Exelon Generation
4300 Winfield Road
Warrenville, IL 60555

www.exeloncorp.com

Exelon. Nuclear

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U. S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Subject: Follow-up Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation

Reference:

1. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation," dated February 4, 2010
2. Letter from U. S. NRC to Mr. Michael J. Pacilio (Exelon Nuclear), "Dresden Nuclear Power Station, Units 2 and 3 – Request for Additional Information Related to a Modification That Replaces the Temperature-Based Isolation Instrumentation with Reactor Pressure-Based Isolation Instrumentation (TAC Nos. ME3354 and ME3355)," dated September 3, 2010
3. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. NRC, "Additional Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation," dated September 15, 2010

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Renewed Facility Operating License Nos. DPR-19 and DPR-25 for Dresden Nuclear Power Station (DNPS), Units 2 and 3, respectively. Specifically, the proposed amendment revises Technical Specification (TS) 3.3.6.1, "Primary Containment Isolation Instrumentation," Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," Function 6.a, "Shutdown Cooling System Isolation, Recirculation Line Water Temperature - High," to enable implementation of a modification that replaces the temperature-based isolation instrumentation with reactor pressure-based isolation instrumentation. The proposed modification will address instrumentation reliability problems that have led to interruptions of Shutdown Cooling (SDC) System operation. The proposed change to Primary Containment Isolation System (PCIS) instrumentation Function 6.a is needed to ensure reliable heat removal capability, avert plant transients and challenges to equipment, and minimize unnecessary operator actions during plant shutdowns.

In Reference 2, the NRC forwarded requests for additional information (RAIs) concerning the Reference 1 license amendment request. EGC provided the information requested by the NRC in Reference 3.

During a conference call between the NRC and EGC following submittal of the responses to the NRC RAIs, additional follow-up questions were asked by the NRC reviewer to provide clarification of a number of the EGC responses. EGC agreed to provide this follow-up information and the requested information is provided in Attachment 1 to this letter. In addition, the proposed changes to the TS Bases have been revised and are being re-submitted as Attachment 2 for information only. In addition, the retyped Bases pages are provided in Attachment 3.

EGC has reviewed the information supporting a finding of no significant hazards consideration that was provided to the NRC in Reference 1. The additional information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. No new regulatory commitments are established by this submittal.

If you have any questions concerning this letter, please contact Mr. Timothy A. Byam at (630) 657-2804.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 6th day of October 2010.

Respectfully,

Jeffrey L. Hansen

Jeffrey L. Hansen
Manager – Licensing
Exelon Generation Company, LLC

Attachment:
1. Follow-up Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation
2. Revision to mark-up of Proposed Technical Specifications Bases Pages
3. Retyped Proposed Technical Specifications Bases Pages

# ATTACHMENT 1

Follow-up Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation

ATTACHMENT 1
Follow-up Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation

*NRC Clarification 1:*

***Lack of Clarity of what the proposed I&C change, as documented in LAR and RAI responses:***

*The LAR and RAI responses remain inconsistent with respect to the definition of Trip Channels, Trip Strings, and Trip Systems. While some clarification was derived, the remaining inconsistencies prevent one from determining the adequacy of the LCO Operability requirements for the SDC Isolation Trip Channels on Reactor Vessel Pressure-High. Based upon the figure provided by the licensee on page 153 of the RAI response, the proposed Tech Spec B 3.3.6.1-18 statement "Therefore all four channels are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function" appears to be the true licensee intent; however, the LAR proposed Table 3.3.6.1-1 continues to state "REQUIRED CHANNELS PER TRIP SYSTEM" as "2."*

EGC Response 1:

EGC recognizes that a number of inconsistencies were introduced in our previous submittals associated with the proposed change to Technical Specification (TS) Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," Function 6.a. Specifically, the use of the terms Trip Channels, Trip Strings, and Trip Systems were not consistent with the logic descriptions for other functions as described in the TS Bases. There appeared to be some ambiguity associated with the use of the term Trip String. In this application, the use of the term Trip String is synonymous with Trip Channels. Therefore, to be consistent, the following description of the Reactor Vessel Pressure – High function has been revised using more traditional terms (i.e., Trip Channels and Trip Systems).

The Reactor Vessel Pressure – High Function receives input from four reactor pressure channels. Each pressure channel inputs into one of two trip systems. Two pressure channels make up a trip system in a one-out-of-two taken once logic arrangement and both trip systems must trip to cause an isolation of the Shutdown Cooling (SDC) valves. Therefore, the trip systems are arranged in a one-out-of-two taken twice logic configuration. The above referenced figure has been revised to reflect the terminology described here. The revised figure can be found in Enclosure 1 to this Attachment.

Two pressure channels per trip system are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. The Function is only required to be OPERABLE in MODES 1, 2, and 3, since these are the only MODES in which the reactor coolant temperature exceeds the system design temperature and equipment protection is needed. The pressure Allowable Value (AV) was chosen to be low enough to protect the system equipment from exceeding its design temperature. This logic description supersedes the descriptions provided in the original License Amendment Request (Reference 1) and the Request for Additional Information (RAI) response (Reference 2)

Based on the above description of the Reactor Vessel Pressure – High Function logic configuration, the number of required channels per trip system remains as "2." Therefore, there are no required changes to TS Table 3.3.6.1-1, Function 6.a, other than those identified in Reference 1.

_NRC Clarification 2:_

***Adequacy of the LAR Reference NES-EIS-20.04 in determination of OPERABILITY:***

*A precedent SE which the licensee has referenced in its RAI response regarding review of NES-EIS-20.04 only addressed allowable value calculations and occurred prior to RIS 2006-17. Following the licensee's justification that the LAR setpoints were not Safety Limit-related, the RAI responses #4 described Exelon's administrated controls and engineering procedures, but did not submit these procedures themselves on the docket. Nevertheless, portions of the engineering procedure, ER-AA-520, was described in RAI Response #4; however, the description was not consistent with Staff expectations from RIS 2006-17. Specifically:*

- *The licensee states that "If an As-found instrument setpoint exceeds the AV" then the instrument is "potentially inoperable;" however, it is the staff's position that if an As-found instrument setpoint exceeds the AV then the instrument must be declared INOPERABLE.*
- *The licensee does not state that if an As-found instrument setpoint is within the AV, but exceeds the expanded tolerance, that the instrument must be declared to be in a DEGRADED CONDITION; however, it is the staff's position that if an As-found instrument setpoint is within the AV, but exceeds the expanded tolerance, then the instrument must be declared to be in a DEGRADED CONDITION.*
- *The licensee does not state that if an instrument cannot be reset to within the setting tolerance during calibration, that the instrument must be declared INOPERABLE; however, it is the staff's position that if an instrument cannot be reset to within the setting tolerance during calibration, then the instrument must be declared INOPERABLE.*

*Also, the RAI Response #1 indicates that Appendix J, "Guideline for the Analysis and Use of As-Found/As-Left Data," had been modified after the referenced SE was produced. This document was provided in with RAI response, but has not yet been reviewed.*

EGC Response 2:

As stated in the response to RAI 4, EGC Procedure ER-AA-520, "Instrument Performance Trending," establishes the requried actions to be taken when an As-Found instrument setpoint exceeds the AV, as well as when an As-Found setpoint is within the AV, but exceeds the expanded tolerance (ET). A copy of ER-AA-520 is provided as Enclosure 2 to this Attachment.

In accordance with the requirements of the EGC Corrective Action Program (CAP), it is the responsibility of the Shift Manager to determine if a component or system is operable given the condition documented in the CAP. Therefore, ER-AA-520 directs the instrument technician to enter any condition involving a setpoint exceeding the Allowable Value into the CAP by initiating a Condition Report (CR). In accordance with LS-AA-120, "Issue Identification and Screening Process," Section 4.4, the Shift Manager will screen the condition to determine operability of the instrument and associated system.

A copy of LS-AA-120 is provided as Enclosure 3 to this Attachment. In addition, TS Bases Section 3.3.6.1, page B 3.3.6.1-6 states, "A channel is inoperable if its actual trip setpoint is not within its required Allowable Value." Based on the above, it is expected that the Shift Manager will declare an instrument that exceeds its AV or cannot be reset within its Setting Tolerance (ST) to be inoperable. This is consistent with the direction provided in RIS 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."

In the event that an instrument setpoint is found to be outside the ET but still within the AV, the instrument technician will reset the instrument to within the ST and enter the condition into the CAP. This is consistent with the direction provided in RIS 2006-17. In the request for clarification above it is stated that RIS 2006-17 requires the instrument to be declared degraded. EGC has reviewed the text in RIS 2006-17 and there is no requirement in the RIS for an instrument to be declared degraded if it is found within the AV but outside the ET. The RIS specifically states that if the channel trip setpoint (TSP) "exceeds the predefined limits but the measured TSP is conservative with respect to the AV, and the licensee determines during the surveillance that the instrument channel is functioning as expected and can reset the channel to within the setting tolerance of the NSP, then the licensee may restore the channel to service and the condition is entered into the licensee's corrective action program for further evaluation." There is no requirement to consider the instrument degraded.

Based on the above, EGC believes that the current program for evaluating instrument operability is consistent with the guidance provided by the NRC.

*<u>NRC Clarification 3:</u>*

***Common-cause programming error sources from a nonsafety related digital system***

> *With respect to the proposed modification to the SDC Isolation Reactor Vessel Pressure-High Trip function, demonstrate how the plant will continue to meet its design bases for Shutdown Cooling for any postulated failure of the nonsafety digital Bailey Feedwater Control system including but not limited to inadvertent isolation of the Shutdown Cooling system when Shutdown cooling is needed.*
>
> *This issue deals with the potential common-cause programming errors in the digital Bailey Feedwater Controller and in support of the LAR as justified by the need to address "problems that have led to interruptions of Shutdown Cooling (SDC) system operation," and "to ensure reliable heat removal capability, avert plant transients, and challenges to equipment, and minimize unnecessary operator actions during plant shutdowns."*
>
> *The RAI responses state that the SDC Isolation function is not safety related, and non-safety related equipment had always been responsible for the function. Also, RAI response 2. states that the setpoint is not considered to be an LSSS; however, this statement appears contrary regulations. A review of the arrangement of the trip strings confirms that it precludes a single hardware failure within the trip generating*

*equipment from potentially causing an inadvertent isolation; however, this single failure protection does not extend to the power source. Furthermore, because enabling SDC Cooling connection is only a permissive that requires manual action, no common-cause programming error can inadvertently connect the system.*

*Internal hardware failures notwithstanding, the RAI responses describe the use of the digital Bailey Feedwater Controller to generate two of the four SDC Isolation Trip Signals, where each of its Trip Channel signals feeds a one-out-of-two Trip String leg. Therefore, the descriptions indicate that a common-cause programming error could potentially cause an inadvertent isolation, when Shutdown Cooling is required.*

*The responses did not describe any diversity and defense in depth that would address this issue. In contrast, the licensee response does state "A failure of the power source will cause an inadvertent isolation of the SDC system."*

<u>EGC Response 3:</u>
As stated in RAI 7.b response, the Bailey Feedwater (BFW) System processes the analog pressure signal by digitizing it and verifying that it is a good quality signal while concurrently processing the pressure signal through a lead/lag function. This lead/lag function is currently setup as a pass through function where the output equals input (i.e., no lag). After the lead/lag function, the signal goes to a transfer switch and then to the output digital to analog card, if the input signal quality is good. If the input signal quality is bad, the transfer switch forces the output to the digital to analog card to zero.

As stated in RAI 7.f response, there are no common mode software or hardware failures associated with the BFW System that could cause the SDC System to misoperate, therefore, the revised surveillance procedures are adequate to verify proper operation.

The BFW System does not generate the isolation trip signal. It monitors the quality of the pressure signals from the transmitter then outputs two analog output pressure signals that are used by the new pressure trip units to generate the permissive signal.

The BFW System INFI-90 software is classified as Class CC – Business Critical using the EGC Digital Technology Software Quality Assurance (DTSQA) Procedure. Products in this classification support Departmental or Corporate business information and decision-making, where failure to perform could reduce plant availability, impact business productivity, or cause moderate or greater financial impact. This INFI-90 software is not regulatory required for Class BB software or safety related for Class AA software.

The BFW System uses redundant power supplies and CPU cards to provide a highly reliable feedwater system such that no single hardware or software failure will cause a feedwater transient. The BFW System has been in service for over twelve years and therefore, the software has matured and there are no known common mode failures. Due to the criticality of the BFW System, all software changes are tested on the Bailey process simulator to validate the software change and steps required to implement the change. Any software changes to the BFW System require a third party review and verification prior to installation on the system. Installation of software changes are authorized only during plant shutdowns due to the risk of causing a potential Feedwater transient. Installation of software changes are validated and independently verified to ensure that the changes were properly implemented. These actions ensure that no common mode software failures are introduced as part of a software change or upgrade. Therefore, there is high confidence that there are no single software or hardware failures associated with the BFW System that will affect both reactor pressure channels used for the SDC permissive function.

Operating procedure DOA 1000-01, "Residual Heat Removal Alternatives," provides guidance to the operator on how to establish an alternative core cooling method due to a partial or complete SDC System failure. In the event that a failure of the SDC System occurred, alternate methods have been provided to ensure the capability to remove residual heat from the reactor. In Modes 3 and 4, the required cooling capacity of the alternate method should be ensured by verifying, by calculation or demonstration, its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as, or contributing to, the alternate method capability. Alternate methods that can be used include the Condensate/Feed and Main Steam Systems and the Reactor Water Cleanup System by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate/Feed System.

If the above alternatives are not sufficient to control reactor temperature, safety related emergency core cooling systems are used to remove residual heat. These methods include the Isolation Condenser, High Pressure Coolant Injection, and the Main Steam Relief Valves in conjunction with the Suppression Pool Cooling mode of the Low Pressure Coolant Injection System.

*NRC Clarification 4:*
***Surveillance Procedures as may be impacted by inclusion of the digital Bailey Feedwater Controller System***

> *The associated RAI Response 7.f simply states the applicable surveillance procedures are being revised to incorporate the required TS surveillance requirements; however the revised surveillance procedures were neither described nor supplied.*

EGC Response 4:
The Dresden Nuclear Power Station (DNPS) Operating and Instrumentation Surveillance procedures are in various stages of development and revision to support implementation of the design change. Changes to the operating procedures will provide the applicable operator actions to determine if SDC is operable prior to changing modes during startup. These operating procedure changes also provide guidance on establishing alternative core cooling when the SDC function is discovered to be inoperable during the applicable modes of operation. The changes to the maintenance procedures will provide precautions and guidance for entry into the LCO during performance of the surveillances. They will also provide guidance for determining what actions are required when a loop component is determined to be outside its acceptance criterion. As noted above, the revision of these procedures is still in progress and therefore, cannot be provided as a part of this response. The revised procedures will be ready to support

implementation of the proposed modification to the SDC System isolation instrumentation and the proposed license amendment.

