissive when the LPCS pump is aligned for injection and the pumps are not running. The actual operating point of this Function is not assumed in any transient or accident analysis. Implementing the proposed Allowable Values has no effect on the design bases of the ADS Trip System 1 (Logic A and E) – LPCS Pump Discharge Pressure – High permissive.

AmerGen has added an upper Allowable Value for the LPCS Pump Discharge Pressure – High permissive. This Allowable Value is based on the analytical limit defined in the GE Residual Heat Removal System Design Specification Data Sheets. Both the upper and lower analytical limits exist in the current design

bases and appropriate Allowable Values are calculated to ensure that the associated analytical limits are not exceeded due to instrumentation uncertainties.

4.10. Automatic Depressurization System (ADS) Trip Systems 1 and 2 (Logics A, B, E, and F) – LPCI Pumps A, B, & C Discharge Pressure – High

The setpoint for the ADS Trip Systems 1 and 2 – LPCI Pumps A, B & C Discharge Pressure - High was initially calculated by General Electric in the absence of a formal CPS calculation. AmerGen has subsequently completed a calculation to provide a plant-specific basis for the required ADS Trip Systems 1 and 2 – LPCI Pumps A, B, & C Discharge Pressure – High function. The objective of this calculation is to determine the instrument uncertainty, setpoint and Allowable Values for the permissive.

The new Allowable Value has been calculated in accordance with the guidance provided in Reference 1 as implemented by References 2 and 3. The Allowable Value is derived from an analytic limit corrected for process and instrument errors. As such, the Allowable Value has been determined in a manner that has been reviewed and approved as suitable to determine limits for establishing emergency core cooling.

The ADS utilizes a number of reactor SRVs to reduce reactor pressure in the event that the HPCS System fails. This allows the low pressure systems (LPCI and LPCS) to inject water into the vessel and protect the fuel barrier. The Pump Discharge Pressure – High signals from the LPCS and LPCI pumps are used as permissive signals for ADS initiation, indicating that there is a source of low pressure ECCS available once ADS has depressurized the reactor vessel. This function is assumed to be operable and capable of permitting ADS initiation during the events analyzed in the CPS USAR. The low pressure core cooling function, in addition to the scram function of the Reactor Protection System, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

The LPCI A, B, and C Pumps Discharge Pressure – High Allowable Value is less than the pump discharge pressure when the pumps are operating in a full flow mode, and high enough to avoid any condition that results in a discharge pressure permissive when the LPCI pumps are aligned for injection and the pumps are not running. The actual operating point of this Function is not assumed in any transient or accident analysis. Implementing the proposed Allowable Values has no effect on the design bases of the ADS Trip Systems 1 and 2 – LPCI Pumps A, B, & C Discharge Pressure – High permissive.

AmerGen has added an upper Allowable Value for the LPCI Pump Discharge Pressure – High permissive. This Allowable Value is based on the analytical limit defined in the GE Residual Heat Removal System Design Specification Data Sheets. Both the upper and lower analytical limits exist in the current design bases and appropriate Allowable Values are calculated to ensure that the

associated analytical limits are not exceeded due to instrumentation uncertainties.

4.11. Reactor Core Isolation Cooling (RCIC) System Suppression Pool Water Level – High

The setpoint for the RCIC System Suppression Pool Water Level – High was initially calculated by General Electric in the absence of a formal CPS calculation. AmerGen has subsequently completed a calculation to provide a plant-specific basis for the required suppression pool high water level transfer of the RCIC suction source from the RCIC storage tank to the suppression pool. The objective of this calculation is to determine the instrument uncertainty, setpoint and Allowable Values for the Suppression Pool Water Level – High.

The new Allowable Value has been calculated in accordance with the guidance provided in Reference 1 as implemented by References 2 and 3. The Allowable Value is derived from an analytic limit corrected for process and instrument errors. As such, the Allowable Value has been determined in a manner that has been reviewed and approved as suitable to determine limits for establishing emergency core cooling.

