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PWR Owners Group

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WCAP-15981-NP, Rev 0 (Non-Proprietary)
Project No. 694

August 22, 2007

OG-07-376

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

Subject: **Response to Requests for Clarification of June 28, 2007 RAI Responses for WCAP-15981-NP, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," (LSC-0072 R1/MUHP-3038)**

References:

1. WOG Letter, F. Schiffley to Document Control Desk, "Transmittal of WCAP-15981-NP (Non-Proprietary), Rev. 0, "Post Accident Monitoring Instrumentation Redefinition for Westinghouse NSSS Plants," WOG-02-474, September 17, 2004.
2. NRC E-Mail, G. Shukla (NRC) to S. DiTommaso (W), "RAIs on WCAP-15981 – Post Accident Monitoring Instrumentation Re-Definition," April 11, 2005.
3. NRC E-Mail, G. Shukla (NRC) to S. DiTommaso (W), "I&C RAIs on WCAP-15981, "Post Accident Monitoring Instrumentation Re-definition for Westinghouse NSSS Plants," May 16, 2005.
4. NRC E-Mail, G. Shukla (NRC) to S. DiTommaso (W), "RAIs on WCAP-15981, Post Accident Monitoring Instrumentation Redefinition," May 26, 2005.
5. PWROG Letter, F. Schiffley to Document Control Desk, "Responses to the NRC Request for Additional Information (RAI) Regarding the Review of WCAP-15981-NP, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," WOG-06-104, March 20, 2006.
6. NRC E-Mail, R. Palla (NRC) to R. Lutz (W), "WCAP-15981 (PAM)," May 10, 2006.
7. PWROG Letter, F. Schiffley to Document Control Desk, "Additional Revisions to WCAP-15981-NP, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," OG-06-259, August 10, 2006.
8. NRC E-Mail, S. Peters (NRC) to T. Laubham (W), "WCAP-15981 Final Draft RAIs", March 5, 2007.
9. NRC E-Mail, S. Peters (NRC) to C. DiMuzio (W), "WCAP-15981 RAIs", May 3, 2007.
10. PWROG Letter, F. Schiffley to Document Control Desk, "Response to Additional Requests for Information for WCAP-15981-NP, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," OG-07-292, June 28, 2007

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11. NRC E-Mail, S. Peters (NRC) to C. DiMuzio (W), "WCAP-15981 Conference Call Request", July 11, 2007.
12. NRC E-Mail, S. Peters (NRC) to C. DiMuzio (W), "WCAP-15981 RAI 3 Response Followup", July 30, 2007.
13. NRC E-Mail, S. Peters (NRC) to C. DiMuzio (W), "Revised Clarifications of PWROG June 28, 2007 Response to RAIs", July 31, 2007.
14. NRC E-Mail, S. Peters (NRC) to C. DiMuzio (W), "OG-06-259", July 31, 2007.

The purpose of this letter is to transmit supplemental information regarding RAIs responses transmitted in support of WCAP-15981-NP, "Post Accident Monitoring Instrumentation Redefinition for Westinghouse NSSS Plants,".

In September 2004, the Pressurized Water Reactor Owners Group (PWROG) submitted WCAP-15981-NP (Non-Proprietary), Rev. 0, "Post Accident Monitoring Instrumentation Redefinition for Westinghouse NSSS Plants," for NRC review and approval (Ref. 1). In April 2005 and May 2005, the NRC provided Requests for Additional Information (RAIs) on WCAP-15981 (Ref. 2, 3 and 4). Responses to the RAIs and WCAP revisions, as mark-ups, were transmitted to the NRC on March 20, 2006 (Ref. 5).

Additional RAIs on WCAP-15981 were received in May 2006 (Ref. 6) and responses to the RAIs and WCAP revisions, as mark-ups, were transmitted to the NRC on August 10, 2006 (Ref. 7).

Additional RAIs on WCAP-15981 were received in March and May 2007 (Ref. 8 and 9) and the responses to those the RAIs and WCAP revisions, as mark-ups, were transmitted to the NRC on June 28, 2007 (Ref. 10).