<u>NRC Clarification 5:</u>
***Clarification of Proposed Change:***

*Clarify or correct the following to ensure it accurately reflects the change proposed:*

> *There is one SDC Reactor Vessel Pressure-High Isolation Trip System with two halves (A and B), each half is composed of one Trip String, and each Trip String is composed of two Reactor Vessel Pressure-High Trip Channels. One of the two of the Analog Trip System (ATS) Reactor Vessel Pressure-High Trip Channels along with one of the two Bailey Feedwater System (BFW) Reactor Vessel Pressure-Trip Channels is assigned to each Trip String. Specifically ATS-Loop 1B and BFW-Loop 2A are assigned to the Trip String in Trip System Half A, and ATS-Loop 1A and BFW-Loop 2B are assigned to the other Trip String in Trip System Half B.*

*As necessary, provide responses to address inconsistencies between the definitions and sketch provided for RAI Response #5 and the associated Technical Specification use of the terms.*

<u>EGC Response 5:</u>
The above description of the SDC Reactor Vessel Pressure – High isolation trip function is incorrect. Since there appeared to be some ambiguity associated with the use of the term Trip String, EGC has revised the logic description using more traditional terms (i.e., Trip Channels and Trip Systems). In this application, the use of the term Trip String is synonymous with Trip Channels and therefore, to be consistent with the terms used in other TS Bases function descriptions, the following description of the Reactor Vessel Pressure – High function is provided.

The Reactor Vessel Pressure – High Function receives input from four reactor pressure channels. Each pressure channel inputs into one of two trip systems. Two pressure channels make up a trip system in a one-out-of-two taken once logic arrangement and both trip systems must trip to cause an isolation of the SDC isolation valves (i.e., one-out-of-two taken twice arrangement). Each trip system will consist of one Reactor Vessel Pressure channel from the Analog Trip System (ATS) and the BFW System. Specifically, ATS-Loop 1B and BFW-Loop 2A are assigned to the Trip System A and ATS-Loop 1A and BFW-Loop 2B are assigned to Trip System B. This configuration ensures that there are no hardware or common mode software failures within either the ATS or the BFW System that will prevent the SDC System from isolating when required. It is important to note that, as described in the response to RAI question 7.b, the BFW System does not provide the trip function. The BFW System passes the reactor pressure signals to new trip channels that feed into the SDC System Reactor Pressure – High isolation function.

The sketch provided previously in RAI response 5 and the associated mark-ups of the TS Bases sections provided in Attachment 2 to Reference 2 have been revised to reflect this description of the logic. The revised sketch is provided as Enclosure 1 to this Attachment and the revised Bases mark-ups are provided in Attachment 2 to this letter.

In the event of a partial or complete failure of the SDC System, alternate methods of providing core cooling have been identified. The Residual Heat Removal alternatives procedure provides guidance to the operator on how to establish alternative core cooling due to a partial or complete SDC System failure. The alternate methods include the Condensate/Feed and Main Steam Systems and the Reactor Water Cleanup System by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate/Feed System.

*NRC Clarification 6:*
***Regarding General Operability:***

> *Clarify the conditions, if any, that a single trip channel becoming declared INOPERABLE, will direct manual/forced SDC isolation of both SDC loops.*

EGC Response 6:
There are no known conditions that would require declaring the other trip channels inoperable when a single trip channel is declared inoperable. Consistent with the TS required action associated with one or more required channels being inoperable, DNPS will place the inoperable channel in trip within 24 hours.

As described above, in the event that a failure of the SDC System occurred, alternate methods have been provided to ensure the capability to remove residual heat from the reactor.

*NRC Clarification 7:*
***Regarding General Operability:***

> *Clarify if when any Trip Channel is undergoing surveillance or calibration (as applicable to Modes 1, 2 or 3) whether its associated Trip String will be forced to generate a ½ trip, and, if so, confirm that placing the Trip System in this state will not adversely affect the ability to perform the surveillance.*

EGC Response 7:
In accordance with Note 2 prior to the TS 3.3.6.1 Surveillance Requirements, 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 isolation capability. If during performance of the surveillance the channel is determined to be inoperable, then TS Condition A is entered and the station complies with the associated Required Action. Performance of the surveillance will typically occur while the plant is at normal operating conditions where the four-trip units contacts are normally open and SDC System is isolated. During performance of the surveillances, only one trip channel is tested at a time and therefore, only one trip system is affected during performance of the surveillance. The trip unit contact for the applicable trip channel under test is verified to

change state as the channel test signals are varied. Performance of the surveillance will not adversely affect the SDC isolation function during normal plant operations.

The proposed circuit configuration allows performance of the surveillance of the SDC System while the system is in service. Performance of the surveillance while the SDC System is in service, only affects one trip system and will place it in a half trip condition. This does not adversely affect the SDC System since it is still able to perform its design function while performing the surveillance.

*NRC Clarification 8:*
***Regarding digital Bailey Feedwater expected operations:***

- *Clarify whether a half trip results (or is forced) when the Bailey Feedwater System sets the A/D output to zero in response to a "input signal of bad quality" or other self-check failures are detected, which is seen as consistent with the RAI response 7.e. statement of "the failed channel will be required to be placed in a half trip condition."*

EGC Response 8:
The BFW System output will only be set to zero if a bad quality input signal is sensed. Other self-check failures do not set the pressure signal output to zero. When the BFW System senses a bad quality input signal it is an indication of either a transmitter, power supply, and/or input card failure. The BFW System reactor pressure signal uses two different input cards to prevent a signal card failure from affecting both pressure channels. The BFW System uses redundant power supplies and CPU cards to provide a highly reliable feedwater system such that no single hardware or software failure will cause a loss of the feedwater system. This ensures that no single failure within the BFW System will affect both reactor pressure channels being used for the SDC permissive function.

The BFW System logic monitors the reactor pressure signal and determines the signal quality. If the pressure signal exceeds its upper or lower range limit it will set a bad quality flag and the transfer switch will set the digital to analog (D/A) output card output to zero for the applicable pressure channel. This will cause the pressure trip unit to change state if the reactor pressure was above the nominal pressure setpoint. If the reactor pressure were below the nominal pressure setpoint, the trip unit contact would remain closed.

If this failure occurs while above the nominal setpoint, only one contact within the trip system will be closed. The permissive logic would not be satisfied since the second trip unit contact would remain open and manual operator action is required to un-isolate the SDC System. Therefore, SDC System cannot be inadvertently un-isolated by this single failure.

If this failure occurs while the SDC System is in operation, the SDC System would not inadvertently isolate because the applicable pressure trip unit contact will remain in a closed state since the pressure signal is below the nominal pressure setpoint. In addition, this failure does not affect the second trip system. Therefore, this single failure cannot inadvertently isolate the SDC System.

In either case, the Dresden annunciator alarm procedure guidance being developed will require the operator to declare the applicable pressure channel inoperable and pull the associated pressure trip unit fuse, placing the trip system in a half trip condition until the failure condition can be resolved.

*NRC Clarification 9:*
***Regarding digital Bailey Feedwater expected operations:***

> *Clarify what actions, if any, are initiated for SDC isolation when the Bailey Feedwater System sets the A/D output to zero in response to a "input signal of bad quality."*

EGC Response 9:
The BFW System output will only be set to zero if a bad quality input signal is sensed. Other self-check failures do not set the pressure signal output to zero. When the BFW System senses a bad quality input signal it is an indication of either a transmitter, power supply, and/or input card failure. The BFW System reactor pressure signal uses two different input cards to prevent a single card failure from affecting both pressure channels. The BFW System uses redundant power supplies and CPU cards to provide a highly reliable feedwater system such that no single hardware or software failure will cause a loss of the feedwater system. This ensures that no single failure within the BFW System will affect both reactor pressure channels being used for the SDC System isolation permissive function.

The BFW System logic monitors the reactor pressure signal and determines the signal quality. If the pressure signal exceeds its upper or lower range limit it will set a bad quality flag and the transfer switch will set the digital to analog (D/A) output card output to zero for the applicable pressure channel. This will cause the pressure trip unit to change state if the reactor pressure was above the nominal pressure setpoint. If the reactor pressure were below the nominal pressure setpoint, the trip unit contact would remain closed.

If this failure occurs while above the nominal setpoint, only one contact within a trip system will be closed. The permissive logic would not be satisfied since the second pressure channel trip unit contact would remain open and manual operator action is required to un-isolate the SDC System. Therefore, the SDC System cannot be inadvertently un-isolated by this single failure.

If this failure occurs while the SDC System is in operation, the SDC System would not inadvertently isolate because the applicable pressure trip unit contact will remain in a closed state since the pressure signal is below the nominal pressure setpoint. In addition, this failure does not affect the second trip system. Therefore, this single failure cannot inadvertently isolate the SDC System.

In either case, the trip unit will alarm an annunciator and the DNPS annunciator alarm procedure guidance being developed will require the operator to declare the applicable pressure channel inoperable and pull the associated pressure trip unit fuse which places the pressure channel in a half trip condition until the failure condition can be resolved.

References:

1. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation," dated February 4, 2010

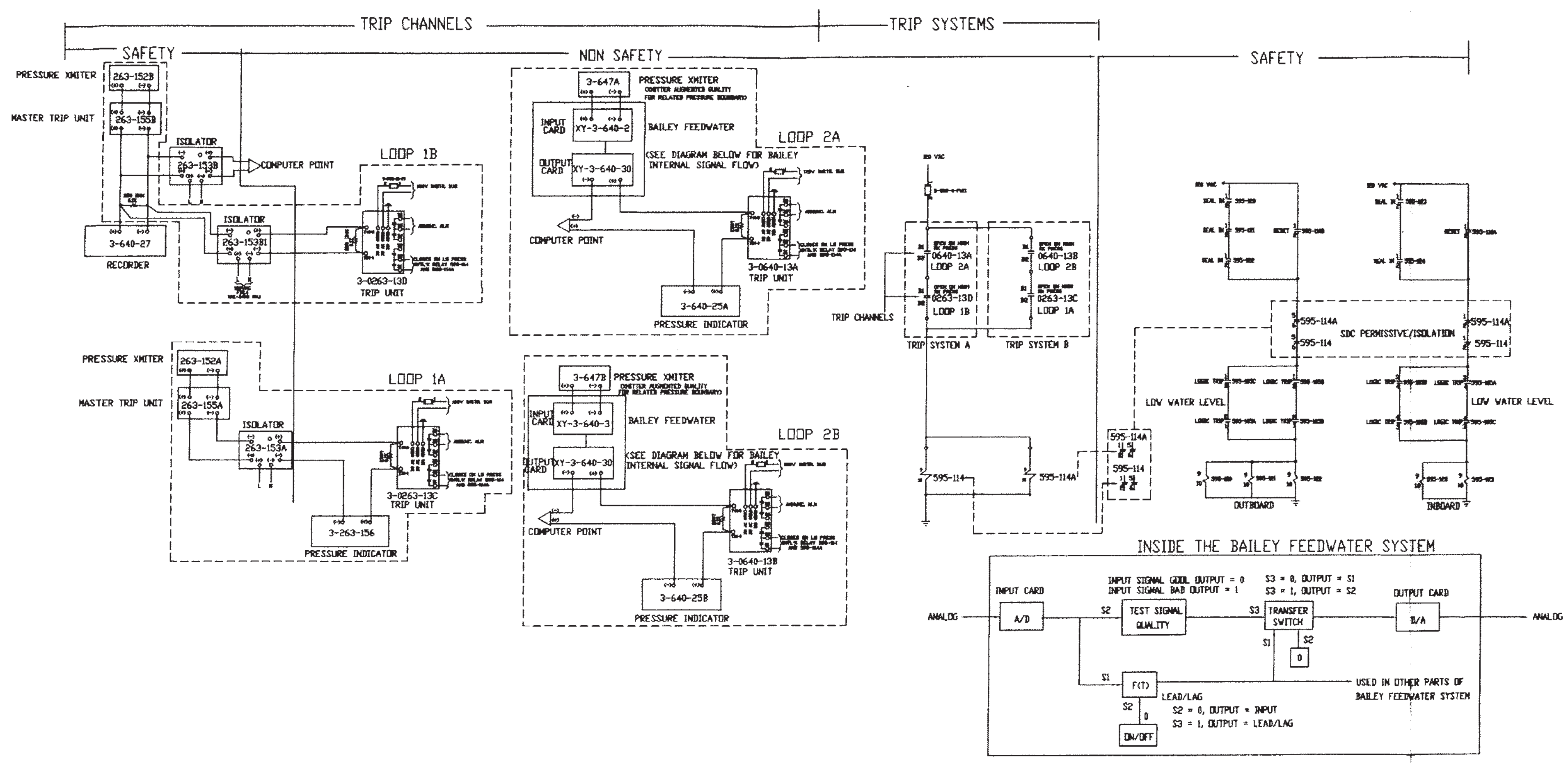
2. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. NRC, "Additional Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation," dated September 15, 2010

ATTACHMENT 1
Follow-up Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation

Enclosure 1

Sketch of the Proposed Circuit Layout for Shutdown Cooling System Instrumentation Logic

# U3(2) CONFIGURATION AFTER LAR (ONLY UNIT 3 COMPONENTS SHOWN)



ATTACHMENT 1
Follow-up Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation

Enclosure 2

ER-AA-520
Instrument Performance Trending
Revision 3

# INSTRUMENT PERFORMANCE TRENDING

## 1. PURPOSE

1.1. This procedure provides the administrative process to implement an instrument trending program. An instrument trending program is a good engineering practice to monitor the behavior of instrumentation to provide early warning of failure.

1.2. This program monitors the results of calibrations of applicable instrumentation in the plant and generates periodic reviews of the data collected during these calibrations to determine what instruments are not performing to expectations.

1.3. This procedure identifies poor performance, which can occur in three basic ways:

1.3.1. An individual instrument could begin to show signs of failure by not meeting Setting Tolerance or exceeding the Leave Alone Zone (LAZ) for repeated calibrations. This is indicative of potential failure of the instrument at some future time.

1.3.2. Most or all of the instruments monitoring a specific plant parameter could begin to show signs of failure by not meeting Setting Tolerance or LAZ for repeated calibrations. This is indicative of the instrument being assigned a Setting Tolerance that is too constrictive for the make/model used. If the Setting Tolerance can not be expanded to prevent repetitive failures, then the instrument may not be the correct one for the parameter of concern.

1.3.3. Most or all of the instruments of a given make/model could begin to show signs of failure by not meeting Setting Tolerance or LAZ for repeated calibrations. This is indicative of the instrument being assigned a Setting Tolerance that is too constrictive for the make/model used. If the Setting Tolerance can not be expanded to prevent repetitive failures, then the instrument may not be the correct one for use. If this occurs after calibrations were successful, then the potential for a common mode failure exists.

1.4. This procedure provides control of the As-Found/As-Left data analysis. This program maintains the analysis conducted as part of the 24-month cycle extension project as required by Generic Letter 91-04 for applicable stations.

1.5. This procedure is applicable to all Exelon Operating Nuclear Stations. All instruments within the stations calibration program that are safety related, tech spec related, Reg. Guide 1.97, or Maintenance Rule instruments shall be included in this program. Those instruments that do not fall into one of the categories listed in the applicability statement are not required to be entered into the out of tolerance and trending program. The stations may choose to include other items in the trending program as they feel appropriate.

1.6. This procedure allows the sites to choose from several methods of trend data recording. Trending may be accomplished by the use of any of the methods outlined within this procedure. This includes the coded CR method, or by use of the as found condition codes within Passport, or the PIMS as found condition codes, or a suitable instrument "As-Found / As-left" data trending tool.