The function of the RCIC System is to provide makeup water to the reactor in response to transient events. The RCIC System provides water to the reactor until the reactor vessel water level reaches the high water level (Level 8) trip, at which time the RCIC steam supply and cooling water supply valves close. The injection valve also closes due to the closure of the steam supply valve. The RCIC System restarts if the vessel level again drops to the low-level initiation point. The RCIC System monitors the water levels in the RCIC storage tank and the suppression pool, since these are the two sources of water for RCIC operation. Reactor grade water in the RCIC storage tank is the normal source. Upon receipt of a RCIC initiation signal, the RCIC storage tank suction valve is automatically signaled to open. However, excessively high water level in the suppression pool could result in the loads on the suppression pool exceeding design values should there be a blowdown of the reactor vessel pressure through the SRVs. Therefore, signals indicating high suppression pool water level are used to transfer the suction source of RCIC from the RCIC storage tank to the suppression pool to eliminate the possibility of RCIC continuing to provide additional water from a source outside primary containment. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valve must be open before the RCIC storage tank suction valve automatically closes. The Allowable Value for the Suppression Pool Water Level – High function is set low enough to ensure that RCIC will be aligned to take suction from the suppression pool before the water level reaches the point at which suppression pool design loads would be exceeded. Implementation of the proposed Allowable Value has no effect on the design basis of the RCIC System Suppression Pool Water Level – High. The Allowable Value has been calculated to ensure that the associated safety limits are not exceeded due to instrumentation uncertainties.

4.12. Main Steam Line Pressure – Low

General Electric initially calculated the setpoint for the Main Steam Line Pressure – Low trip function in the absence of a formal CPS calculation. AmerGen subsequently completed a calculation to provide a plant-specific basis for the main steam line isolation on Main Steam Line Pressure – Low trip. The objective of this calculation is to determine the instrument uncertainty, setpoint, and Allowable Value for the Main Steam Line Pressure – Low trip.

The new Allowable Value was calculated in accordance with the guidance provided in Reference 1 as implemented in References 2 and 3. The Allowable Value is derived from an analytic limit corrected for calibration, process and instrument errors. As such, the allowable value has been determined in a manner that has been reviewed and approved as suitable to establish limits for establishing emergency core cooling.

Low main steam line pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100 degrees Fahrenheit per hour (°F/hour) if the pressure loss is allowed to continue. The Main Steam Line Pressure – Low function is directly assumed in the analysis of the pressure regulator failure. For this event, the closure of the MSIVs ensures that the RPV temperature change limit (i.e., 100 °F/hour) is not reached. In addition, this function ensures the MSIVs are closed prior to reactor pressure decreasing below 785 psig, thus ensuring TS Safety Limit 2.1.1.1 is not exceeded. The Allowable Value was selected to be high enough to prevent excessive RPV depressurization. Implementing the proposed Allowable Value has no effect on the design basis of the main steam line isolation on the Main Steam Line Pressure – Low trip. The Allowable Value has been calculated to ensure that the associated safety analysis limit is not exceeded due to instrumentation uncertainties.

4.13. Relief and Low–Low Set Functions

AmerGen has completed a calculation to provide a CPS specific basis for the Relief and Low-Low Set (LLS) Instrumentation – Relief and LLS functions. The objective of this calculation is to determine the instrument uncertainty, Allowable Value, and setpoint for the SRV relief and LLS functions. As a result of this calculation, more conservative Allowable Values have been calculated to replace the current relief and LLS functions as specified in TS SR 3.3.6.5.3.b.

The new Allowable Values have been calculated in accordance with the guidance provided in Reference 1 as implemented by References 2 and 3. The Allowable Values are derived from analytic limits corrected for process and instrument errors. As such, the Allowable Values have been determined in a manner that has been reviewed and approved as suitable to determine limits for establishing a means of ensuring reactor overpressurization does not occur.