A conference call regarding the responses to RAIs 3 and 4 in Ref. 10 was requested by NRC to discuss an additional RAI received on July 11, 2007 (Ref. 11). A conference call was held on July 30, 2007 and the response to this RAI is provided in Attachment 1 to this letter.

Additional requests for clarifications to the RAI responses provided in Ref. 10 were received in July 2007 (Ref. 12, 13, and 14). Attachment 2 to this letter provides the clarifications requested in Ref. 12. Attachment 3 to this letter provides the clarifications requested in Ref. 13 and 14. Attachment 4 to this letter provides a revision, as a mark-up, to WCAP-15981 that addresses the clarification requested in Ref. 14.

Following receipt of the Safety Evaluation for WCAP-15981, the WCAP revisions (as mark-ups contained in the attachments to letters WOG-06-104, OG-06-259, OG-07-292, and this letter), as well as these RAI response letters, will be incorporated into the approved version and will be issued as WCAP-15981-NP-A, Revision 1.

If you have any questions concerning this matter, please contact Christine DiMuzio at 412-374-5680.

Sincerely yours,

Christine A DiMuzio for

Frederick P. "Ted" Schiffley, II, Chairman
Pressurized Water Reactor Owners Group

FPS:CD:mjl

Attachments

cc: Licensing Subcommittee
Steering Committee
S. Peters, NRC (via FedEx)
T. Mensah, NRC (via FedEx)
S. Rosenberg, NRC (via FedEx)
J. D. Andrachek
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C. B. Brinkman
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Attachment 1

NRC E-Mail, S. Peters (NRC) to C. DiMuzio (W), "WCAP-15981 Conference Call Request", July 11, 2007 (Reference 11 in Cover Letter)

RAI:

In its response, the PWROG referenced the information in the Westinghouse Emergency Response Guideline (ERG) critical safety function (CSF) status trees to claim that the reactor vessel water instrumentation system (RVLIS), RCS cold-leg and hot-leg temperature indicators can be relocated out of the TS.

In the CSF Status Trees (F-0.1 though F.0.6) of Westinghouse ERG-HP, 16 variables are used for the operator to determine the priority of functional recovery guidelines to enter for the CSF recovery. In considering the use of the variables in the CSF status trees, discuss why all the variables in the CSF status trees are not categorized as REG 1.97 category 1 variables and included in the TS. The variables include:

A. Reactivity (F-0.1)

1. power range neutron flux
2. intermediate neutron flux
3. source range neutron flux

B. Core Cooling (F-0.2)

1. CET
2. subcooling monitor
3. RVLIS

C. Heat Sink (F-0.3)

1. SGWL (narrow range)
2. total feedwater flow
3. SG pressure

D. Integrity (F-0.4)

1. RCS cold-leg temperature
2. RCS pressure
3. RCS hot-leg temperature

E. Containment (F-0.5)

1. containment pressure
2. containment sump level
3. containment radiation

F. Inventory (F-0.6)

1. presssurizer water level
2. RVLIS

Note that the variables to be moved out of the WCAP-1431 are the RCS hot-leg temperature, RCS cold-leg temperature, source range neutron flux, containment sump level and RVLIS. The variables not included in the WCAP-1431 are the intermediate range neutron flux, SGWL (narrow range), and total feedwater flow.

RAI RESPONSE:

As discussed in the response to RAI 2 of the General Comments from the Plant Systems Branch RAIs

(Reference 1), the Emergency Operating Procedures (EOPs) were used in the methodology contained in WCAP-15981 to determine the instrumentation that is utilized to perform specific manual actions assumed in the DBA analyses for which there is no automatic actuation of equipment provided. The instrumentation utilized in the Emergency Operating Procedures (EOPs) to cue specific manual actions that are assumed in the Design Basis Analysis (DBA) analyses was determined to satisfy Criterion 3 of 10 CFR 50.36, and should be included in the Post Accident Monitoring (PAM) Technical Specification.