## 2. TERMS AND DEFINITIONS

2.1. **Allowable Value (AV)**: The limiting value that the trip setpoint may have when tested periodically, beyond which appropriate action shall be taken. The allowable value provides operability criteria for those setpoints or channels that have a limiting operating condition. This limiting condition is typically imposed by the Technical Specifications, but may also result from regulatory requirements, vendor requirements, design basis criteria or other operational limits.

2.2. **Leave Alone Zone (LAZ) -Applicable to MAROG**: The LAZ is a range of acceptable values around a nominal value established by adding or subtracting the required accuracy from the nominal value. When an instrument reading (cardinal point of calibration or trip setpoint) is found within this band during Surveillance Testing or calibration check, no calibration adjustment is required. In special cases, the LAZ can be established as a non-uniform band around a nominal value.

2.3. **Reference Accuracy (RA)**: A number or quantity that defines a limit that errors will not exceed, when a device is used under specified operating conditions. Includes the combined effects of linearity, hysteresis, deadband, and repeatability.

2.4. **Setting Tolerance (ST)/As Left Tolerance (ALT)**: Inaccuracy or offset introduced into the calibration process due to procedural allowances given to technicians performing the calibration. Proper selection of ST/ALT should take into account the effects of reading error and ease of instrument adjustment. The limits allowed for the "As-left" value of a setpoint or cardinal point during calibration (see Attachment 1).

2.5. **Expanded Tolerance (ET)/As Found Tolerance (AFT) -Applicable to MWROG**: The tolerance established for trending instruments that are found beyond the ST/ALT. This is a generic term that encompasses other terms presently used for an "As-Found" acceptance criteria including Administrative Limit, Reportable Limit, Performance Limit etc. It is the value established by applying the process described in Attachment 1. Trends will be evaluated against this value rather than the Setting Tolerance.

2.6. **Out of Tolerance (OOT)**: The condition that exists when the As Found values for an instrument calibration exceed some pre-established limit or tolerance value (ET or LAZ).

3. **RESPONSIBILITIES**

3.1. The **Site Engineering Director** is responsible for:

3.1.1. Implementing the site Instrument Trending Program.

3.1.2. Developing calculations of tolerances pertaining to instruments covered in this procedure with the exception of the "Quick Expanded Tolerance (ET)/As Found Tolerance (AFT)", which is provided for the maintenance supervisors determination and use.

3.2. The **Site Maintenance Director** is responsible for:

3.2.1. Implementing the site Instrument Calibration Program.

3.2.2. Coding of calibration work order activity cause and repair codes.

3.3. The **Senior Manager, Design Engineering** is responsible for:

3.3.1. Updating the drift analysis for the instrumentation at those sites committed to a Drift Monitoring program using the supplied data once each operating cycle.

3.3.2. Evaluating the Trend Report for indication of common mode failures once per operating cycle.

3.4. The **Senior Manager, Plant Engineering** is responsible for:

3.4.1. Reviewing the trend report and evaluating instruments associated with a system within a month of receipt of the report.

3.5. The **Surveillance Test Coordinator** (MAROG only) is responsible for:

3.5.1. Coding of Surveillance Testing work order activity ST Grade codes.

## 4. MAIN BODY

4.1. **Exceptions** – The CR trend codes used in this procedure are a station option. The CR trend codes simply provide the engineer with a way to "bin" all instrument out of tolerances. Therefore, the stations may use the trend codes at their own discretion. Trend Codes are defined in section 4.2.2.1.D

### 4.2. Requirements

NOTE A: The calibration program is defined within station specific procedures and shall be incorporated into the station work control process to ensure compliance with technical specifications and station commitments.

NOTE B: This procedure requires that any instrument covered under the applicability stated in section 1.5 of this procedure, that is out of tolerance, is entered into an appropriate trending process and the trends evaluated as described in this procedure

4.2.1. Reporting Out of Tolerances of instruments or control devices covered by the station Calibration Program for Stations using Passport and CR Trending method.

1. If an instrument can not be reset to within Setting Tolerance/As Left Tolerance during calibration, then **INITIATE** a CR to document the information and the instrument will be repaired/replaced. For plants required to collect as-found information, **RECORD** the information unless the instrument has failed.

    A. If any as-found data is greater than the AV, **WRITE** a CR (Subject line to read: Inst. OOT, *Equipment ID*, and "Trend Code B1").

    B. If all as-found data is less than the AV, then **WRITE** a CR (Subject line to read: Inst. OOT, *Equipment ID*, and "Trend Code B3").

2. If the calibration of the instrument/loop had at least one calibration point found outside the Setting Tolerance/As Left Tolerance (i.e. requiring adjustment of the instrument/loop) but the loop is left within the Setting Tolerance/As Left Tolerance, then the following actions are required by maintenance during review of the calibration procedure data:

    A. If an ET/AFT exists (in the controlled Plant Equipment Database or in the calibration procedure data), **COMPARE** the as-found data to that data and:

1. If all as-found data is within the ET/AFT, then document this evaluation on the procedure and close the WR without additional required action.

2. If any as-found data is outside the ET/AFT, then proceed to 4.2.1.2.D.

B. If an ET/AFT does not exist (in the controlled Plant Equipment Database or in calibration procedure data), then:

1. If the instrument loop provides a Technical Specification automatic initiation function, initiate an ER or AR/AR Eval. to obtain a calculation, ET/AFT, and Allowable Value as required.

2. If the instrument loop does not provide a Technical Specification automatic initiation function, then proceed to Section 4.2.1.2.C.

C. **DETERMINE** a Quick ET/AFT using Attachment 1 and **EVALUATE** the data against the Quick ET/AFT using the following criteria:

1. If all of the calibration data are within the Quick ET/AFT, close the WR without further action. If it is desired to incorporate the ET into the controlled Plant Equipment Database or the calibration procedure, INITIATE an ER or AR / AR Eval. to do so.

2. If any calibration data is outside the ET/AFT, then proceed to 4.2.1.2.D.

D. If any data point exceeds the ET/AFT, **DETERMINE** if any data point exceeds the Allowable Value (AV) for the instrument/ loop using the following process:

1. If an AV exists for the instrument/loop and a data point exceeds the AV, then WRITE a CR (Subject line to read: Inst. OOT, Equipment ID, and "Trend Code B2", if desired) and notify the shift manager that that instrument loop is potentially inoperable. If an appropriate instrument data trending tool is used, enter the as found and as left information in addition to writing the CR.

2. If no AV exists for the instrument loop, then write a CR (Subject line to read: Inst. OOT, Equipment ID, and "Trend Code B4", if desired) indicating that the instrument/loop was outside its ET/AFT OR record the as found and as left data in an appropriate instrument data trending tool.

3. If data point exceeds the ET/AFT but is inside AV, then write a CR (Subject line to read: Inst. OOT, Equipment ID, and "Trend

Code B4", if desired) **OR** record the as found and as left data in an appropriate instrument data trending tool.

E. The threshold for generating an OOT CR for relays, time delay relays, and level switches calibrated by the Electrical Maintenance Department, Operational Analysis Department, or other department performing maintenance and calibration on these devices should continue to be based on the "TOLERANCE" currently stated in calibration procedures. **WRITE** a CR (Subject line to read: Inst. OOT, *Equipment ID*, and "Trend Code B4") if the stated tolerance is exceeded.

4.2.2. Trend Reporting Using The CR

Note: The following section, 4.2.2.1 does not apply to stations using PIMS for action tracking and an appropriate instrument data trending tool.

1. To provide for a simple trending process, the CR will be used as the documentation process. CR's written to solely document the trend code of an instrument's calibration should be able to be "closed to trending" (or an equivalent of this) since ER-AA-520 requires periodic reporting. CR's that document inoperability or exceeding tech spec's shall not be closed to trending only. To ensure that the CR will document the necessary information, the following are the minimum requirements that must be included in addition to that normally put in the CR:

   A. As a minimum, **ENSURE** that the subject field includes: "Instrument Out of Tolerance (OOT)"

   B. On the Originator Screen, **ENSURE** that the Equipment ID is included and that the Equipment ID represents the loop or instrument that is out of tolerance. Also include reference to applicable procedure, WR or surveillance number.

   C. On the Originator Screen, in the Action Request Description section, **ENTER** one of the following that most correctly states the degree of Out of Tolerance:

      1. "Calibration Data exceeded the ET/AFT (Quick ET/AFT) but did not exceed the AV. Instrument/loop recalibrated to within ST/ALT." If no ET/AFT previously established and a computed ET/AFT is used, document here.

      2. "Calibration Data exceeded an AV. Instrument/loop recalibrated to within ST/ALT."

3. "Instrument has failed or can not be recalibrated to within ST/ALT."

   In addition, **PROVIDE** the following information:

   – The magnitude and direction of the as-found value and the ET/AFT/ST/AFT; that is, whatever tolerances were exceeded.
   – The Trend Code, as applicable, in both Action Request Description and Subject sections.

D. On the Originator Screen in the Subject section, one of the following statements should be made (trend codes may be omitted at site discretion):

   1. "Trend Code B1" – At least one as-found data point exceeded the AV for the instrument or loop and the instrument can not be reset to within ST. Notify shift manager that the instrument loop is potentially inoperable. Repair or replace as appropriate.

   2. "Trend Code B2" – At least one as-found data point exceeded the AV for the instrument or loop and the instrument can be reset to within ST. Notify shift manager that the instrument loop was potentially inoperable. Recalibrate, repair or replace as appropriate.

   3. "Trend Code B3" – No as-found data point exceeded the AV for the instrument or loop and the instrument can not be reset to within ST. Repair or replace as appropriate.

   4. "Trend Code B4" – No as-found data exceeded the AV but at least one data point exceeded the ET for the instrument or loop and the instrument can be reset to within the ST. Close CR to trend data point.

4.2.3. Reporting Out of Tolerances of instruments or control devices covered by the station Calibration Program for Stations using PIMS for trending.

**IF** an instrument is Out of Tolerance (beyond LAZ), then:

1. **TAKE** the appropriate corrective actions in accordance with the applicable site procedures.

2. For Surveillance Testing, **DOCUMENT** the Test Grade in the PIMS Work Order per the site governing procedure (Ref.6.7)

3. For PIMS corrective / preventive Work Order activities, **DOCUMENT** the cause and repair codes per the site governing procedure.

A. **DOCUMENT** the As-Found and As-Left conditions in the work order Completion Remarks in accordance with MA-MA-716-010-1008, Section 8.5.

B. **DOCUMENT** the appropriate Cause and Repair codes on all Work Order activities in accordance with MA-MA-716-010-1008, Exhibits 8.4.1and 8.5.1.

4. If already existing, **RECORD** the As-Found and As-Left data in the instrument calibration program record.

4.2.4. Reporting Out of Tolerances of instruments or control devices covered by the station Calibration Program for Stations using PASSPORT for trending.

If an instrument is Out of Tolerance ( beyond its setting tolerance ), then:

1. In Passport document the as found condition in the work order by selecting the appropriate As-Found condition code (summary listing of available codes are in MA-AA-716-011 attachment 2.).

2. Take appropriate actions per site procedures in generating a CR and notifying station management of potential inoperability.

## 4.3. As-Found/As-Left Program

4.3.1. An As-Found / As-Left Program is required only if the plant has committed to it as part of extending it operating cycle to 24 months. It may be implemented for other instrumentation at the discretion of the specific plant. The purpose of this program is to maintain a continuing evaluation of instrument drift based on calibration data and to incorporate any increase in observed drift into the appropriate calculations.

1. Instruments that are required to be trended, will be designated in the appropriate section of the controlled Plant Equipment Database .

2. The Site Design Engineering group will **UPDATE** the drift analysis for the instrumentation in the Drift Monitoring program using the supplied data once each operating cycle.

## 4.4. Trending Program

4.4.1. The trending program will provide the plant with the analysis of the data provided by the above Sections.

4.4.2. For Plants using the CR trending approach: Approximately once per operating cycle, Engineering will **RUN** a Trend Report on the CR database. The trend report should be created by searching on the Subject field for "instrument out of tolerance", "OOT", "Tol" or something similar that will encapsulate all the out of tolerance CR's that were generated during the applicable period of time. This report can be sorted by System, Equipment ID, and, as applicable, trend code.

4.4.3. For Plants using the PIMS trending approach: Once per Operating Cycle, Engineering will **RUN** a Trend Report on the PIMS Work Order database. Engineering will **REVIEW**, at a minimum, Surveillance Test Work Orders with grades of "R", "A", and "U", and Work Orders with Cause Codes equal to "C4" and Repair Codes equal to "AA", "AG", "AH", and "AK".

4.4.4. For Plants using the Passport trending approach: Once per Operating Cycle, Engineering will RUN a Trend Report on the Passport Work Order database As Found condition codes.

4.4.5. Cognizant System Managers shall **REVIEW** the report and **EVALUATE** instruments associated with their systems. If a potential problem with the instrumentation on a system is determined, the System Manager should **INITIATE** a Trending CR to document the specific adverse trend and to evaluate the instrumentation of concern for appropriate corrective action. Instruments to be considered for evaluation are defined as 2 or more CR's over the last 5 calibration periods for a given instrument, <u>OR</u> 2 or more adverse trend codes (Passport or PIMS conditions reports) over the last 5 calibration periods for a given instrument.

4.4.6. Site Design Engineering will **EVALUATE** the Trend Report for indication of common mode failures once per operating cycle. If an adverse trend is identified, Design Engineering will **INITIATE** a Trending CR to evaluate the instrumentation of concern.

1. Adverse Trend CR's should contain the following:

    A. A description of "instrument out of tolerance trending report".

    B. A listing of what system / instruments were reviewed.

    C. A brief description of the resolution. Possible resolutions include:

        1. Revise calibration acceptance criteria (i.e. ST, AV, ET, LAZ)

        2. Increase surveillance / calibration frequency

        3. Replace the instrument

        4. Evaluation of correct instrument application

2. At least once per operating cycle, Site Design Engineering will **PERFORM** the following for drift analysis as required per site commitment:

    A. For those instruments in the As-Found/As-Left program, Site Design Engineering will **UPDATE** the drift analysis for the make/model groups.

B. For any updated drift value that either is a larger magnitude or changes from time independent to time dependent, a CR will be **WRITTEN** to require all associated setpoint calculations to be updated.

C. The required ER's or AR / AR eval. will be **WRITTEN** for any changes in setpoints or tolerances in accordance with CC-AA-103.

5. **DOCUMENTATION**

5.1. Trend reports per Section 4.4

6. **REFERENCES**

6.1. Nuclear Engineering Standard NES –EIC-20.04 (includes Industry Standards)

6.2. Exelon Procedure CC-AA-103, Configuration Change Control

6.3. Nuclear Design Informational Transmittal, DIT-BRW-2000-004, PIF Threshold for "Out-Of-Tolerance" Reporting for instruments or Channels Which Have Only an Instrument Calibration Setting Tolerance, 1-18-2000.

6.4. Exelon Procedure LS-AA-105, Operability Determinations

6.5. Exelon Procedure LS-AA-125, Corrective Action Program

6.6. Exelon Procedure MA-MA-716-010-1008, Work Order (W/O) Work Performance

6.7. Site Specific Procedure for Surveillance Testing

6.8. Exelon Procedure MA-AA-716-011, Work Execution and Closeout.

6.9. ComEd Licensing submittal to NRC dated March 3, 2000 for technical specification changes for Dresden, Quad Cities, and LaSalle Stations to convert to Improved Standard Technical Specifications.