The SRVs prevent overpressurization of the reactor coolant pressure boundary (RCPB). Instrumentation is provided to support the relief and LLS modes of SRV operation. The relief function of the SRVs prevents overpressurization of the RCPB. The LLS function of the SRVs is designed to mitigate the effects of postulated pressure loads on the containment by preventing multiple actuations in rapid succession of the SRVs subsequent to their initial actuation. The LLS logic increases the time between (or prevents) subsequent actuations to limit SRV subsequent actuations to one valve, so that containment loads will also be reduced. The relief and LLS instrumentation are designed to prevent overpressurization of the RCPB and to ensure that the containment loads remain within the primary containment design basis. Implementing the proposed Allowable Values have no effect on the design basis of the relief and LLS functions. The Allowable Values are calculated to ensure that the associated safety analysis limits are not exceeded due to instrumentation uncertainties. The safety analysis limits (i.e., analytical limits) from which the Allowable Values are derived remain the same.

5.0 REGULATORY ANALYSIS

5.1. No Significant Hazards Consideration

AmerGen Energy Company (AmerGen), LLC is requesting a revision to the Facility Operating License No. NPF-62 for Clinton Power Station (CPS), Unit 1. The proposed changes are requested to make revisions to instrument channel trip setpoint Allowable Values. Specifically, the changes revise the Allowable Values for thirteen (13) Technical Specification (TS) defined functions.

The current Allowable Values were determined to require revision during a detailed review of all CPS instrumentation setpoints and Allowable Values. The revisions are needed to remedy a previously identified and corrected concern regarding the control and quality of calculations for instrumentation setpoints and Allowable Values. As a result of the concern, an extensive program was implemented to identify the full scope of the issue and to revise the associated calculations as required. Administrative controls consistent with the guidelines of NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," were established to assure that operability of systems and components would continue to be met. These administrative controls consisted of placing into effect conservative administrative limits for use in plant procedures as a short-term corrective action to ensure that the associated setpoint is maintained within the required Allowable Value. These administrative controls ensure that adequate margins to the design or accident or transient analyses assumptions are maintained until the final supporting calculations and changes to the TS are approved as the long-term corrective action. The necessary calculations required to support revisions to the affected Allowable Values were completed and therefore, the proposed changes to the TS are requested.

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

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

Response: No.

The proposed amendment implements revised Allowable Values for the following instrument functions.

- Main Steam Isolation Valve – Closure
- Anticipated Transient Without Scram Recirculation Pump Trip Reactor Steam Dome Pressure – High
- Reactor Vessel Pressure – Low (Injection Permissive)
- Reactor Vessel Water Level – Low Low Low, Level 1
- Reactor Vessel Water Level – Low Low, Level 2
- High Pressure Core Spray (HPCS) System Reactor Vessel Water Level – High, Level 8
- Reactor Core Isolation Cooling (RCIC) Storage Tank Level – Low
- HPCS System Suppression Pool Water Level – High (Pump Suction Transfer)
- Automatic Depressurizaton System (ADS) Initiation Permissive, Low Pressure Core Spray (LPCS) Pump Discharge Pressure – High
- ADS Initiation Permissive, Low Pressure Coolant Injection (LPCI) Pumps Discharge Pressure – High
- RCIC System Suppression Pool Water Level – High (Pump Suction Transfer)
- Main Steam Line Pressure – Low, and
- Safety Relief Valve (SRV) Relief and Low-Low Set (LLS) functions channel calibration surveillance requirement

The proposed changes do not require modification to the facility. There is no impact on the accident analysis as a result of the proposed changes to the Allowable Values. The analytical limit, which is used as input to the accident analysis, does not change. The proposed changes will be implemented through revision of the associated surveillance test procedures, where the revised Allowable Value will replace the existing value.

Derivation of the Allowable Value in accordance with Regulatory Guide 1.105, "Instrument Setpoints," uses the analytical limit as a fixed starting point from which instrument uncertainties are added or subtracted, as appropriate. Calculation of the Allowable Value to plant-specific parameters provides additional confidence that protective instrumentation that passes the surveillance testing criteria will perform its design function without exceeding the associated safety analysis limit.

The revised Allowable Values for the affected equipment are not considered an initiator to any previously analyzed accident and therefore, cannot increase the probability of any previously evaluated accident. Implementation of the revised Allowable Values will ensure that the instrumentation will perform its required function to meet the accident analysis assumptions. The proposed Allowable Values will ensure that the fuel is adequately cooled, containment and drywell are isolated as required, primary containment temperature and pressure design limits are met, and overpressurization of the nuclear steam supply system is prevented following an accident or transient. The proposed changes do not increase the probability of any accident previously evaluated.