WCAP-15981 also contains the results of a generic evaluation of plant instrumentation from a risk importance perspective, based on risk insights obtained from a survey of Probabilistic Risk Assessments (PRAs) from Westinghouse NSSS PWRs. The PRA includes EOP operator actions for all accident sequences, including those that are beyond the design basis of the plant. Also included under the broad category of risk insights is the use of instrumentation in the Severe Accident Management Guidance (SAMG), and Emergency Plan Implementing Procedures (EPIP). Instrumentation that was determined to have a high risk importance from the PRA (which includes EOPs), SAMG, or EPIP perspectives was concluded to satisfy Criterion 4 of 10 CFR 50.36 and should be included in the PAM Technical Specification.

Implementation of the proposed generic changes to the PAM Technical Specification contained in NUREG-1431 will be accomplished by performing a plant-specific assessment based on the methodology presented in WCAP-15981. Therefore, DBA analyses and EOPs would be used for the purpose of identifying instrumentation that satisfies Criterion 3 of 10 CFR 50.36 and risk insights (PRA, SAMG and EPIP) would be used solely for the purpose of identifying the instrumentation that satisfies Criterion 4 of 10 CFR 50.36.

Further, as discussed in the response to RAI 4 of the General Comments from the Plant Systems Branch RAIs contained in Reference 1, there are two types of instrumentation utilized in the EOPs, PRA, SAMG, and EPIP: a) "key" instrumentation that is necessary for the operator to effectively diagnose and mitigate accidents, and b) "backup" instrumentation that supplements the "key" instrumentation that supports operator actions to recover the plant.

The "key" instrumentation provides the primary information required to permit the control room operating staff to:

- Perform the diagnosis, in accordance with the plant EOPs, of plant conditions required to initiate manual actions assumed in DBAs that are required to bring the plant to a safe stable state (as discussed in the UFSAR),
- Perform the pre-planned manual actions in accordance with the plant EOPs*, for which no automatic control is provided, that are required for safety systems to accomplish their safety function to mitigate DBAs,
- Perform the pre-planned manual actions in accordance with the plant EOPs* to bring the plant to a safe stable state for a wide range of beyond design basis accidents included in the PRA, and
- Diagnose plant conditions that may pose a threat to the health and safety of the general public in accordance with the plant SAMG and EPIP.

The "backup" instrumentation permits the control room operating staff to:

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* Not all of the actions in the EOPs for which no automatic control is provided or which bring the plant to a safe stable condition satisfy the criteria contained in 10 CFR 50.36, which are used to determine whether the indication should be included in the Technical Specifications. The indications associated with the operator actions that should be included in the Technical Specifications need to be determined using a process such as the methodology described in WCAP-15981. The methodology described in WCAP-15981 evaluated the operator actions identified in the CSF Status Trees.

- Verify the indications of the key instrumentation,
- Operate plant systems utilized to achieve a safe shutdown, including the verification of the automatic actuation of safety systems, and
- Operate other systems normally utilized for achieving a safe shutdown condition.

Regulatory Guide 1.97 provides guidance for determining the Category, i.e., 1, 2 or 3 depending on the importance to safety of the measurement of a specific variable. The plant variables being measured fall into either key or backup instrumentation as discussed above. Specifically, Category 1 is intended for key variables, while Category 2 applies to instrumentation designated for indicating system operating status, and Category 3 applies to backup and diagnostic instrumentation.

The EOPs prioritize the instrumentation to be used to diagnose and take actions to mitigate the consequences of accidents. The operators are trained to use the EOPs in a "verbatim compliance" mode. The SAMG and EPIPs also prioritize instrumentation for diagnosis and actions based on several considerations, including the availability of the instrumentation following an accident. Therefore, the EOPs, SAMG and EPIPs can be used to define the key and backup instrumentation from the perspective of both Criteria 3 and 4 of 10 CFR 50.36.