7. **ATTACHMENTS**

7.1. Attachment 1: Establishing Setting and Expanded Tolerances/As Found Tolerances (Applicable to MWROG Only)

## ATTACHMENT 1
## Establishing Setting (As Left) and Expanded (As Found) Tolerances
## (Applicable to MWROG only)


### SETTING (As Left) TOLERANCE:

The setting tolerance is selected to allow the technician a band in which an instrument can be left after calibration. This will minimize the amount of adjustment that the technician performs in attempting to set the instrument. This setting tolerance should be included in the evaluation of the uncertainty of the instrument/loop to indicate the monitored process parameter. Allowing too large of a ST can allow too much uncertainty in the loop calibration and/or not allow for detection of potential instrument failure.

**Establishing a New Setting Tolerance:**

In some cases new instruments are added to the plant's equipment, or old instruments have not had a setting tolerance established. The following guidance will be used to select the initial setting tolerance of the instrument:

1. If the instrument has a Reference Accuracy defined, then that value should be selected as the Setting Tolerance. Some adjustment to this value can be accommodated to provide the technician with easy to read values. This value can be adjusted based on system operability requirements.

2. If the ability of the Measurement & Test Equipment (M&TE) to meet the above ST is not possible then select the ST at the value of the M&TE accuracy. As before, some adjustment to this value can be accommodated to provide the technician with easy to read values. This value can be adjusted based on system operability requirements.

3. To determine STs for loops or partial loops the Square Root Sum of the Squares (SRSS) of the individual instrument STs can be taken. As before, some adjustment to this value can be accommodated to provide the technician with easy to read values.

**ATTACHMENT 1**
**Establishing Setting (As Left) and Expanded (As Found) Tolerances**
**Page 2 of 3**

**EXPANDED (As Found) TOLERANCE: (MWROG only)**

Note: For some stations, the ET is similar to the LAZ and need not be calculated as directed in this attachment.

The expanded tolerance is a value that incorporates some of the additional uncertainty that can occur between calibrations. This expanded tolerance is very close to an Allowable Value as defined and explained in ISA S67.04 - 1994 Part I and II. The principle involved is that the instrument will show some drift from calibration to calibration and there are intrinsic uncertainties in calibration itself. If the instrument is in an As-Found state that is within this amount of uncertainty then the instrument is performing as expected in the loop uncertainty calculation. To select an ET perform the following:

**CALCULATED ET (BY ENGINEERING):**

1. If there is a formal loop uncertainty calculation that has an Allowable Value calculated for the loop and/or any individual instruments, the ET should be the AV or some percentage of the AV.

2. If the calculation does not compute an AV, then the assumed STs for each instrument can be combined with the Drift and Reference Accuracy of the instrument in a SRSS to determine the ET. The ET for the loop will then be the individual ETs in the loop combined in the same manner as the channel uncertainty was determined.

3. If there is no formal loop uncertainty calculation, then the ET can be computed by conducting a SRSS of the ST, RA and drift of the instrument of concern. If drift is not known then the value of RA or the specified values in NES-EIC-20.04 can be used. The ET for the loop will then be the SRSS of the individual ETs in the loop.

4. Other processes have been used in ComEd to compute ETs. These values are still valid and the process, if documented in site procedures, can still be used.

**QUICK ETs (AFTs):**

The Maintenance Supervisor may need a value to determine the failure code during calibrations when no THE CONTROLLED PLANT EQUIPMENT DATABASE value exists. The following is the acceptable way to compute a quick ET to use for close out of the work package:

1. If the ST is set at the RA of the instrument, multiply the ST by 1.5 and use this as the ET.

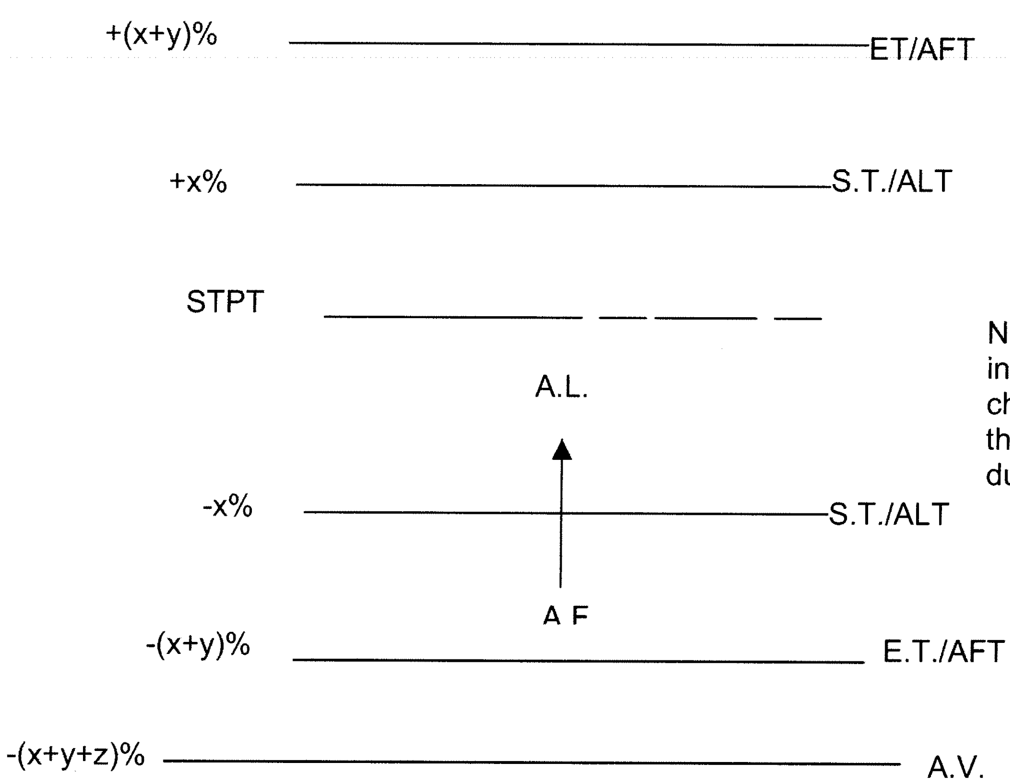
2. If the ST is larger than the RA, then use the ST as the ET.

3. If no RA is available, then multiply the ST by 1.5 and use this product as the ET.

**ATTACHMENT 1**
**Establishing Setting (As Left) and Expanded (As Found) Tolerances**
**Page 3 of 3**

**Typical Instrument With Setpoint**

**(Example of Decreasing Trip With Allowable Value)**




NOTE: The arrow indicates the change made by the technician during calibration.

WHERE:

| | | |
|---|---|---|
| STPT | = | Station Setpoints Value |
| A.L. | = | "As Left" Value. |
| S.T. | = | Setting Tolerance Value. |
| A.F. | = | "As Found" Value. |
| E.T. | = | Expanded Tolerance Value. |
| A.V. | = | Allowable Value. |
| x, y, z | = | Tolerances and Uncertainty Values (Illustration Purposes Only) |

ATTACHMENT 1
Follow-up Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation

Enclosure 3

LS-AA-120
Issue Identification and Screening Process
Revision 12

# ISSUE IDENTIFICATION AND SCREENING PROCESS

## 1. PURPOSE

1.1. Defines a common process through which personnel at Exelon Nuclear facilities can identify and gain assignment for resolution of identified issues.

1.2. Establishes the roles, responsibilities, and requirements for the identification, screening and classification of identified issues.

## 2. TERMS AND DEFINITIONS

2.1. **Assignment**: A task required to be implemented to resolve a Condition Report (CR). For all stations, assignments are captured as Assignments within a PassPort Action Request.

2.2. **Computer Program**: Applicable Computer program used to capture an Issue. For all stations the applicable computer program is PassPort Action Tracking.

2.3. **Condition Adverse to Quality (CAQ)**: An all-inclusive term used in reference to any of the following: failures, malfunctions, deficiencies, defective items, and non-conformances. Attachment 2 provides a listing of CAQs that Exelon Nuclear has determined require specific management screening to ensure the issues are addressed.

2.4. **Condition Report (CR)**: A document in the Computer Program used to record and address Corrective Action Program items.

2.5. **Equipment Failure**: Damage to or degradation of a System, Structure, or Component (SSC) that may cause or contribute to an event.

2.6. **Extent of Condition**: The extent to which the actual condition exists with other plant processes, equipment, or human performance. While Issues, Apparent Cause Evaluation (ACEs), and Root Cause Report (RCRs) all include a discussion of extent of condition, it is expected that the level of effort in determining and documenting the extent of condition is commensurate with the level of investigation and significance of the event.

2.7. **Immediate Action**: Action taken immediately upon discovery to mitigate or terminate the consequences of a condition.

2.8. **Interim Corrective Action**: Action(s) taken to temporarily prevent the effects of a condition or make an event less likely to recur during the period when the condition is being evaluated and until final corrective actions or Corrective Actions to Prevent Recurrence (CAPRs) are completed.

2.9. **Issue**: Includes any equipment deficiencies, equipment or document non-conformances, programmatic deficiencies, human performance errors, enhancements (improvements), and commendable behaviors.

2.10. **Significant Condition Adverse to Quality (SCAQ)**: A condition, which if left uncorrected, could have a serious effect on safety or operability. Severe operating abnormalities or large deviations from expected plant performance of safety related structures, systems, or components; "events" such as described in the plant Technical Specifications; pervasive breakdowns in the quality assurance program; recurring deficiencies or errors that **cannot** be dispositioned or brought into conformance by established corrective action systems; or violations of the ASME Code that **cannot** be readily brought into compliance.

## 3. RESPONSIBILITIES

3.1. Plant Manager, or designee

- Responsible for overall execution of the screening process at the site.
- Designate Site Ownership Committee Chair Person and members.

3.2. Vice President, Operations Support, or designee

- Responsible for overall execution of the screening process at the corporate facilities.
- Designate Corporate Ownership Committee Chair Person and members.

3.3. Ownership Committee Chairman, or designee

- Responsible for the proper conduct of the Ownership Committee Meetings for the Facility.
- Provides integrated plant knowledge to lead Ownership Committee Meetings.
- Responsible for understanding organizational resource impact when assigning work based on risk and significance of issue.

3.4. Regulatory Assurance Manager/Licensing Director

- Provide oversight of the Screening Process and maintain overall responsibility for effective implementation of the process for the facility, as appropriate.

3.5. Site/Corporate Corrective Action Program Coordinator (CAPCO)

- Serve as the facilities point of contact on issue tracking matters.
- Provide guidance on Screening Process expectations.
- Perform oversight of the effective implementation of the Screening Process

3.6. Ownership Committee Members

- Prior to the Ownership Committee Meeting, review the daily facility issues and recommend disposition of identified issues.
- Ensure each issue screened is reviewed and classified to ensure necessary immediate actions have been taken and the owner is assigned to take additional actions as appropriate.
- Provides knowledge of represented department for Ownership Committee Meetings.
- Able to make decisions for represented department for resource commitments associated with CAP actions and products.
- Identify issues for station Rework Reduction Program and create assignments to address unexpected corrective maintenance (CMU) and rework issues.

3.7. Exelon Nuclear Personnel and Contractors at Exelon Nuclear Facilities (Originators):

- Identify conditions that have or could have an undesirable effect on performance of equipment, programs, or organizations.
- Ensure necessary immediate actions to place the situation in a safe and stable condition are completed or initiated as appropriate.
- Verbally report the condition to a supervisor or the Control Room, when appropriate, including communication of immediate corrective actions taken.
- Ensure that the issue is properly documented, with all required information and fields populated.
- Identify opportunities for improvement and commendable behaviors.
- Identifies if fieldwork is required.

3.8. Supervisor

- Discuss the details of the issue with the originator.
- Ensure appropriate immediate actions are taken.
- Notify the Control Room as appropriate.
- Ensure any safety issues (nuclear, radiological and industrial) are immediately addressed.

3.9. Operations Shift Management Reviewer

- Ensure appropriate immediate actions are taken including:
- Implement quarantine measures for areas, equipment, or records to preserve physical evidence as appropriate
- Prompt Investigation as directed by the OP-AA106-101-1001
- Determine impact on Operability and determine reportability.
- Determine impact on shutdown or online risk.
- Determine environmental risk.

## 4. MAIN BODY

4.1. Precautions

4.1.1. Ensure appropriate immediate actions are completed or initiated to place the plant in safe condition or temporarily restore the deficient condition before documenting the issue.

4.1.2. Documenting an issue does **not** substitute for proper communications with personnel impacted by, or who should be aware of, the condition.

4.1.3. Personnel should correct any identified condition to the extent possible as soon as practical.

4.2. Limitations

4.2.1. **Only** use the Issue Initiation Website (when available), accessible from the Exelon Home Page to document an issue.

4.3. Issue Origination

NOTE: Guidance is provided in Attachment 4 for different Functional Areas to provide guidance on the types of issues, as a minimum that should be identified.

NOTE: Individuals identifying an escalating condition (e.g. a valve leak increasing from the original value of 5 drops per minute to 30 drops per minute) should contact the main control room to ensure the related Work Order is updated. If the previously identified condition has deteriorated to the point the operability or reportability is affected, a new IR should be generated.

NOTE For performance management issues, individual names and specific actions taken should not be contained in Issue Reports. It is recommended to use job titles and verbiage such as "coaching."

4.3.1. **IMPLEMENT or INITIATE** appropriate immediate actions upon discovery of an issue to ensure the following:

1. Equipment, area, or situation is in a safe and stable condition;
2. Control Room, radiation protection, chemistry, security or other departments are notified to ensure appropriate immediate actions are taken;
3. Contact the Affected Facility/Unit Operations Shift Management to discuss potential Operability or Reportability of the issue.

NOTE: For potential safety issues, ensure supervisor is aware of safety concern and the supervisor is then accountable to ensure immediate/interim action is taken. The supervisor should update the IR, if no immediate/interim action is required.

4.3.2. **VERBALLY CONTACT** your immediate supervisor to ensure necessary immediate actions are taken and appropriate routing is applied.

1. **If** the Originator's immediate supervisor is **not** available, **then CONTACT** another facility supervisor.

2. **If** anonymity is desired, **THEN EXIT** this procedure and **REFER** to EI-AA-101, "Employee Concerns Program."

NOTE: For equipment issues, all known information pertaining to the equipment should be provided, including the component identification.

4.3.3. **ORIGINATE** the issue in the Computer Program.

1. **If** the Computer Program is **not** available, **then COMPLETE** Attachment 1, "Issue Reporting Form" and provide the completed form to your supervisor.

NOTE: An immediate review of an issue is only required when immediate Operations actions are required, because the issue is screened by an Ownership Committee within one business day.

4.3.4. **ROUTE** the issue for immediate review by Operations Shift Management if immediate actions are required by Operations.

4.4. Operations Shift Management Review

4.4.1. **ENSURE** that appropriate immediate actions have been implemented or initiated to place the plant in a safe condition or temporarily restore the deficient condition, upon notification of the issue.

4.4.2. **COMPLETE** required Shift reviews within the same shift as the notification **or ENSURE** the issue will be addressed by the oncoming shift, with the exception of an Operability Determination which should be completed within a 24-hour period, in accordance with Reference 6.8.

4.4.3 **DETERMINE** whether the condition impacts shutdown or online risk in accordance with Reference 6.4, 6.18, 6.19, and/or 6.20. **(CM-1)**

4.4.4 **REVIEW** the issue utilizing the applicable Computer Program.