Since the proposed changes ensure the same level of protection as assumed in the accident analyses, the conclusions of the accident scenarios remain valid. As a result, no changes to radiological release parameters are involved. Therefore, the proposed changes do not increase the consequences of an accident previously evaluated.

In summary, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not affect the design, functional performance or operation of the facility. Similarly, they do not affect the design or operation of any structures, systems, or components involved in the mitigation of any accidents, nor do they affect the design or operation of any component in the facility such that new equipment failure modes are created. Setpoints remain the same and therefore, there is no impact on the operation of any of the associated systems.

As such the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes do not involve a change to the plant design or operation. The proposed changes will be implemented through revisions to the associated surveillance test procedures where the revised Allowable Value replaces the existing Allowable Value. No changes to the instrument setpoints are involved. Since the availability of the systems will be maintained and since the system designs are unaffected, the proposed changes ensure the instrumentation is capable of performing their intended functions. The proposed changes do not affect the accident analyses that assume the operability of the instrumentation associated with these Allowable Values. The margins associated with the analytical limits are not impacted by the proposed Allowable Values since the analytical limits remain unchanged.

Therefore, operation of CPS in accordance with the proposed changes will not involve a significant reduction in a margin of safety.

Conclusion

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

5.2 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. New Allowable Values have now been calculated in accordance with the guidance provided in Regulatory Guide 1.105, "Instrument Setpoints," as implemented by the CPS Setpoint Methodology, and the Instrument Society of America (ISA) Standard S67.04, "Setpoints for Nuclear Safety Related Instrumentation," Parts I and II, dated September 1994. These calculations determine the instrument uncertainties, setpoints, and allowable values for the affected functions. The Allowable Values have been determined in a manner that has been reviewed and approved as suitable to establish limits for its application. As such, the revised Allowable Values ensure that sufficient margins exist in the applicable safety analyses to confirm the affected instruments are capable of performing their intended design function. AmerGen has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the Technical Specifications, and do not affect conformance with any General Design Criteria (GDC) differently than described in the CPS USAR.

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

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not

requiring environmental review.", Paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22, Paragraph (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Regulatory Guide 1.105, "Instrument Setpoints," Revision 1, dated November 1976

2. CI-01.00, "CPS Setpoint Methodology," Revision 2

3. Instrument Society of America (ISA) S67.04, "Setpoints for Nuclear Safety Related Instrumentation," Parts I and II, dated September 1994

ATTACHMENT 2
Markup of Proposed Technical Specification Page Changes

Revised TS Pages

3.3-8
3.3-29
3.3-39
3.3-40
3.3-41
3.3-42
3.3-43
3.3-47
3.3-55
3.3-56
3.3-57
3.3-58
3.3-59
3.3-60
3.3-64
3.3-68
3.3-72
3.3-74

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. Reactor Vessel Steam Dome Pressure - High	1,2	4	H	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 1080 psig
4. Reactor Vessel Water Level-Low, Level 3	1,2	4	H	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≥ 8.3 inches
5. Reactor Vessel Water Level-High, Level 8	≥ 21.6 % RTP	4	F	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 52.6 inches
6. Main Steam Isolation Valve-Closure	1	4	G	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	13% ≤ 12% closed
7. Drywell Pressure-High	1,2	4	H	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1.88 psig
8. Scram Discharge Volume Water Level-High					
a. Transmitter	1,2	4	H	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 40-1/4 inches for 1C11-N601A,B and ≤ 39-3/16 inches for 1C11-N601C,D
	5(a)	4	I	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 40-1/4 inches for 1C11-N601A,B and ≤ 39-3/16 inches for 1C11-N601C,D

(continued)