In summary, the instrumentation utilized in the EOPs to cue operator actions for DBA accidents for which no automatic control is provided is considered to be Regulatory Guide 1.97 Type A instrumentation and satisfies Criterion 3 of 10 CFR 50.36. The non-Type A instrumentation was further evaluated from a risk perspective based on its use in the PRA (including the EOPs), SAMG, and EPIP to determine whether it was risk significant. Instrumentation that was determined to be risk significant satisfies Criterion 4 of 10 CFR 50.36, and is therefore a key variable consistent with the definition of Category 1 as discussed in Regulatory Guide 1.97. This assessment of determining whether non-Type A instrumentation is risk significant is consistent with the NRC comment on the relocation of Category 1 variables from the Standard Technical Specifications discussed in Reference 2. The specific comment was: "The Owners Groups' should develop further risk-based justification in support of relocating any or all Category 1 variables from the Standard Technical Specifications." Additionally, if the instrumentation does not satisfy Criterion 3 or 4 of 10 CFR 50.36, it can be relocated out of the PAM Technical Specification to a licensee controlled document.

However, it needs to be recognized that not all of the indications identified in the CSF Status Trees in the EOPs satisfy the criteria contained in 10 CFR 50.36, which are used to determine whether the indication should be included in the Technical Specifications. As documented in WCAP-15981, the Power Range Neutron Flux, Core Exit Temperature, Steam Generator Pressure, Steam Generator Wide Range Water Level, RCS Wide Range Pressure, Containment Area High Range Radiation, and Pressurizer Water Level indications have been determined to satisfy Criterion 3 or Criterion 4, or both, of 10 CFR 50.36, and therefore would be included in the PAM Technical Specification. Therefore the key indication(s) on a red path for each of the ERG CSF Status Trees will be included in the Technical Specifications.

The source range neutron flux (as discussed on page 30 of WCAP-15981), subcooling monitor (as discussed on page 38 of WCAP-15981), RVLIS (as discussed on page 31 of WCAP-15981), SG Narrow Range Water Level (as discussed on page 39 of WCAP-15981), total feedwater flow (as discussed on page 36 of WCAP-15981), RCS cold-leg temperature (as discussed on pages 30 and 31 of WCAP-15981), RCS hot-leg temperature (as discussed on page 30 of WCAP-15981), containment pressure (as discussed on page 33 of WCAP-15981), and containment sump level (as discussed on pages 32 and 37 of WCAP-15981) indications have not been identified as key indications, and therefore do not satisfy either Criterion 3 or Criterion 4 of 10 CFR 50.36, and should not be included in the Technical Specifications. These variables are considered to be backup variables that are used to verify the indications of the key instrumentation in the CSF Status Trees. The intermediate neutron flux indication is not specifically addressed in the WCAP; however the discussion of the source range neutron flux (page 30 of WCAP-15981) is also applicable to the intermediate range neutron flux instrumentation. As stated above, the inclusion of a variable within an ERG CSF Status Tree is not a criterion for including that variable in the Technical Specifications. An ERG CSF Status Tree variable must satisfy Criterion 3 and/or Criterion 4 to be included in the Technical Specifications.

References:

1) Westinghouse Owners Group Letter, WOG-06-104, "Response to the NRC Request for Additional Information (RAI) Regarding the Review of WCAP-15981-NP, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," (LSC-0072 R1/MUHP-3038) dated March 20, 2006.
2) NRC letter, from Thomas E. Murley (NRC) to R. A. Newton (WOG), dated May 9, 1988.

Attachment 2

NRC E-Mail, S. Peters (NRC) to C. DiMuzio (W), "WCAP-15981 RAI 3 Response Followup", July 30, 2007 (Reference 12 in Cover Letter)

RAI:

The RAI 3 response indicated that reactor vessel instrumentation system (RVLIS) will provide the most useful information for small break loss-of-coolant-accidents (SBLOCAs). However, It claimed that other indications (e.g., CETs and subcooling monitors based on CETs) can be used by operators as an alternate method for mitigation of the consequences of an SBLOCA. Thereby, the RVLIS was classified as non-RG 1.97 category-1 instrumentation and proposed to relocated out of the Standard Technical Specifications (STS).