4.4.5 **DETERMINE** environmental risk, such as spills or potential NPDES non-compliances.

4.4.6 **PERFORM** the following:

1. **INITIATE** appropriate work management document, if immediate action is required in accordance with Reference 6.4, 6.9, and/or 6.15, as applicable.

2. **DOCUMENT** whether the condition impacts a Technical Specification Function.

3. **DETERMINE** if the Operability of any system, structure, or component (SSC) is affected by the condition described in the issue and document the basis of the determination.

   **INITIATE** an Operability Evaluation Assignment and notify the assigned organization in accordance with Reference 6.8, if additional information or analysis is required to determine Operability.

4. **DETERMINE** if issue is Reportable in accordance with Reference 6.5 and document the basis of the determination and actions.

5. **INITIATE** a Prompt Investigation by initiating a Prompt Investigation Assignment and notify the assigned organization, if required in accordance with Reference 6.7.

6. **DOCUMENT** any additional comments in the Issue.

4.5. <u>Supervisor Review</u>

NOTE: The documented Supervisor review is optional and is only performed when requested by the Originator or when the Ownership Committee **<u>cannot</u>** determine the appropriate actions based on the originator's information.

4.5.1. **VERIFY** that appropriate immediate actions have been implemented or initiated to place the plant in a safe condition or temporarily restore the deficient condition, upon verbal notification of an issue.

4.5.2. **CONTACT** the Affected Facility/Unit Operations Shift Management to discuss, when issue has potential Operability or Reportability impact.

4.5.3. **REVIEW** the documented issue by accessing the Computer Program.

4.5.4. **DOCUMENT** the resolution of any questions in the Reviewer comments and generate any assignments and complete the review through the Computer Program.

4.6. <u>Ownership Committee</u>

NOTE: For conditions adverse to non-radiological environmental permit or regulatory requirements, contact the Corporate Nuclear Environmental Manager for assistance with the issue electronically or with Attachment 1.

4.6.1. **OBTAIN** the "Station Ownership Committee (SOC) Report" and perform the review of the issues prior to the Station's Ownership Committee Meeting.

4.6.2. **VERIFY** that a quorum is present for the Ownership Committee meeting. A quorum consists of a minimum of five Ownership Committee Members with the following discipline's knowledge represented consistent with the requirements of reference 6.4:

- Current Licensed Senior Reactor Operator (SRO) (Nuclear Duty Officer-Corporate) (Note: A Current Licensed SRO is required to review all plant equipment issues and those issues that impact the facility's operating license. An SRO knowledgeable member may substitute, provided those plant equipment issues and those issues impacting the plant operating license have a documented review by a current Licensed SRO)
- Regulatory Assurance/Licensing
- Maintenance
- Engineering
- Work Control
- Work Planning
- FIN
- Radiation Protection
- Chemistry

4.6.3. **VERIFY** all issues are reviewed and documented for the following:

1. **INITIATE** the appropriate work management document, if immediate action is required, in accordance with References 6.4, 6.9, and/or 6.15, as applicable.

NOTE: The follow-up should be completed within 5 business days from origination of the issue, or as directed by the Ownership Committee.

2. **ROUTE** the issue to the appropriate supervisor for follow-up, if sufficient information is **<u>not</u>** provided to assign issue significance or ownership.
3. **ROUTE** the issue to the appropriate supervisor to define the scope of the investigation when an investigation is determined to be appropriate.
4. **DOCUMENT** whether the condition impacts a Technical Specification Function.

5. **DETERMINE** impact on Operability of any system, structure, or component (SSC) affected by the condition described in the issue and document the basis of any "No" determination.

   A. **INITIATE** an Operability Evaluation Assignment in accordance with Reference 6.8, if additional information or analysis is required to determine Operability.

6. **DETERMINE** if issue is Reportable in accordance with Reference 6.5 and as necessary, document the basis of any "Yes" determination and actions.

7. **INITIATE** a Prompt Investigation by initiating a Prompt Investigation Assignment and notify the assigned organization, if required in accordance with Reference 6.7

8. **DETERMINE** if Operational and Technical Decision Making Process applies to the issue in accordance with Reference 6.12.

NOTE: NOS Findings and Adverse Findings, as defined in Reference(s) 6.13 and 6.14, shall at a minimum be assigned an evaluation and routed to MRC review.

NOTE: The investigations associated with an NRC Reportable Event (including a 60-day verbal report) are required to be reviewed by the Plant Onsite Review Committee.

9. **DETERMINE** the Significance and Investigation Class of the issue in accordance with Attachment 2, "Issue Report Level and Class Criteria" and Attachment 3, "Guidance for Determining Investigation Class".

NOTE: For Level 1, 2, or 3 issues, if a formal investigation is **not** recommended, a known cause statement should be documented in the body of the Issue Report.

10. **DETERMINE** the following for Level 1, 2, or 3 issues:

    A. Operating Experience (Nuclear Notification Operating Experience (NNOE) or Nuclear Event Report (NER)) in accordance with Reference 6.11.

    B. Regulatory (Significance Determination Process (SDP) evaluation) in accordance with Reference 6.6

    C. For Critical Component Failures, an Equipment Apparent Cause Evaluation (EACE) is required unless the Site Engineering Director determines an EACE is not required.

11. **DETERMINE** the owning organization.

12. **DETERMINE** any additional actions that are necessary and create appropriate assignments.

NOTE: For conditions where the failed component or failure mode of a critical component cannot be determined, an OTDM shall be completed to assess the risks, and a Special Condition Assignment (SPC) shall be generated to track the issue until the cause has been determined. All other aspects of the issue e.g., programmatic, organizational, etc., shall be investigated, in accordance with procedure guidance.

13. **CREATE** a Special Plant Condition Assignment (SPC) to track issues involving critical components where the plant conditions will not allow the determination of the failed component or failure mode and/or offsite analysis of the failure component is required.

NOTE: If a formal investigation (e.g., Root Cause, Apparent Cause or Common Cause Analysis) is **not** determined to be required, a Work Group Evaluation (Class D) can be assigned. The purpose of the Work Group Evaluation is for the Department Supervisor to address any specific questions from the SOC or complete a minimal investigation into the condition to further define the problem, the cause, and extent of condition commensurate with the significance of the issue. Reference 6.2 provides additional direction for what is required in this evaluation.

14. **ROUTE** the issue to the appropriate organization for evaluation and development of action plan when necessary, in accordance with Reference 6.2.

15. DETERMINE suspected rework issues or unexpected corrective maintenance work requests and generate appropriate actions, in accordance with Reference 6.21.

NOTE: The Issue screening process serves as an initial review to determine if there are other programmatic impacts associated with an issue, but does **not** serve as the only barrier.

16. **IDENTIFY**, when known, other programmatic or organizational impacts such as:

    A. Active Human Performance Error/Event Clock Reset

B. Potential Safety Culture/Safety Conscious Work Environment implications

C. Critical Component Failure (CCF) Clock Reset

D. Station Rework Reduction Program

E. Maintenance Rule Functional Failure (Reference 6.16).

F. Mitigating System Performance Index (MSPI) Failure or MSPI potential failure

G. Maintenance Rule Performance Monitoring

H. Simulator Fidelity

17. **DOCUMENT** any additional comments for the issue.

## 5. <u>DOCUMENTATION</u>

5.1. Guidance on retention of records can be found in Reference 6.3.

## 6. <u>REFERENCES</u>

6.1. 10 CFR 50, Appendix B, Criteria XVI,

6.2. LS-AA-125, "Corrective Action (CAP) Procedure"

6.3. LS-AA-127, "PassPort Action Tracking Management Procedure"

6.4. WC-AA-106, "Work Screening and Processing"

6.5. Exelon Reportability Reference Manual

6.6. LS-AA-2002, "Significance Determination Process Evaluation"

6.7. OP-AA-106-101-1001, "Event Response Guidelines"

6.8. OP-AA-108-115, "Operability Determinations"

6.9. CC-MW-101, "Engineering Change Request"

6.10. Quality Assurance Topical Report

6.11. LS-AA-115, "Operating Experience Procedure"

6.12. OP-AA-106-101-1006, "Operational and Technical Decision Making Process"

6.13. NO-AA-210, "Nuclear Oversight Regulatory Audit Procedure"

6.14. NO-AA-220, “Nuclear Oversight Performance Assessment Procedure"

6.15. CC-AA-103, “Configuration Change Control” (CC-MA-103-1001, “Implementation of Configuration Changes”)

6.16. ER-AA-310, “Implementation of Maintenance Rule”

6.17. ER-AA-1200, “Critical Component Failure (CCF) Clock”

6.18 OP-AA-101-111, "Roles and Responsibilities of On Shift Personnel"

6.19 OU-AA-103, "Shutdown Safety Management Program"

6.20 WC-AA-101, "On-line Work Control Process"

6.21 MA-AA-716-017, “Station Rework Reduction Program”

6.22 Station Commitments

6.22.1 Byron

**CM-1** IR 759945 (Steps 4.4.3)

7. **ATTACHMENTS**

7.1. Attachment 1, “Issue Reporting Form”

7.2. Attachment 2, “Issue Report Level and Class Criteria”

7.3. Attachment 3, “Guidance for Determining Investigation Class”

7.4. Attachment 4, “Functional Area Threshold Guidance”

**ATTACHMENT 1**
**Issue Reporting Form**
**Page 1 of 2**
**Originator Data**

Circle the appropriate discovery method: Self-identified Externally identified Event

Origination Date: ____________ Origination Time: __________

Discovery Date: ____________ Discovery Time: __________

Event Date ____________ Event Time: __________

Affected Facility: ____________ Affected Unit: ____________ Affected System ____________

Equipment/Component Number: ________________________________________________

Subject: ________________________________________________

Originator Name ____________ Originator User ID ________ Originator Dept. ____________
Originator Phone # ____________

Condition Description (The inappropriate action or equipment problem AND its negative result):

________________________________________________

________________________________________________
(Use reverse side if necessary.)

Optional Additional Information

Activities, processes, procedures involved: ________________________________

Why did condition happen: ________________________________

Consequences: ________________________________

Requirements impacted: ________________________________

Adverse physical conditions: ________________________________

Who was notified: ________________________________

Knowledgeable individuals: ________________________________

Repeat or similar condition: ________________________________

Immediate Actions and Recommended Actions: ________________________________
(Use reverse side if necessary.)

**Personally contact Supervision**

Name of Supervisor contacted ____________
Supervisor Department ____________
Supervisor Phone # ____________

Operations Shift Management Review Required: ___Yes ___No

Name of Operations Shift Management Contacted: ________________________________

Date Contacted (MMDDYY): ____________ Time Contacted (00:00): ____________

**ATTACHMENT 1**
**Issue Reporting Form**
**Page 2 of 2**
**Operations Shift Management Review Data**

Additional Immediate Actions: ______________________________
(Use reverse side if necessary.)

Prompt Investigation: _____No _____Yes

Operable: _____No _____Yes

Operability Evaluation: _____No _____Yes Assignment No.: ______________

Reportable: _____No _____Yes Reportability Manual Ref: ______________

Organization Notified: ______________ Notification Date (MMDDYY): ______________

Notification Time (00:00): ______________ Reviewer Name: ______________

Review Date (MMDDYY): ______________ Review Time (00:00): ______________

**Screening Section**

Follow-up Review: _____No _____Yes Assigned Group: ______________

Field Work Required: _____No _____Yes ECR Required: _____No _____Yes

Close to WR/AR: _____No _____Yes If No, Assigned Group: ______________

Recommended Significance Level: __1 __2 __3 __4 __5

Recommended Investigation Class: __A __B __C __D

Identify immediate and interim actions (include extent of condition issues (procedures, equipment, etc.) that require immediate actions: ______________________________

______________________________

______________________________
(Use reverse side if necessary.)

Optional Trend Data:

Problem Type: ______________ Error Precursor: ______________ Failed Defense: ______________

Initiating Action: ______________ Latent Org./Program Weakness: ______________

Reviewer Name: ______________ Date (MMDDYY): ______________ Time (00:00): ______________

**Purpose of Significance Level Assignment**: The Significance Level provides a measurement to Station and Exelon management of how effectively the organization is learning from lower level Issues. Different levels are assigned to each Condition Report to define the actual consequence of the issue. An organization that has a **very** low average number of Level 1 and 2, a limited number of Level 3 issues and an appropriate number of Level 4 issues commensurate with station performance, is effectively learning from minor events and preventing significant events. Issues should be classified using the highest applicable level. The significance characterization of some issues may change following additional analysis by internal or external organizations and is assigned based on the judgment of Ownership Committee (SOC). If the investigation identifies that a higher Significance Level may need to be assigned, then the issue shall be reviewed with SOC to determine if the Significance Level should be changed.

## Significance Level

***1 –An Event that results in a major impact as defined in this Attachment***

***2 –An Event that results in a moderate impact defined in this Attachment***

***3 –An Event that result in a minor impact defined in this Attachment.***

***4 –Low level problem typically closed to immediate actions taken or planned actions. Allows coding and trending of issues.***

***5 –An Enhancement, not for trending.***

## Clock Reset Guidance

The guidance for Station and Critical Component Failure Clock Resets is integrated into this guidance as follows (Note: Department/Crew Clock Resets are responsibility of owning Department Management):

***Station Clock Reset***: Issues that result in a station clock reset consistent with OP-AA-101-113-1001 should, as a minimum, be classified as a Level 3 event.

***Critical Component Failure Clock Reset***: Issues that should be considered for a Critical Component failure clock reset consistent with the requirements of ER-AA-1200 are a direct result of a component failure and are either classified as a Level 1 and 2 Event or a Level 3 event that meets the criteria designated by ***CCFCLK***.

## ******Significance Level 1 Examples******

**Operational Execution – Reg/Nuclear Safety**

- Significance Determination Program (SDP) evaluation or NRC Performance Indicator (PI) is designated as YELLOW or RED
- Declaration of an Alert or higher emergency plan classification
- Exceeding a plant Safety Limit
- Receipt of an NRC Level I, II, or III Violation (**not** associated with the NRC SDP as defined in NUREG-1600)
- INPO Significant Event

**Financial & Generation**

- The loss of 100 MWE or more generation for more than 30 days due to a single event
- Unplanned plant outage or planned outage extension of greater than 7 days from a single event.

**Asset Management**

- Unplanned significant cost to organization ($\geq$ \$750,000 from single issue excluding replacement energy costs)
- Any Equipment Failure that results in a Level 1 Event.
- Fuel damage due to improper reactivity control

**Workforce-Personnel/Plant Industrial Safety**

- Incident that results in a fatality or permanent disability

## ******Significance Level 2 Examples******

**Operational Execution – Reg/Nuclear Safety**

- SDP Evaluation or NRC PI is designated as WHITE
- Licensee Event Report (LER) or optional telephone notification to the NRC Operations Center within 60 days after the discovery of the event (as defined in 10 CFR 50.73).
- Extension to planned/scheduled Shutdown Technical Specification Action (TSA)/Limiting Condition for Operation (LCO) window of greater than 50%, where the Allowed Out of Service Time exceeded 95% of the total time.
- Operation of the plant or Dry Fuel Storage System in a beyond-design basis condition.
- Unplanned increase of shutdown or on-line risk to orange or red color.