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

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1 Remove the associated recirculation pump from service. OR C.2 Be in MODE 2.	6 hours 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.4.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.2.3	Calibrate the trip units.	92 days
SR 3.3.4.2.4	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Reactor Vessel Water Level-Low Low, Level 2: ≥ -50.0 inches; and b. Reactor Steam Dome Pressure-High: ≤ 1150 psig. 1143	18 months
SR 3.3.4.2.5	Perform LOGIC SYSTEM FUNCTIONAL TEST, including breaker actuation.	18 months

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Low Pressure Coolant Injection-A (LPCI) and Low Pressure Core Spray (LPCS) Subsystems					-148.1
a. Reactor Vessel Water Level-Low Low Low, Level 1	1,2,3, 4[a],5[a]	2[b]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -147.7 inches
b. Drywell Pressure-High	1,2,3	2[b]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.88 psig
c. LPCI Pump A Start-Time Delay Logic Card	1,2,3, 4[a],5[a]	1	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 4.5 seconds and ≤ 5.5 seconds
d. Reactor Vessel Pressure-Low (Injection Permissive)	1,2,3	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 452 psig and ≤ 478 psig 454
		494			
	4[a],5[a]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 452 psig and ≤ 478 psig
e. LPCS Pump Discharge Flow-Low (Bypass)	1,2,3, 4[a],5[a]	1	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 750 gpm
f. LPCI Pump A Discharge Flow-Low (Bypass)	1,2,3, 4[a],5[a]	1	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 900 gpm
g. Manual Initiation	1,2,3, 4[a],5[a]	1	C	SR 3.3.5.1.5	NA

(continued)

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

(b) Also required to initiate the associated diesel generator.

Table 3.3.5.1-1 (page 2 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI B and LPCI C Subsystems					
a. Reactor Vessel Water Level-Low Low Low, Level 1	1,2,3, 4[a],5[a]	2[b]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	-148.1 ≥ -147.7 inches
b. Drywell Pressure-High	1,2,3	2[b]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.88 psig
c. LPCI Pump B Start-Time Delay Logic Card	1,2,3, 4[a],5[a]	1	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 4.5 seconds and ≤ 5.5 seconds
d. Reactor Vessel Pressure-Low (Injection Permissive)	1,2,3	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 452 psig and ≤ 478 psig 454
	4[a],5[a]	4 494	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 452 psig and ≤ 478 psig
e. LPCI Pump B and LPCI Pump C Discharge Flow-Low (Bypass)	1,2,3, 4[a],5[a]	1 per pump	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 900 gpm
f. Manual Initiation	1,2,3, 4[a],5[a]	1	C	SR 3.3.5.1.5	NA

(continued)

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

(b) Also required to initiate the associated diesel generator.

Table 3.3.5.1-1 (page 3 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. High Pressure Core Spray (HPCS) System					
a. Reactor Vessel Water Level-Low Low, Level 2	1,2,3, 4[a],5[a]	4[b]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	-48.1 ≥ -47.7 inches
b. Drywell Pressure - High	1,2,3	4[b]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.88 psig
c. Reactor Vessel Water Level-High, Level 8	1,2,3, 4[a],5[a]	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	54.6 ≤ 54.2 inches
d. RCIC Storage Tank Level-Low	1,2,3, 4[c],5[c]	2	D	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	3.0 ≥ 2.5 inches
e. Suppression Pool Water Level-High	1,2,3	2	D	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 12 inches 11
f. HPCS Pump Discharge Pressure-High (Bypass)	1,2,3, 4[a],5[a]	1	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 120 psig
g. HPCS System Flow Rate-Low (Bypass)	1,2,3, 4[a],5[a]	1	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 500 gpm
h. Manual Initiation	1,2,3, 4[a],5[a]	1	C	SR 3.3.5.1.5	NA

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.
(b) Also required to initiate the associated diesel generator.
(c) When HPCS is OPERABLE for compliance with LCO 3.5.2, "ECCS-Shutdown," and aligned to the RCIC storage tank while tank water level is not within the limits of SR 3.5.2.2.