The NRC staff found that the RAI 3 response is not consistent with the Generic Letter (GL) 82-28 guidance for implementing the inadequate core cooling (ICC) instrumentation system. GL 82-28 documented the basis that required the RVLIS, as well as CETs and subcooling monitors, as an instrumentation system for detection of ICC. It indicated that the RVLIS is required during an SBLOCA before the core has boiled dry (indicated by CETs) for the operator to identify void formation in the reactor vessel head or track the inventory of a coolant in the vessel and primary system. The GL indicated that the subcooling monitor gives early indication for an ICC but does not indicate whether the void and coolant in the reactor vessel is getting better or worse. Besides, the RVLIS can also be used by an operator to avoid an ICC when voids in the reactor coolant system and saturation condition result from overcooling events, such as steam line break (SLB), and steam generator tube rupture (SGTR).

Please clarify the differences between the RAI response and GL 82-28 guidance in terms of use of the RVLIS in the Westinghouse Emergency Response Guidelines for void identification and elimination during an SBLOCA, SLB and SGTR. The response should: (1) show that CETs and subcooling monitor are acceptable alternate instruments for tracking the trend of the reactor water level for the conditions indicated in the GL 82-28 discussed above; and (2) demonstrate that the RVLIS is not a RG 1.97 Category 1 instrumentation, and thus can be removed out of the STS.

RAI RESPONSE:

The inadequate core cooling instrumentation system discussed in GL-82-28 requires that the plant design and Emergency Operating Procedures include core exit temperature, reactor coolant subcooling margin and reactor vessel level instrumentation to provide the greatest assurance of early detection of an approach to an inadequate core cooling condition. The RAI response does not propose that any of these indications be eliminated from the plant design and / or Emergency Operating Procedures. The WCAP-15981 methodology, as supported by this RAI response, only proposes that the reactor vessel level and reactor coolant subcooling margin monitor indications be relocated out of the PAM Technical Specification to a licensee controlled document, based on the application of the four criteria contained in 10 CFR 50.36(c)(2)(ii). The methodology proposed in WCAP-15981 shows that the reactor vessel level indication and the reactor coolant subcooling margin monitor indication (where applicable in individual plant TS) do not satisfy any of the four criteria of 10 CFR 50.36(c)(2)(ii), and therefore can be relocated out of the Technical Specifications to a licensee controlled document. The information contained in the WCAP and in the RAI response concludes that the reactor vessel level and reactor coolant subcooling margin monitor indications are important backup indications to the core exit temperature indication, and would be included with the other post accident monitoring instrumentation that is proposed by the WCAP to be relocated from the Technical Specifications to a licensee controlled document. Therefore, the information provided in WCAP-15981 and the RAI response does not impact the inadequate core cooling instrumentation system required by GL-82-28.

Attachment 3

NRC E-Mail, S. Peters (NRC) to C. DiMuzio (W), "Revised Clarifications of PWROG June 28, 2007 Response to RAIs", July 31, 2007 (Reference 13 in Cover Letter)

RAI:

On June 28, 2007 the Pressurized Water Reactor Owners Group (PWROG) provided responses to the staff's Request of Additional Information (RAI) dated March 5, 2007, and May 3, 2007, concerning WCAP-15981-NP, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants." Unless otherwise indicated, the following are questions to clarify the responses to the May 3, 2007 RAIs:

1. In response to RAI 1 Part A, Refueling Water Storage Tank (RWST) Level instrumentation is listed as a key variable for the Core Cooling function. This would make RWST Level a Type B Category 1 variable. Provide a discussion of how RWST Level provides key information for the Core Cooling function.

   Response:

   Appendix A to WCAP-15981 shows that RWST level indication is important to risk because the operator action to refill the RWST for the steam generator tube rupture and LOCA outside containment sequences have RAW / RRW values in excess of the values typically used for risk importance (e.g., RG 1.174). The instrumentation used in the ERGs to determine the need to refill the RWST is the RWST level indication. Therefore, the RWST level indication is a key indication that satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii), and is proposed to be included in PAM Technical Specification. In addition, for those plants that require manual operator action to complete the ECCS and containment spray switchover from injection to sump recirculation, RWST level indication is a Type A variable and, therefore, required to be in the TS per Criterion 3.

2. In response to RAI 1 Part A, Core Exit Temperature (CET) is listed as a key variable for the Maintaining Reactor Coolant System Integrity function, the Reactor Coolant Pressure Boundary function and Primary Coolant System Status. Provide a discussion of how CET provides key information for the Maintaining Reactor Coolant System Integrity function, Reactor Coolant Pressure Boundary function and Primary Coolant System Status.