**Operational Execution – Reactivity Management**

- Reactivity Management Program Level 1 or 2 Event as defined by OP-AA-300-1540.

**Operational Execution – Radiological**

- - RP personnel exercise "Stop Work" authority and the work group does **not** adhere to it
- Over exposure above admin limit.
- Personnel contamination resulting in greater than or equal to 25 REM shallow dose equivalent from discrete particle (50% of the NEI limit).
- Any very high Rad Area occurrence as defined by NEI-99-02.
- Radioactive material or material with removable surface contamination is found above 1,000 dpm/100 cm2 beta/gamma or above 20 dpm/100 cm2 alpha outside the RCA
- Violation of Tech. Spec with actual radiation levels greater than 1,000 mrem/hr
- Cited violation of NRC or DOT radioactive material shipment regulations
- Lost or missing Licensed RAM > 1000 x App. 'C'
- Overexposure or unintended exposure $\geq$ 100 mrem
- Public exposure due to RAM outside the RCA > 5 mrem
- Violation of NRC spent fuel storage regulations
- Dry storage system leakage greater than Tech. Spec. Limits

**Operational Execution – Emergency Preparedness**

- Failure to implement or meet a risk significance planning standard of 10CFR 50.47, Emergency Plans, and 10CFR 50 Appendix E
- Failed Exercise

**Operational Execution – Chemistry/Environmental**

- Non-compliance with an environmental permits limit or environmental permit condition including NPDES non-compliance that results in a violation.
- Hazmat event as classified by response team leader.

**Financial & Generation Commitments**

- Reactor Scram
- Loss of greater than 20% power from a single unplanned issue
- The loss of 100 MWE or more generation for more than one day due to a single event

**Asset Management-Equipment Reliability**

- Unplanned large cost to organization (> \$100,000, but < \$750,000 from a single issue excluding replacement energy costs)
- Un-recovered material in the refueling cavity or fuel pool which could or does result in undesirable consequences
- Issues or events represent a loss of safety function of a single train for longer than the Tech. Spec. Allowed Out of Service Time

**Workforce-Personnel/Plant Industrial Safety**

- OSHA Lost Work Day Case

## ******Significance Level 3 Guidance******

**Operational Execution – Reg/Nuclear Safety**

***CCF*** -Unplanned TSA/LCO entry requiring plant shutdown consistent with the definition found in Performance Indicator OM.01.

- Entry into Abnormal/Emergency Operating/Off-normal procedures due to a valid plant transient except for enunciator procedure
- Conditions that draw negative media attention, resulting in a press release.
- The declaration of an Unusual Event
- Issues or events requiring an immediate NRC notification (as defined in 10 CFR 50.72).
- Receipt of NRC Non-Cited Violation (NCV) or NRC finding
- Failure to meet a programmatic non-nuclear regulatory obligation (FAA, OSHA, DEP, EPA, DOT, FERC, SRBC)
- INPO Areas for Improvements (AFIs)
- Issues requiring submittal of a 10CFR 21 report
- Failure to perform a Surveillance Requirement within the required time
- Failure to meet a Technical Specification related Surveillance Procedure acceptance criteria
- Ineffective CA/CAPR as determined by a collective EFR or an ineffective CAPR as determined by a single EFR
- Inadequate causal analysis resulting in: Repeat Level 1,2, or 3 Event or inappropriate CA/CAPR
- Untimely CA/CAPR that results in undesirable consequences
- NOS Finding

**Operational Execution – Fundamentals**

- Failure to comply with Level 1 procedure criteria
- Failure to follow a Level 2 or 3 procedure that results in undesirable consequences
- Equipment status control discrepancies that result in undesirable plant condition
- Configuration management discrepancies that result in undesirable plant condition
- "Near miss" conditions that under different circumstances would reasonably be expected to result in a Significance Level 1 or 2 Event.
- Errors in calculations/data reduction that invalidate surveillance tests
- Inadequacy/inaccuracy in procedures or guidelines that caused or could have caused unexpected operation or inoperability of equipment, or results in equipment damage.

**Asset Management-Equipment Reliability**

***CCFCLK*** – Any consequential Critical Component Failure (CCF) as defined in ER-AA-1200.

- Unplanned increase in shutdown or online risk by one color.
- Maintenance Rule Functional Failure or potential Maintenance Rule a(1) Condition, IAW Maintenance Rule Procedure).
- Mitigating System Performance Index (MSPI) Failure
- Foreign material (FME) found or left in plant systems.
- An unplanned half scram (BWR-1/2 logic; for PWR 1/3 logic or ¼ logic where 2/4 will cause a Reactor trip).

# ATTACHMENT 2
# "Issue Report Level and Class Criteria"
# Page 2 of 2

## ******Significance Level 3 (Cont'd) ******

**Operational Execution – Reactivity**

- Reactivity Management Program Level 3 events as defined by OP-AA-300-1540.

**Operational Execution – Radiological**

- Failure to complete RWP access and exposure control documentation
- Violation of Radiation Protection procedures, RWPs, signs, or postings with the potential to cause significant radiological consequences.
- Contamination, airborne radioactivity, or radiation levels significantly above normal levels resulting from unplanned events.
- Improper work practices or operation of equipment which have the potential to result in significant:
  - Skin or clothing contamination
  - Unexpected spread of contamination
  - Increase in worker radiation dose
- Poor worker practices with the potential to cause significant radiological or industrial safety concerns.
- Improper handling of radioactive material at a contaminated area boundary (Step Off Pad) that result in the spread of contamination.
- Eating, drinking, or smoking in the RCA.
- Improper operation of equipment or lack of adherence to contamination boundaries causing a spread of contamination.
- Failure to maintain the Rad Log or improper storage or control of SNM or Rad materials in the warehouse.

**Operational Execution – Emergency Preparedness**

- Failure that requires compensatory measures to meet 10CFR 50.47, Emergency Plans, and 10CFR 50 Appendix E
- Failure to implement or meet a **non**-risk significance planning standard of 10CFR 50.47, Emergency Plans, and 10CFR 50 Appendix E
- Failed Drill required per section II.N.2 of NUREG-0654 (**not** communications drill)
- Failure of EP related systems, equipment, scenario or procedures that would have precluded the implementation of the emergency plan.
- Failed overall objective or DEP opportunity.
- Scenario issue resulting in misclassification, controller interjection or a failed performance indicator.
- Failed Facility Objective
- Failure to provide required 10CFR50.4 or other regulatory submittals to Regulatory Agencies.
- Significant ERO staffing or augmentation issues (i.e., Minimum staffing of ERO position less than three deep for greater than one month or unqualified ERO personnel on-call.

**Operational Execution – Security**

- Security Reportables
- Access authorization revoked based on discovery of inaccurate or incomplete information
- Security report per 10CFR 73.71 or inadvertent weapon discharge.

## ******Significance Level 3 (Cont'd) ******

**Operational Execution – Chemistry/Environmental**

- Any unplanned exceeding of Action Level 1, 2 or 3.
- Exceeding any Chemistry limit as listed in an approved system chemistry control procedure.
- Any chemical or hazardous waste spillage reportable to an outside agency or any spillage that exceeds thresholds established in site procedures.
- Hazardous waste generation in excess of 1000 kg in one month.
- Any mixed waste generation at a PWR or cumulative mixed waste generation in excess of 150 kg in one year at a BWR.
- Any shipment of hazardous material, hazardous waste or radioactive waste that results in a spill or leak.
- Unplanned generation of large amounts (> 250 Kg) of hazardous waste from a single job.
- Repeated failures of work group to ensure controlled materials are properly labeled and stored.
- Repeated failures of chemical storage cabinet owners to perform required inspections, housekeeping integrity and paperwork for Chemical Storage Cabinets.
- Non-compliance with an environmental permit limit or environmental permit condition including NPDES non-compliance.

**Financial & Generation Commitments**

- Any issue that results in an unplanned power rise or an unplanned drop in power output of greater than 5%
- Extension of planned/scheduled TSA/LCO work window of greater than 5%, where the Allowed Out of Service Time exceeded 75% of the total time
- Significant Unplanned Expenditure (i.e., >$50,000)
- Undesired effect (high-impact or consequential event) on major equipment and support systems needed for plant safety or power production (i.e., inadvertent trip/start, mis-operation, improper maintenance that results in significant delays in returning the equipment to service or damage to the equipment, wrong unit/train error).
- Extension in the outage > 1 day from a single event.

**Operational Execution – Training**

- Examination security is compromised.
- Inadequacy/inaccuracy in training materials that results in a performance-based problem in the plant.
- Training activity, (e.g., inaccurate record keeping, failing to maintain training material, etc.), that results, or has the potential to result in non-qualified personnel performing work.
- Unexcused missed training for licensed operator training.
- Any deficiency that indicates one or more accreditation objectives may **not** be met.
- In-field training activity performed on the wrong unit, system, train, or component that has organizational impact to plant operations.
- Technically inaccurate material is used to conduct a training activity
- Any individual initial NRC license exam failure.
- More than 15% of licensed operators fail any portion of the annual or biennial exam.
- Any crew failure during simulator evaluation.
- SSTC or TAC determination that a training program is ineffective.
- Throughput goal of I< 80% for ROs and < 85% for SROs for removal of a student.

## ******Significance Level 3 (Cont'd) ******

**Workforce-Personnel/Plant Industrial Safety**

- Fire requiring application of extinguishing agents by site fire brigade to control or that results in significant damage to plant structures and/or equipment.
- OSHA Violation
- Personnel injury that is OSHA recordable
- Personnel injury caused by failure of an individual to adhere to established personnel safety guidelines.
- Personnel error that caused or could have caused serious personnel injury, equipment damage, or equipment inoperability.
- Any vehicle accident that results in personal injury or damages.
- Rejection of an approved operability evaluation due to an inadequate technical evaluation.
- Exceeding overtime guidelines without approval per LS-AA-119.
- MRC or PORC rejection of a department generated document.

## Investigation Class

***(Note: Attachment 3 provides further guidance)***

***A – Root Cause Analysis (RCA) is required to determine the root causes and corresponding Corrective Actions to Prevent Recurrence (CAPRs).***

***B – Apparent Cause Evaluation (ACE) is required to determine the apparent cause and corrective actions.***

***C – Common Cause Analysis (CCA) is required to determine if there are any common failure modes for an apparent adverse trend.***

***D – No formal investigation is required to determine causes or corrective actions.***

| | | Uncertainty | | |
|---|---|---|---|---|
| | | **High** | **Medium** | **Low** |
| **Risk** | **High** | A | A | B |
| | **Medium** | A | B | D |
| | **Low** | B | D | D |

Risk: Risk involves two elements: consequences (actual and potential), and probability of recurrence (assuming no corrective actions are taken). The higher the consequence and the probability the greater the need to ensure effective corrective actions and therefore, the greater the need to utilize formal analysis techniques.

Uncertainty: Uncertainty involves two elements: uncertainty regarding the cause and uncertainty regarding the corrective actions. Uncertainty is directly related to the complexity of the event. The more complex (i.e., the more problems) the greater then uncertainty and the need to utilize formal analysis tools.

**ATTACHMENT 3**
**Guidance for Determining Investigation Class**
**Page 1 of 3**

The following matrix should be used to determine the class of investigation required for a particular issue. It should be understood that this guidance is designed to aid in determining the appropriate class of investigation to be applied to an issue. This guidance does **not** supercede external requirements that may **mandate** a certain level of investigation. The following are provided as examples:

- A full Root Cause Evaluation shall be performed for any White, Yellow or Red NRC Inspection Finding or NRC Performance Indicator or a degraded cornerstone.

- Strong consideration should be given to perform a full Root Cause Evaluation when the issue involves an LER or an Adverse NOS Finding, as identified in References 6.13 and 6.14.

- Strong consideration should be given to performing an apparent or root cause evaluation for any issue that indicates that a training program may **not** meet one or more accreditation objectives.

- If the actions to limit future unplanned failures are **not** known, strong consideration should be given to performing at least an Apparent Cause Evaluation for:

    - All externally identified Significance Level 3 or above issues.

    - Any Critical Component Failure Clock Reset (Site Engineering Director approval is required to perform any investigation lower than an EACE.)

    - Component failure that if it had occurred while the system or plant was in-service would have resulted in a plant trip or derate.

    - For any critical component that fails between PM intervals

    - Any non run-to-failure component that fails and results in a trip, derate or entry into a short duration LCO.

The following definitions of risk and uncertainty should be used in the analysis to determine the class of investigation for an Issue.

A matrix has been provided below to provide guidance as to the investigation class that should be used depending upon the risk revealed by the event/condition and the amount of confidence that exists regarding the cause of the event/condition and how effective the corrective actions will be at preventing a recurrence.

**Risk**: Risk involves two elements, consequences (actual and potential), and probability of recurrence (assuming no corrective actions are taken).

The actual consequence of an event or condition can be determined by using the "Significance Level".

**ATTACHMENT 3**
**Guidance for Determining Investigation Class**
**Page 2 of 3**

In determining the potential consequence, consider **not** only what happened but also what could have happened if the circumstances were different. For example, if under different circumstances additional components could be rendered inoperable or a more significant event could have occurred, then the potential consequence may be higher. For equipment, issues that could result in the same problem in a different system or a greater consequence in a different plant-operating mode, a higher risk should be assigned.

An example would be a relay failure in an annunciator circuit that provides alarm function only may have a smaller risk than the same model relay installed in the feedwater level control circuitry, but if the failure is determined to be age related, the risk of failure of similar related relay failures throughout the plant needs to be considered.

It should also be understood that risk could involve issues such as Technical Specification Operability, compliance with federal, state and local requirements, insurance requirements, and violation of NRC requirements. All these considerations and others should be evaluated when determining the consequence of an issue.

The probability of occurrence can be determined qualitatively -- "how likely is recurrence of the actual or potential consequence?" This should be based on the evaluator's experience and knowledge of the occurrence of previous similar issues, including Exelon and industry operating experience. The evaluator should determine if this is a one-time occurrence or is it likely to repeat. For equipment issues, the number of the same components in service at a plant will increase the likelihood of repeat failures.

In addition, risk could be associated with a reduction in margin. For example, if a condition resulted in a reduction in the design or operating margin in a calculation or equipment, this type of impact should be considered in the overall risk associated with the identified problem.

**Uncertainty**: Uncertainty involves two elements, uncertainty regarding the cause and uncertainty regarding the corrective actions.

Uncertainty is how well the what, how and why of the issue is understood. When determining uncertainty, consider the following:

- How many different problems led to the issue? The greater number of unique problems leading to the event, the more complex the issue. With increase complexity, the uncertainty would increase.

- Does a good problem statement exist (i.e., who or what did what that resulted in the issue)? If low confidence in the problem statement exists, then the uncertainty increases.

**ATTACHMENT 3**
**Guidance for Determining Investigation Class**
**Page 3 of 3**

- For equipment issues, uncertainty involves several elements that need to be considered to ensure confidence that the problem can be adequately addressed with the actions taken or the planned actions. The following provides discussion of some key elements that should be considered in determining the uncertainty of equipment issues:

    - **Failure Mode**: Failure mode is the method by which the component failed. For example, a pump has a failure mode of decreasing flow. Uncertainty around the failure mode would result in the organization **not** being able to detect the failure of the equipment. Understanding the failure mode and the ability of the organization to monitor and detect a failure prior to a consequential event can be used to justify a low uncertainty, even if the specific component and the cause were **not** known.