Table 3.3.5.1-1 (page 4 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Automatic Depressurization System (ADS) Trip System 1 (Logic A and E)					
a. Reactor Vessel Water Level-Low Low Low, Level 1	1,2[d],3[d]	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	-148.1 ≥ -147.7 inches
b. Drywell Pressure-High	1,2[d],3[d]	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.88 psig
c. ADS Initiation Timer	1,2[d],3[d]	1	G	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3 (Confirmatory)	1,2[d],3[d]	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 8.3 inches
e. LPCS Pump Discharge Pressure-High	1,2[d],3[d]	2	G	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 125 psig and ≤ 176.3 psig
f. LPCI Pump A Discharge Pressure-High	1,2[d],3[d]	2	G	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 115 psig and ≤ 135 psig
g. ADS Drywell Pressure Bypass Timer	1,2[d],3[d]	2	G	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 6.5 minutes
h. Manual Initiation	1,2[d],3[d]	2	G	SR 3.3.5.1.5	NA

(continued)

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

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. ADS Trip System 2 (Logic B and F)					−148.1
a. Reactor Vessel Water Level-Low Low Low, Level 1	1,2(d),3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -147.7 inches
b. Drywell Pressure-High	1,2(d),3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.88 psig
c. ADS Initiation Timer	1,2(d),3(d)	1	G	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3 (Confirmatory)	1,2(d),3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 8.3 inches
e. LPCI Pumps B & C Discharge Pressure-High	1,2(d),3(d)	2 per pump	G	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 115 psig and ≤ 135 psig
f. ADS Drywell Pressure Bypass Timer	1,2(d),3(d)	2	G	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 6.5 minutes
g. Manual Initiation	1,2(d),3(d)	2	G	SR 3.3.5.1.5	NA

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

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

FUNCTION	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level-Low Low, Level 2	4	B	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.4 SR 3.3.5.2.5	≥ -47.7 inches -48.1
2. Reactor Vessel Water Level-High, Level 8	2	C	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.4 SR 3.3.5.2.5	≤ 52.6 inches 3.0
3. RCIC Storage Tank Level-Low	2	D	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.4 SR 3.3.5.2.5	≥ 2.5 inches -5
4. Suppression Pool Water Level-High	2	D	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.4 SR 3.3.5.2.5	≤ -3 inches
5. Manual Initiation	1	C	SR 3.3.5.2.5	NA

Table 3.3.6.1-1 (page 1 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level-Low Low Low, Level 1	1,2,3	4	G	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	≥ -147.7 inches -148.1
b. Main Steam Line Pressure-Low	1	4	H	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 840
c. Main Steam Line Flow-High	1,2,3	4	G	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	≤ 284 psid
d. Condenser Vacuum-Low	1,2[a], 3[a]	4	G	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	4	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 171°F
f. Main Steam Line Turbine Building Temperature-High	1,2,3	4	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	Modules 1-4 ≤ 142°F, Module 5 ≤ 150°F
g. Manual Initiation	1,2,3	4	J	SR 3.3.6.1.6	NA

(continued)

(a) With any turbine stop valve not closed.

Table 3.3.6.1-1 (page 2 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment and Drywell Isolation					
a. Reactor Vessel Water Level-Low Low, Level 2	1,2,3	4[b]	K	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	≥ -47.7 inches -48.1
	[c]	4	O	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	≥ -47.7 inches
b. Drywell Pressure-High	1,2,3	4[b]	K	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 psig
c. Deleted					
d. Drywell Pressure-High (ECCS Divisions 1 and 2)	1,2,3	4[b]	I	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 psig -48.1
e. Reactor Vessel Water Level-Low Low, Level 2 (HPCS NSPS Div 3 and 4)	1,2,3	4	I	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	≥ -47.7 inches
f. Drywell Pressure-High (HPCS NSPS Div 3 and 4)	1,2,3	4	I	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 psig

(continued)

(b) Also required to initiate the associated drywell isolation function.

(c) During operations with a potential for draining the reactor vessel.