   Response:

   Appendix A to WCAP-15981 shows that the Core Exit Temperature indication provides the direct indication of satisfying the core cooling function and is therefore a key variable for the Core Cooling Function. The Core Exit Temperature indication is also used in conjunction with the Reactor Coolant Pressure indication to identify when pressurized thermal shock conditions are being approached, and thus would be an indicator for Maintaining Reactor Coolant System Integrity following an accident. The Core Exit Temperature indication is also used in conjunction with the Reactor Coolant Pressure indication to determine RCS subcooling, which is an indicator of the Reactor Coolant Pressure Boundary Function and the Primary Coolant System Status. The ability to maintain RCS subcooling is an indication that there are no large breaches of the reactor coolant pressure boundary and that the primary system function of removing decay heat to the steam generators is available.

3. In response to RAI 1 Part A, RCS Pressure is not listed as a key variable for the Containment function. However, RG 1.97 lists RCS Pressure as a key variable for the detection of potential breach and accomplishment of mitigation of the Containment function and thus a Type C Category 1 variable. Provide a discussion of RCS Pressure and its relationship to the Containment function. Under

WCAP-15981 would RCS Pressure remain a Type C Category 1 variable for the Containment function?

Response:

The Reactor Coolant Pressure indication is not considered to be an indicator of a potential breach and accomplishment of mitigation of the Containment function. The double ended rupture of a reactor coolant pipe during full power operation (normal operating reactor coolant system pressures) would not result in a containment pressure that exceeds the containment design pressure. A higher RCS pressure, even if it were not mitigated by the pressurizer safety valves, that resulted in a rupture of a reactor coolant pipe would not result in a containment pressure that would threaten the containment integrity. Risk studies have shown that the typical containment has an ultimate pressure capability whose median value is two to two and a half times the design pressure. Therefore, RCS pressures above the normal operating pressure are not a threat to containment integrity. Thus, the Reactor Coolant Pressure indication is not considered to be an indicator of a potential breach of containment integrity.

4. The response to RAI 1 Part C.a concerning Neutron Flux (Source Range) did not include a discussion of the instrumentation to be used to provide an early indication of a return to criticality. Provide a discussion of the instrumentation to be used to provide an early indication of a return to criticality and how that instrumentation satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

Response:

As discussed in the response to RAI No. 2 in OG-07-292, dated June 28, 2007, the primary indication of shutdown margin and therefore the potential approach to recriticality is provided by the measurement of the reactor coolant system boron concentration by sampling. The ERGs clearly indicate that adequate shutdown margin, as determined by RCS sampling, be completed prior to RCS cooldown and depressurization to conditions that might result in an approach to recriticality. The source range flux measurement provides a backup to the RCS boron concentration measurement to ensure that adequate shutdown margin is maintained.

5. The responses to RAI 1 Part C.b concerning RCS Hot Leg Water Temperature and RAI 1 Part C.c concerning RCS Cold Leg Water Temperature does not include a discussion of the instrumentation to be used to provide information for determining temperature differences between RCS loops. Provide a discussion of the instrumentation to be used to provide information about temperature difference between RCS loops and how that instrumentation satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

Response:

RCS hot leg and cold leg temperatures can provide information regarding natural circulation cooling without the reactor coolant pumps operating to indicate the Primary Coolant System Status function following an accident that does not result in a breach of the reactor coolant system. However, the primary indications of the effectiveness of natural circulation cooling without the reactor coolant pumps operating are the auxiliary feedwater flow rate and steam generator pressure indications. Differences in the RCS loop temperatures are only a concern from the perspective of stagnant loops that can result in the potential for a boron dilution event if a reactor coolant pump is restarted. However, the primary indications of a stagnant loop condition are the auxiliary feedwater flow rate and steam generator pressure indications. The auxiliary feedwater flow rate and steam generator pressure indications are already identified as key variables for the Secondary System Status and AFW System Status functions in response to RAI 1, Part A in OG-07-292, dated June 28, 2007. {Note behind change: per OG-07-292 only WR RCS pressure, PZR level, and the CETs are identified for the Primary Coolant System Status function.}

Finally, risk studies do not identify that differences in the RCS loop temperatures are risk significant and therefore the RCS loop temperature indications do not satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii).