    - **Component that failed**: The specific component or subcomponent that failed is key to understanding the cause of the failure. If the specific component that failed is **not** known, then the cause **cannot** be determined and therefore, the organization would have a higher uncertainty as to the right corrective actions and in this case would drive for additional investigation/troubleshooting.

    - **Cause of Failure**: The cause is the programmatic reason why the specific component is **not** reliable. Uncertainty would be high if the cause is **not** known and the failure mode was **not** detectable for the specific component. At this point it would be critical to determine what programmatic problem (e.g., operating practices and maintenance programs) have **not** been effective at preventing failure and the consequential event.

| | | *Uncertainty* | | |
|---|---|---|---|---|
| | | *High* | *Medium* | *Low* |
| *Risk* | *High* | ***A*** | ***A*** | ***B*** |
| | *Medium* | ***A*** | ***B*** | ***D*** |
| | *Low* | ***B*** | ***D*** | ***D*** |

For example, if the condition created minor or no consequences, and the probability of recurrence is low, the risk would be low. If there is confidence that the cause is understood and there is confidence that corrective actions will adequately address the cause, then the uncertainty is low. Therefore, investigation Class D should be recommended. If the cause is **not** initially known and investigation is required to determine the cause of failure then a higher level of uncertainty should be assigned and some level of formal investigation would be necessary.

NOTE: The risk vs. uncertainty matrix is one of several inputs to determine the level of causal analysis.

**ATTACHMENT 4**
**Functional Area Threshold Guidance**
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**NOTE**

This attachment provides examples defined by functional area of typical issues that should result in a documented Issue in order to aid in the consistent application of the Issue initiation thresholds.

Examples, defined by the functional area, are provided as guidance in the next several paragraphs. This guidance is based largely on the individual circumstance and may involve management judgment. This information is **not** intended to provide examples for **all** potential issues but rather provide examples that may be analogous to many different circumstances. Strict adherence to these thresholds is **not** required but documentation of the determination as to why the specific issue may or may **not** meet the threshold should be considered.

The evaluator of an issue should consider whether the event is an anticipated response. Such an event may **not** require the generation of an Issue. For example: the actuation of an Area Radiation Monitor (ARM), where the actuation was an expected response and actions (established by existing procedures or a previous Issue) are in place to respond to the actuation, would **not** require the generation of a new Issue.

1.0. **WORK MANAGEMENT** (This attachment provides examples only and is not a complete list and shall not limit a person from documenting an IR for any problem):

1.1. Inappropriate direction given, via schedule or meeting, to remove required equipment from service.

1.2. Use of outdated/superseded documents/procedures/drawings resulting in inadequate scheduling of plant work.

1.3. Less than adequate work coordination that results in increased radiation exposure, reduction in plant safety, unnecessary challenge to plant equipment, or misuse of station resources.

1.4. Work Management programmatic trends identified through self-assessment, self-check programs, or performance gap analysis.

1.5. Adverse trends identified during inventory accuracy verification.

1.6. Material unavailability that does **not** meet the customer's expectations.

1.7. NOS/Supply Management receipt inspection problems or errors.

1.8. Formal vendor recommendations.

1.9. Safety related material discrepancies (i.e., 10CFR21).

1.10. Inadequate training in the use of critical scheduling tools (i.e., ORAM).

1.11. A planned increase in risk as a result of lack of training.

1.12. Entering a passing grade in Action Tracking for a failed Surveillance Test (ST).

1.13. Failure to adequately verify Action Tracking data, prior to committing, which results in a scheduling error.

1.14. Identification of adverse trends in the ST Program.

1.15. Inadequate work coordination resulting in removal of wrong system, train, component, or device.

**ATTACHMENT 4**
**Functional Area Threshold Guidance**
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1.16. ST work order activity completion on wrong work order.

1.17. Missed ST resulting in a challenge to Tech Specs.

1.18. Failure to identify entries into abnormal operating conditions/LCOs prior to issuance of Rev. 0 schedule.

1.19. Critical path delays.

**2.0** **MAINTENANCE** (This attachment provides examples only and is not a complete list and shall not limit a person from documenting an IR for any problem):

2.1. Check Point or Hold Point (QA) misses.

2.2. Maintenance personnel error during a maintenance work activity that results in an extended TSA or LCO work window.

2.3. Mispositioned valves or other equipment following the completion of maintenance, testing, or calibration activities.

2.4. Maintenance Division activities which cause unplanned risk significant system or equipment inoperability.

2.5. Equipment failure that could have been prevented by the predictive monitoring program.

2.6. Potential Maintenance programmatic trends identified through observations, self-assessment, in-process maintenance issues, including recurrent non-compliance with plant rules.

2.7. Conditions of a repetitive or generic nature associated with hardware non-conformances that are **not** tracked by a work request and are **not** tracked as a chronic system problem.

2.8. Maintenance/I&C training materials that contain incorrect information that has the potential to lead to adverse performance results for the plant or personnel.

2.9. Procedures found to provide incorrect direction that would cause plant or personnel risk if followed.

2.10. Incomplete or erroneous data recorded during conduct of procedures which impact equipment operability.

2.11. Errors in calculations/data that invalidates surveillance test data.

2.12. Equipment readiness & reliability issues such as, failed PMTs, degraded or failed PMs, & parts delays as described in MA-AA-716-017.

2.13. Incorrect PMT specified and **not** corrected prior to performance or PMT **not** completed as specified.

2.14. Inoperable equipment is found to exist during review of Out of Tolerance Reports.

2.15. Unexpected conditions found during surveillance testing where preliminary troubleshooting does **not** resolve issue.

**ATTACHMENT 4**
**Functional Area Threshold Guidance**
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**3.0** **REGULATORY ASSURANCE** (This attachment provides examples only and is not a complete list and shall not limit a person from documenting an IR for any problem):

3.1. Errors found in UFSAR or Technical Specifications.

3.2. Reportability determinations that err on the non-conservative side (i.e. initial reportability determination 'N', but later changed to 'Y') or exceed reportability time requirements.

3.3. Regulatory Assurance personnel error that caused or could have caused serious personnel injury, equipment damage, or equipment inoperability.

3.4. Issues identified by outside agencies and are reportable to them that were **not** previously identified.

3.5. Regulatory Assurance document errors (i.e., LERs, Tech Spec Change Requests, NRC PI, etc.) that exist after the independent review process has been completed and are determined to have an impact.

3.6. Failure to provide relevant industry events to site organizations for review.

3.7. An inadequacy or inaccuracy in training materials that results in a performance based problem in the plant.

3.8. An inadequacy or inaccuracy in procedures or guidelines that causes Regulatory Assurance product or service errors.

3.9. Documentation/data/calculation errors that goes undetected following review/approval.

3.10. Transmittal of incomplete or inaccurate information in LERs, Tech Spec Change Requests, Non-Routine Reports, and other off-site communications.

3.11. Perceived programmatic administrative control trends for the following programs: OPEX, Nuclear Network, Commitment Tracking, and Corrective Actions.

3.12. NSRB or NEIL recommendations. Examples of NSRB items include:
- Issues that impact nuclear safety performance.
- Issues/deficiencies identified as part of the NSRB review process.
- Recommendations, issues or deficiencies identified in the NSRB subcommittee minutes or the NSRB meeting minutes executive summary.

3.13. The PORC Chair should direct that a Issue Report be initiated under the following conditions:
- An issue under review is rejected by PORC.
- PORC identifies conditions for approval that should have been identified in management reviews.
- PORC identifies editorial comments that indicate less than adequate pre-PORC review (e.g. typos, incorrect information, etc).
- Late withdrawal of the issue from the PORC agenda that resulted in unnecessary review of the issue by the PORC quorum.

3.14. MRC, PORC, or MRC rejection of a department generated document.

3.15. Corrective Action (CA) or Corrective Action to Prevent Recurrence (CAPR) assignments that are **not** completed in accordance with the requirements of LS-AA-125.

3.16. Overdue Action Tracking Item.

**ATTACHMENT 4**
**Functional Area Threshold Guidance**
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**4.0** **TRAINING** (This attachment provides examples only and is not a complete list and shall not limit a person from documenting and IR for any problem:

4.1. Non-compliance with Training Procedures.

4.2. Any deficiency that indicates one or more accreditation objectives may **not** be met.

4.3. In-field training activity performed on the wrong unit, system, train, or component.

4.4. Examination security is compromised.

4.5. Training activity, (e.g., inaccurate record keeping, failing to maintain training materials, etc.), that results, or has the potential to result, in non-qualified personnel performing work.

4.6. Technically inaccurate material is used to conduct a training activity.

4.7. Any individual initial NRC license exam failure.

4.8. More than 15% of licensed operators fail any portion of the annual or biennial exam.

4.9. Any crew failure during simulator evaluation.

4.10. Any shortfalls in Emergency Plan performance or EAL classification identified in the LORT end of cycle roll-up report. These shortfalls can include failures noted during exams or improper performance during training that was corrected by the instructor.

4.11. STC or TAC determination that a training program is ineffective.

4.12. Simulator unavailability that results in or could have resulted in lost training time.

4.13. Laboratory facility or equipment unavailability that results in lost training time.

4.14. “Near miss” in examination security where potential for compromise was created.

4.15. Any population of trainees where >20% and $\geq$2 individuals fail an exam or evaluation.

4.16. Training activities **not** held as scheduled for any reason.

4.17. STC/CRC/TAC **not** held as scheduled within the quarter.

4.18. Any training performance indicator changes to less than WHITE.

4.19. Any candidate is removed from an initial training program for any reason.

4.20. Training designed to improve performance that does **not** result in the expected improvement.

4.21. Any individual does **not** attend training as scheduled.

4.22. Potential trend(s) identified from reviews of management observations of training

4.23. Exam analysis identifies deficiencies in exam construction or grading.

**ATTACHMENT 4**
**Functional Area Threshold Guidance**
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4.24. Other significant simulator performance issues that impact training, simulator equipment failures, or restarts required of simulator computers.

4.25. If the removal of a student results in the throughput goal of less than 80% for ROs and less than 85% for SROs.

**5.0** **ENGINEERING** (This attachment provides examples only and is not a complete list and shall not limit a person from documenting an IR for any problem):

5.1. Procedure/Process Related Issues:

5.1.1. Engineering product errors (i.e., ECRs, Calculations, etc.) that have been issued for implementation that would have had impact on the operation or qualification of a system or component. Examples may include product errors resulting from personnel errors, procedure violations, breakdown in controls, or inadequate equipment status controls.

5.1.2. Equipment status control discrepancy that results in an adverse plant condition.

5.1.3. Equipment failure that could have been prevented by the performance-monitoring program.

5.1.4. Any device or component found out of its expected position.

5.1.5. Engineering personnel activity performance on the wrong unit, system, train, or component.

5.1.6. Engineering personnel error that could have caused serious personnel injury, equipment damage, equipment inoperability.

5.1.7. An inadequacy or inaccuracy in training materials that results in a performance based problem in the plant.

5.1.8. An inadequacy in a vendor's Engineering Product discovered during the owner's acceptance review. This should include, but is **not** limited to "non-station related technical human performance issues".

5.1.9. An inadequacy or inaccuracy in procedures or guidelines that caused or could have caused unexpected operation, inoperability of equipment, or results in equipment damage.

5.1.10. Errors in calculations, data reduction, data transmittal, or data verification that results in a performance based problem in the plant.

5.1.11. Risk significant plant system or component performance that is abnormal or is **not** the result of normal wear and is **not** tracked as a chronic system problem (e.g., EP/MC Focus List). Examples may include minor equipment damage, repeat equipment failures, and/or potential equipment trends.

5.1.12. Perceived engineering programmatic trends identified through self-assessment or self-check programs.

5.1.13 Configuration management discrepancies. Examples include plant modifications without appropriate procedure or drawing revisions

5.1.14. Software error in Class AA or BB software application.

5.1.15. Generic or repetitive software error in Class AA or BB application.

5.1.16. Improper use of an Engineering Change Process.

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**Functional Area Threshold Guidance**
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5.1.17. Identification of any unexpected impact on Design Margin. (e.g., Review of design calculation determined that an assumption was non conservative resulting in a direct impact on the design margin).

5.1.18. Improper 10CFR50.59 screening.

**6.0** **EMERGENCY PREPAREDNESS** (This attachment provides examples only and is not a complete list and shall not limit a person from documenting an IR for any problem):

6.1. Station occurrences resulting in declaration of an event and implementation of the Emergency Plan.

6.2. Failure of EP related systems equipment, scenario or procedures that would have precluded the implementation of the Emergency Plan.

6.3. Discovery of a failure of greater than 22% of the Emergency Sirens.

6.4. EP staff error that could have caused serious personnel injury, equipment damage, or equipment inoperability.

6.5. Failure to perform required surveillances, inventories or tests within the timeframes required for maintenance of the EP program.

6.6. Failure to provide required 10 CFR 50.4 or other regulatory submittals within the required time frame.

6.7. Transmittal of incomplete or inaccurate information in EP Submittals to Regulatory Agencies.

NOTE

The following guidance is to be used when assessing timeliness of corrective actions:

- A Risk Significant Planning Standard (RSPS) related drill/exercise performance WEAKNESS is typically corrected within 90 days of identification.
- A Planning Standard (PS) related drill/exercise performance WEAKNESS is typically corrected within 180 days of identification.
- Resolution of other drill/exercise performance WEAKNESSES is expected within the next evaluated biennial exercise cycle because of the lower risk significance of these efforts and expected lower priority of such efforts. EP-related corrective action systems may track enhancement suggestions that result from the drill program. These enhancement suggestions often add value to the EP program, but are not required and do not address WEAKNESSES. There is no NRC timeliness expectation for resolution of enhancement suggestions.

6.8. Failure to properly perform or demonstrate a Risk Significant EP Planning Standard during an actual event or evaluated Drill or Exercise. These Planning Standards include; Classification; Notification; Development of Protective Action Recommendations (PARs) and Off-site Dose Projections.

- ERO Staffing or augmentation issues, such as Minimum staffing ERO position less than three deep for greater than one month.
- Unqualified ERO personnel on-call.

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**Functional Area Threshold Guidance**


6.9. Station issues that were considered (evaluated) for classification but were determined to have **not** exceeded any Emergency Action Levels (EALs). This is **not** necessary if similar information is included as part of a Licensee Event Report (LER).

6.10. FEMA "Deficiencies" identified in an Exercise requiring demonstration in a remedial exercise/drill or by other remedial actions. This does **not** apply to FEMA identified ARCAs (Areas Requiring Corrective Action) if they do **not** require a remedial exercise/drill, and does **not** apply to ARFIs (Areas Recommended for Improvement).

6.11. An inadequacy or inaccuracy in approved EP related training materials, identified during actual presentation of the training, impacting the ability to complete the training as scheduled.

6.12. Any item related to any level 3 issue per 6.1 through 6.8 above, but below the associated threshold, for which trending is desired by the EP Manager.

6.13. Items as defined within the EP Administrative Maintenance procedures and T&RMs for which CAP trending has been specified.

6.14. Failed demonstration criteria. Minor equipment or scenario or procedure problems that did not impact performance.

6.15. ERO low level performance issues and enhancements that warrant trending

6.16 ERO Performance – Overall (not monthly) percentage value meets the following conditions:

- R.EP.01: < 93% and decreasing, or when a negative trend is identified.
- EPPI.01a-c: < 90% and decreasing, or when a negative trend is identified.
- EPPI.01d-e: < 90% and decreasing, or when a negative trend is identified.