Table 3.3.6.1-1 (page 3 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment and Drywell Isolation (continued)					
g. Containment Building Fuel Transfer Pool Ventilation Plenum Radiation-High	(c),(d)	4	N	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 500 mR/hr
h. Containment Building Exhaust Radiation-High	1,2,3	4(b)	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
	(c),(d)	4	N	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
i. Containment Building Continuous Containment Purge (CCP) Exhaust Radiation-High	1,2,3	4(b)	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
	(c),(d)	4	N	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
j. Reactor Vessel Water Level-Low Low Low, Level 1	1,2,3	4(b)	I	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	≥ -147.7 inches -148.1
	(c)	4	O	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	≥ -147.7 inches
k. Containment Pressure-High	(e)	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 3.0 psid
l. Manual Initiation	1,2,3	2(b)	J	SR 3.3.6.1.6	NA
	(c),(d)	2	N	SR 3.3.6.1.6	NA

(continued)

(b) Also required to initiate the associated drywell isolation function.

(c) During operations with a potential for draining the reactor vessel.

(d) During movement of recently irradiated fuel assemblies in the primary or secondary containment.

(e) MODES 1, 2, and 3 with the associated PCIVs not closed.

Table 3.3.6.1-1 (page 4 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. Auxiliary Building RCIC Steam Line Flow-High	1,2,3	2	I	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	≤ 118.5 inches water
b. RCIC Steam Line Flow-High, Time Delay	1,2,3	2	I	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 13 seconds
c. RCIC Steam Supply Line Pressure-Low	1,2,3	2	I	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	≥ 52 psig
d. RCIC Turbine Exhaust Diaphragm Pressure-High	1,2,3	4	I	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	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 207°F
f. Main Steam Line Tunnel Ambient Temperature-High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 171°F
g. Main Steam Line Tunnel Temperature Timer	1,2,3	2	I	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 28 minutes
h. Reactor Vessel Water Level-Low Low, Level 2	1,2,3	4	I	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	≥ -47.7 inches -48.1
i. Drywell RCIC Steam Line Flow - High	1,2,3	2	I	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	≤ 188 inches water

(continued)

Table 3.3.6.1-1 (page 5 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. RCIC System Isolation (continued)					
j. Drywell Pressure - High	1,2,3	2	I	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 psig
k. Manual Initiation	1,2,3	1	J	SR 3.3.6.1.6	NA
4. Reactor Water Cleanup (RWCU) System Isolation					
a. Differential Flow - High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 66.1 gpm
b. Differential Flow-Timer	1,2,3	2	I	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	2 per room	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 205°F
d. RWCU Pump Rooms Temperature-High	1,2,3	2 per room	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 202°F
e. Main Steam Line Tunnel Ambient Temperature-High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 171°F
f. Reactor Vessel Water Level-Low Low, Level 2	1,2,3	4	I	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	≥ -47.7 inches -48.1
	(c)	4	O	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	≥ -47.7 inches
g. Standby Liquid Control System Initiation	1,2	2	L	SR 3.3.6.1.6	NA
h. Manual Initiation	1,2,3	2	J	SR 3.3.6.1.6	NA
	(c),(d)	2	N	SR 3.3.6.1.6	NA

(continued)

(c) During operations with a potential for draining the reactor vessel.
(d) During movement of recently irradiated fuel assemblies in the primary or secondary containment.

Table 3.3.6.1-1 (page 6 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. RHR System Isolation					
a. RHR Heat Exchanger Ambient Temperature-High	1,2,3	2 per room	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 160°F
b. Reactor Vessel Water Level - Low, Level 3	1,2,3[f]	4	I	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	≥ 8.3 inches
c. Reactor Vessel Water Level - Low, Level 3	3[g],4,5	4[h]	M	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	≥ 8.3 inches
d. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3	4	I	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	-148.1 ≥ -147.7 inches
e. Reactor Vessel Pressure-High	1,2,3	4	I	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	≤ 110 psig
f. Drywell Pressure-High	1,2,3	8	I	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 psig
g. Manual Initiation	1,2,3	2	J	SR 3.3.6.1.6	NA

(f) With reactor steam dome pressure greater than or equal to the RHR cut in permissive pressure.

(g) With reactor steam dome pressure less than the RHR cut in permissive pressure.

(h) Only one trip system required in MODES 4 and 5 with RHR Shutdown Cooling System integrity maintained.