6. The response to RAI 1 Part C.e concerning Reactor Vessel Water Level did not discuss accident conditions that include determining whether the reactor core is uncovered. Provide a discussion of why the potential for uncovering the reactor core and the level of core exposure should not be considered a Category 1 variable and how that instrumentation satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

   In response to the March 5, 2007 RAI 3, the PWROG stated, "The executive volume of the ERGs concludes that RVLIS will provide the most useful information for breaks in the RCS ranging from small leaks to breaks in the limiting small break range." This information appears to contradict the conclusion of the March 5, 2007 RAI 3, that RVLIS does not satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii). Provide a clarification of this apparent conflict and discuss how Reactor Vessel Water Level or Coolant Inventory instrumentation satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

   Response:

   The intent of the discussion in the ERG executive volume concerning the use of RVLIS is that the RVLIS indication is most useful for small breaks, as compared to its use for larger breaks. The information provided in Section 5.1 of WCAP-15981, which was based on risk studies such as those summarized in Appendix A of WCAP-15981, concludes that the Core Exit Temperature indication is a more reliable indication of core uncovery under a wide range of conditions and provides a timely indication of an inadequate core cooling condition. Further, the risk studies show that small amounts of core uncovery that do not lead to significant core exit temperature increases are not a threat to fuel cladding integrity. Thus, RVLIS indication is not a key indicator of inadequate core cooling and does not satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii).

7. In response to RAI 1 Part C.f, Containment Sump Water Level (Wide Range) is recommended for reclassification from Type B Category 1 to Type B Category 3 for the Maintaining Reactor Coolant System Integrity function and from Type C Category 1 to Type C Category 3 for the Reactor Coolant Pressure Boundary function. Provide a discussion of why the containment flooding and the amount of water in the containment should not be considered a Category 1 variable and a discussion of how instrumentation to monitor containment water level satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

   Response:

   As discussed in the response to RAI 1 Part C.f in OG-07-292, dated June 28, 2007, and in Section 5.1 of WCAP-15981, the plant operators use the ERGs to respond to accident conditions. The Containment Sump Wide Range level indication is not used by the operators as a key indicator for the Reactor Coolant System Integrity or Reactor Coolant Pressure Boundary functions. The only location in the ERGs in which the Containment Sump Wide Range level indication is used by the operator is for entry to FR-Z.2, Response to Containment Flooding. Since the setpoint for entry into this procedure is the design basis containment flooding level, entry into this procedure would only occur if there is a significant amount of water entering containment following an accident from a non-safeguards source. An example of such an event would be a coincident large LOCA inside containment and a rupture of a service water pipe supplying cooling water to a containment fan cooler. The risk studies show that these types of events are not likely to occur and are therefore not risk significant. Therefore, the Containment Sump Wide Range water level indication is not a key variable for Reactor Coolant System Integrity or Reactor Coolant Pressure Boundary functions and does not satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii).

8. In response to the multiple March 5, 2007 RAIs statements were made that indicated that the CETs along with RCS subcooling could be used in lieu of several current Category 1 variables. RCS subcooling monitor is currently classified in RG 1.97 as a Type B Category 2 variable. However, WCAP-15981 has recommended that the RCS subcooling monitor be reclassified as a Type B Category 3 variable. The arguments presented in the March 5, 2007 RAIs do not support a decrease in the category of the RCS subcooling monitor, but rather an increase to Category 1. Provide a discussion on the use of the RCS subcooling monitor and clarify this apparent conflict.