6.17 ERO Readiness – Overall (not monthly) percentage value meets the following conditions:

- R.EP.02: < 85%, or when a negative trend is identified.
- EPPI.02a: < 85%, or when a negative trend is identified.
- EPPI.02b-c: < 50%, or when a negative trend is identified.
- EPPI.02d-e: Any minimum or non-minimum staffing ERO position is filled at 2 deep for the month.
- < 95% minimum staffing depth for the month.
- < 90% non-minimum staffing depth for the month.
- EPPI.02f: < 95% minimum staffing response for the 12 month rolling average.
- EPPI.02g: < 90% non-minimum staffing response for the 12 month rolling average.

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**Functional Area Threshold Guidance**
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6.18 Emergency Response Facilities and Equipment

- When the 12-month average percentage value of the Siren System Test is less than or equal to 97% operability, or when a negative trend is identified.
- Anytime the siren monthly operability report percentage drops below 94% operability.
- When the same type failure occurs to the same siren > 2 times during a 6 month period.
- Common failures occurring to the same model of siren when:
- > 33% failures of the same type occurs during a 6-month period if the quantity of sirens is > 10.
- > 50% failures of the same type occurs during a 6-month period if the quantity of sirens is < 10.
- Loss of a function listed in EP-AA-121, Attachment 1, ERF and Equipment Function Matrix.
- ERF Readiness is < 99% and decreasing, or when a negative trend over a 3- month period is identified.
- Equipment Availability is < 95% and decreasing, or when a negative trend over a 3- month period is identified.

6.19 Problem Identification & Resolution – Overall (not monthly) percentage value meets the following conditions:

- Significant variation occurs (>10%) in any category, or when a negative trend is identified.
- Data results indicate a doubling in % in any category not attributable to numeric changes in other categories or data drop off.

**7.0** **CHEMISTRY/RADWASTE** (This attachment provides examples only and is not a complete list and shall not limit a person from documenting an IR for any problem):

7.1. Any human performance event or condition adverse to quality resulting from procedural non-compliance, less than adequate communication of procedure/program/process change or errors contained in approved procedures.

7.2. Any other human performance event as determined by the Chemistry/Radwaste/Environmental Manager.

7.3. Any power reduction or derating of the unit that could have been prevented by proper chemistry controls (i.e., condenser fouling).

7.4. Any configuration control event involving a chemistry personnel error.

7.5. Technical Specification/Technical Requirements Manual (TRM)/ Offsite Dose Calculation Manual (ODCM) non-compliance.

7.6. National Pollutant Discharge Elimination System (NPDES), FESOP, Air Permit or other permit violation.

7.7. Incomplete or inaccurate information sent in a report (i.e., NEI Indicators, Discharge Monitoring Report, Fuel Warranty, etc.).

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**Functional Area Threshold Guidance**
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7.8. Performing an activity on the wrong unit, system, train, or component.

7.9. Negative trends identified through self-assessment, self check programs, or evolution critiques, after review by Chemistry/Radwaste/Environmental Management that require corrective actions to prevent further occurrence.

7.10. Any notice of violation received for chemistry, radwaste or environmental issues.

7.11. Radwaste Issues.

7.11.1. Radwaste system unavailability causing inadequate liquid or solid processing capacity such that plant operations or reactor chemistry is impacted.

7.11.2. Violation of a radioactive waste processor Waste Acceptance Criteria (WAC) or burial site acceptance criteria.

7.12.1. All Issues:

1. Any planned exceedance of EPRI Action Level 1, 2 or 3.
2. Any potential trend for chemistry parameters listed in approved system chemistry control procedures.
3. Technical Specifications, TRM, ODCM, or NPDES Permit near miss.
4. Erroneous data or analysis received from an off-site laboratory.
5. Negative trends identified through self-assessment, self check programs, or evolution critiques, after review by Chemistry/Radwaste/Environmental Management that require further monitoring or program enhancements.
6. Failure of equipment used to maintain chemistry within specification (e.g. chemical addition equipment, condensate polisher, etc.) that causes chemistry to be outside of goal/specification.
7. Failure of equipment used for environmental monitoring (e.g., REMP, NPDES, or MET Tower, etc.) that causes environmental monitoring requirements to not be met.

7.12.2. Personnel/Plant Safety Issues:

1. Failures of work group to ensure controlled materials are properly labeled and stored.
2. Failures of chemical storage cabinet owners to perform required inspections, housekeeping integrity and paperwork for Chemical Storage Cabinets.
3. Hazardous waste generation in excess of 1000 kg in one month.
4. Any mixed waste generation at a PWR or mixed waste generation in excess of 150 kg at a BWR.
5. Unplanned generation of large amounts (> 250 Kg) of hazardous waste from a single job.

7.12.3. Radiological Issues

1. Generation of radwaste that was not pre-planned or adequately prepared for.

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2. Spread of radioactive contamination in the laboratory/sample sinks to normally non-contaminated areas.

7.12.4. Security Issues

1. Any Security violation, such as violation of security barriers, resulting from Chemistry/Radwaste/Environmental activities.

**8.0** **RADIATION PROTECTION** (This attachment provides examples only and is not a complete list and shall not limit a person from documenting an IR for any problem):

8.1. Conditions of a repetitive or generic nature associated with hardware nonconformances that continually challenge Radiation Protection personnel in the performance of assigned tasks and are not tracked by a chronic system problem.

8.2. An inadequacy or inaccuracy in training materials that results in a performance based problem in the plant.

8.3. Documentation/data/calculation errors that go undetected following review/approval and could cause minor challenges to radiological protection.

8.4. Identification of plant instrumentation or equipment that cannot meet reliability standards for usage and results in an increase potential for radiological exposure.

8.5. Unplanned instrumentation or major equipment inoperability that compromises Radiation Protection standards or leads to the inability to evaluate radiological conditions or falls below the number required by the UFSAR.

8.6. Radiation Protection programmatic potential trends identified through self-assessment or self check programs.

8.7. Exceeding Micro ALARA planned work in excess of 25%.

**9.0** **SECURITY** (This attachment provides examples only and is not a complete list and shall not limit a person from documenting an IR for any problem):

9.1. Any issue resulting in the generation of a Security Event Report excluding environmental conditions on PIDS and CCTV.

9.2. Discovery of inadequate or inaccurate procedures or guidelines which could produce an unexpected or adverse result.

9.3. Potential trends (equipment condition or personnel performance) identified through self-assessment, scorecards, or other self-critical programs.

9.4. NRC or NOS identified violations, findings, or deficiencies.

9.5. Security training program issues.

9.6. Force on Force drill issues and enhancement opportunities.

9.7. Injury to department personnel.

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**Functional Area Threshold Guidance**
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9.8. Failure of security equipment.

9.9. Vehicle accidents/damage.

9.10. Confirmed positive indications on the Itemizer 3.

9.11. Unusual, suspicious, or abnormal situations or conditions discovered by, or reported to, security personnel.

9.12. Any human performance event not meeting procedural requirements or management expectations.

9.13. Work hour deviations.

**10.0** **<u>OPERATIONS</u>** (This attachment provides examples only and is not a complete list and shall <u>not</u> limit a person from documenting an IR for any problem):

10.1. Procedure/clearance violations, incorrect procedure or revision use, equipment found out of position, and inaccuracy/inadequacy in training materials that result in any one of the following:

10.1.1. A large spill requiring additional assistance to contain/clean-up.

10.1.2. Loss of generating capacity.

10.1.3. Disabling a redundant system/train/component.

10.1.4. Entry into an abnormal operating procedure.

10.1.5. Equipment damage that makes the component or device inoperable.

10.1.6. C & T issues or component mispositionings that could have resulted in:

1. Personnel injury.
2. Equipment inoperability.
3. Equipment damage.

10.1.7. Documentation/data/calculation errors that goes undetected following review/approval.

10.1.8. Unplanned entries into abnormal operating procedures due to equipment failures or hardware non-conformances.

10.1.9. Programmatic concerns that result in, or could have resulted in, a reduction in the effectiveness of an established barrier to personnel or plant safety.

10.1.10. Potential trends identified during review of any Operations Section managed process (e.g., Equipment Tagging).

10.1.11. Preliminary troubleshooting during surveillance testing does not resolve the issue.

10.1.12. Identification of any unexpected impact on operational margin. (e.g., During operation of a pump it was identified that in a specific configuration there was an unexpected low operational margin).

ATTACHMENT 2

Revision to Mark-up of Proposed Technical Specifications Bases Pages

BASES

BACKGROUND

5. Reactor Water Cleanup System Isolation (continued)

initiation switch is considered to provide 1 channel input into each trip system. Each of the two trip systems is connected to one of the two RWCU valves.

RWCU Functions isolate the Group 3 valves.

6. Shutdown Cooling (SDC) System Isolation

The Reactor Vessel Water Level—Low Function receives input from four reactor vessel water level channels. Each channel inputs into one of four trip strings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of the SDC ~~suction~~ isolation valves. Any channel will trip the associated trip string. Only one trip string must trip to trip the associated trip system. The trip strings are arranged in a one-out-of-two taken twice logic to initiate isolation.

The ~~Recirculation Line Water Temperature~~**Reactor Vessel Pressure**-High Function receives input from four ~~temperature~~ **reactor pressure** channels. Each channel inputs into one of ~~the four~~**two** trip ~~strings~~**systems**. Two ~~trip strings~~ **pressure channels** make up a trip system **in a one-out-of-two taken once logic arrangement** and both trip systems must trip to cause an isolation of the SDC ~~suction~~ isolation valves. ~~Any channel will trip the associated trip string. Only one trip string must trip to trip the associated trip system. The trip strings are arranged in a one-out-of-two taken twice logic to initiate isolation function. Each of the two logic systems is connected to one of the two valves on the SDC suction penetration. Only one of the logic systems isolates the SDC return penetration.~~

Shutdown Cooling System Isolation Functions isolate some Group 3 valves (SDC isolation valves).

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The isolation signals generated by the primary containment isolation instrumentation are implicitly assumed in the safety analyses of References 2 and 3 to initiate closure of valves to limit offsite doses. Refer to LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Applicable Safety Analyses Bases for more detail of the safety analyses.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This Function isolates the Group 3 valves.

Shutdown Cooling (SDC) System Isolation

6.a. ~~Recirculation Line Water Temperature~~**Reactor Vessel Pressure**–High

The ~~Recirculation Line Water Temperature~~**Reactor Vessel Pressure**–High Function is provided to isolate the Shutdown Cooling **(SDC)** System. This interlock is provided for equipment protection **only** to prevent exceeding the **SDC** system design temperature, and credit for the interlock is not assumed in the accident or transient analysis in the UFSAR.

The ~~Recirculation Line Water Temperature~~**Reactor Vessel Pressure**-High Isolation Function receives input from four ~~Recirculation Line temperature~~**reactor pressure** channels. Each **pressure** channel inputs into one of ~~four~~ **two** trip ~~strings~~**systems**. Two ~~trip strings~~**pressure channels** make up a trip system **in a one-out-of-two taken once logic arrangement** and both trip systems must trip to cause an isolation of the ~~shutdown cooling (~~SDC~~)~~ ~~suction~~ valves. ~~Any channel will trip the associated trip string. Only one trip string must trip to trip the associated trip system. The trip strings are arranged in a one-out-of-two taken twice logic to initiate isolation. Therefore all four channels~~**Two pressure channels per trip system** are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. The Function is only required to be OPERABLE in MODES 1, 2, and 3, since these are the only MODES in which the reactor coolant temperature exceeds the system design temperature and equipment protection is needed. The **pressure** Allowable Value was chosen to be low enough to protect the system equipment from exceeding its design temperature.

This Function isolates the Group 3 shutdown cooling valves.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.2 and SR 3.3.6.1.5 (continued)

The 92 day Frequency of SR 3.3.6.1.2 is based on the reliability analyses described in References 8 and 9. The 24 month Frequency of SR 3.3.6.1.5 is based on engineering judgement and the reliability of the components.

SR 3.3.6.1.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in 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 that accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analyses of References 9 and 10.

SR 3.3.6.1.4 and SR 3.3.6.1.6

A 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 Function 6.a only, there is a plant-specific program which verifies that the instrument channel functions as required, by verifying the as-left and as-found settings 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 is based on the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.6.1.6 is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

ATTACHMENT 3

Retyped Proposed Technical Specifications Bases Pages

BASES

BACKGROUND

5. Reactor Water Cleanup System Isolation (continued)

initiation switch is considered to provide 1 channel input into each trip system. Each of the two trip systems is connected to one of the two RWCU valves.

RWCU Functions isolate the Group 3 valves.

6. Shutdown Cooling (SDC) System Isolation

The Reactor Vessel Water Level—Low Function receives input from four reactor vessel water level channels. Each channel inputs into one of four trip strings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of the SDC isolation valves. Any channel will trip the associated trip string. Only one trip string must trip to trip the associated trip system. The trip strings are arranged in a one-out-of-two taken twice logic to initiate isolation.

The Reactor Vessel Pressure-High Function receives input from four reactor pressure channels. Each channel inputs into one of two trip systems. Two pressure channels make up a trip system in a one-out-of-two taken once logic arrangement and both trip systems must trip to cause an isolation of the SDC isolation valves.

Shutdown Cooling System Isolation Functions isolate some Group 3 valves (SDC isolation valves).

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The isolation signals generated by the primary containment isolation instrumentation are implicitly assumed in the safety analyses of References 2 and 3 to initiate closure of valves to limit offsite doses. Refer to LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Applicable Safety Analyses Bases for more detail of the safety analyses.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This Function isolates the Group 3 valves.

Shutdown Cooling (SDC) System Isolation

6.a. Reactor Vessel Pressure—High

The Reactor Vessel Pressure—High Function is provided to isolate the Shutdown Cooling (SDC) System. This interlock is provided for equipment protection only to prevent exceeding the SDC system design temperature, and credit for the interlock is not assumed in the accident or transient analysis in the UFSAR.

The Reactor Vessel Pressure-High Isolation Function receives input from four reactor pressure channels. Each pressure channel inputs into one of two trip systems. Two pressure channels make up a trip system in a one-out-of-two taken once logic arrangement and both trip systems must trip to cause an isolation of the SDC valves. Two pressure channels per trip system are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. The Function is only required to be OPERABLE in MODES 1, 2, and 3, since these are the only MODES in which the reactor coolant temperature exceeds the system design temperature and equipment protection is needed. The pressure Allowable Value was chosen to be low enough to protect the system equipment from exceeding its design temperature.

This Function isolates the Group 3 shutdown cooling valves.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.2 and SR 3.3.6.1.5 (continued)

The 92 day Frequency of SR 3.3.6.1.2 is based on the reliability analyses described in References 8 and 9. The 24 month Frequency of SR 3.3.6.1.5 is based on engineering judgement and the reliability of the components.

SR 3.3.6.1.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in 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 that accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analyses of References 9 and 10.

SR 3.3.6.1.4 and SR 3.3.6.1.6

A 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 Function 6.a only, there is a plant-specific program which verifies that the instrument channel functions as required, by verifying the as-left and as-found settings 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 is based on the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.6.1.6 is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)