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1.	Reactor Vessel Water Level-Low Low, Level 2	1,2,3,(a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ -47.7 inches -48.1
2.	Drywell Pressure-High	1,2,3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 1.88 psig
3.	Containment Building Fuel Transfer Pool Ventilation Plenum Exhaust Radiation-High	(a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 500 mR/hr
4.	Containment Building Exhaust Radiation-High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 400 mR/hr
5.	Containment Building Continuous Containment Purge (CCP) Exhaust Radiation-High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 400 mR/hr
6.	Fuel Building Exhaust Radiation-High	1,2,3, (c)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 17 mR/hr
7.	Manual Initiation	1,2,3, (a),(b)	1	SR 3.3.6.2.5	NA

(a) During operations with a potential for draining the reactor vessel.

(b) During movement of recently irradiated fuel assemblies in the primary or secondary containment.

(c) During movement of recently irradiated fuel assemblies in the fuel building.

Table 3.3.6.3-1 (page 1 of 1)
RHR Containment Spray System Instrumentation

FUNCTION	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Drywell Pressure-High	2	B	SR 3.3.6.3.1 SR 3.3.6.3.2 SR 3.3.6.3.3 SR 3.3.6.3.4 SR 3.3.6.3.5	≤ 1.88 psig
2. Containment Pressure-High	2	B	SR 3.3.6.3.1 SR 3.3.6.3.2 SR 3.3.6.3.3 SR 3.3.6.3.4 SR 3.3.6.3.5	≤ 22.4 psia
3. Reactor Vessel Water Level-Low Low Low, Level 1	2	B	SR 3.3.6.3.1 SR 3.3.6.3.2 SR 3.3.6.3.3 SR 3.3.6.3.4 SR 3.3.6.3.5	-148.1 ≥ -147.7 inches
4. Timers, System A and System B	1	C	SR 3.3.6.3.2 SR 3.3.6.3.4 SR 3.3.6.3.5	≥ 10.10 minutes and ≤ 10.23 minutes
5. Timer, System B Only	1	C	SR 3.3.6.3.2 SR 3.3.6.3.4 SR 3.3.6.3.5	≤ 90.6 seconds
6. Manual Initiation	1	C	SR 3.3.6.3.5	NA

Table 3.3.6.4-1 (page 1 of 1)
Suppression Pool Makeup System Instrumentation

FUNCTION	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Drywell Pressure-High	2	B	SR 3.3.6.4.1 SR 3.3.6.4.2 SR 3.3.6.4.3 SR 3.3.6.4.6 SR 3.3.6.4.7	≤ 1.88 psig
2. Reactor Vessel Water Level-Low Low Low, Level 1	2	B	SR 3.3.6.4.1 SR 3.3.6.4.2 SR 3.3.6.4.3 SR 3.3.6.4.6 SR 3.3.6.4.7	≥ -147.7 inches -148.1
3. Suppression Pool Water Level-Low Low	2	B	SR 3.3.6.4.1 SR 3.3.6.4.2 SR 3.3.6.4.4 SR 3.3.6.4.6 SR 3.3.6.4.7	≥ 29 inches
4. Timer	1	C	SR 3.3.6.4.2 SR 3.3.6.4.5 SR 3.3.6.4.7	≤ 30 minutes
5. Manual Initiation	2	C	SR 3.3.6.4.7	NA

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains LLS or relief initiation capability, as applicable.

SURVEILLANCE		FREQUENCY
SR 3.3.6.5.1	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.5.2	Calibrate the analog trip module.	92 days
SR 3.3.6.5.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Relief Function Low: 1103 ± 15 psig Medium: 1113 ± 15 psig High: 1123 ± 15 psig b. LLS Function Low open: 1033 ± 15 psig close: 926 ± 15 psig Medium open: 1073 ± 15 psig close: 936 ± 15 psig High open: 1113 ± 15 psig close: 946 ± 15 psig	18 months ≤ 1118 ≤ 1128 ≤ 1138 ≤ 1044 ≤ 937 ≤ 1084 ≤ 947 ≤ 1124 ≤ 957
SR 3.3.6.5.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months