   Response:

   The RCS Subcooling control room indication typically uses the RCS Wide Range Pressure and the Core Exit Temperature indications as "inputs" to the determination of the RCS subcooling. As discussed in Section 5.1 of WCAP-15981, RCS subcooling is an important parameter in the ERGs for operator actions for termination of safety injection for secondary side breaks (e.g., feed line and steam line breaks) and other non-LOCA events that result in the automatic start of safety injection. Failure to terminate safety injection for these events could lead to the relief of water from the pressurizer relief or safety valves as the pressurizer is filled. The ERGs specify that the operators determine the RCS subcooling without specifying the indication to be used for that purpose. The operators would typically use the RCS Subcooling Monitor indication for expediency and are trained to verify the subcooling indication using the RCS Wide Range Pressure and Core Exit Temperature indications. However, if the RCS Subcooling Monitor is not available, the subcooling margin would be determined from the RCS Wide Range Pressure and Core Exit Temperature indications. Since both the RCS Wide Range Pressure and Core Exit Temperature indications have already been determined to be Category 1 indications based on other considerations, it is not necessary that the computed RCS subcooling indication also be a Category 1 indication. Therefore, it was determined that the RCS Subcooling indication does not satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii).

9. In response to RAI 1 Part A, Pressurizer Level is listed as a key variable for the Reactor Coolant Pressure Boundary function. Provide a discussion of how Pressurizer Level provides key information for the Reactor Coolant Pressure Boundary function.

   Response:

   As discussed in Section 5.1 of WCAP-15981, the Pressurizer level indication is an important parameter in the ERGs for operator actions to terminate safety injection for secondary side breaks (e.g., feed line and steam line breaks) and other non-LOCA events that result in the automatic start of safety injection. Failure to terminate safety injection for these events could lead to the relief of water from the pressurizer relief or safety valves as the pressurizer is filled. The ERGs specify that the operators determine that the pressurizer water level is at least at a low level prior to starting to terminate safety injection and remains on-scale following termination of safety injection for these events. Thus, Pressurizer level indication is an important parameter for maintaining Reactor Coolant Pressure Boundary function.

   In addition, termination of safety injection for secondary side breaks and for steam generator tube rupture was found to be a risk important operator action. This action is cued, in part, by Pressurizer Level indication. Therefore, it was determined that the Pressurizer Level indication satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

10. In response to RAI 1 Part A, Steam Generator Level (Wide Range) is listed as a key variable for the Reactor Coolant Pressure Boundary function. Provide a discussion of how Steam Generator Level (Wide Range) provides key information for the Reactor Coolant Pressure Boundary function.

    Response:

The Steam Generator Level (Wide Range) indication provides the operators with information on the availability of a secondary side heat sink for core decay heat removal. Without a secondary side heat sink for core decay heat removal in the early stages of an accident, the reactor coolant fluid would heat up and RCS pressure would increase to the pressurizer power operated relief / safety valve opening setpoint. Opening of the pressurizer power operated relief / safety valves represents a breach of the Reactor Coolant Pressure Boundary function. Therefore, the Steam Generator Level (Wide Range) provides key information for maintaining the Reactor Coolant Pressure Boundary function. Note that this is only applicable to those portions of an accident sequence when the RCS pressure and temperature are above the cut-in point for shutdown cooling using the residual heat removal system.

11. In response to RAI 1 Part A, RCS Pressure is listed as a key variable for Primary Coolant System status. Provide a discussion of how RCS Pressure provides key information for the Primary Coolant System status.

Response:

As discussed in the response to Clarification No. 8 above, the Reactor Coolant Pressure indication is used in conjunction with the Core Exit Temperature indication to determine RCS subcooling, which is an indicator of the Reactor Coolant Pressure Boundary Function and the Primary Coolant System Status. The ability to maintain RCS subcooling is an indication that there are no large breaches of the reactor coolant pressure boundary and that the primary system function of removing decay heat to the steam generators is available.

NRC E-Mail, S. Peters (NRC) to C. DiMuzio (W), "OG-06-259", July 31, 2007 (Reference 14 in Cover Letter)

RAI:

To be consistent with the Figure 2, the following item should be added to "Details" of Step 4 of Table 14 (page 56b1), "Process to Determine Instrumentation to be Included in the PAMs Technical Specification": Important operator actions based on a review of EOPs.

RAI RESPONSE:

Step 4 of Table 14 in WCAP-15981 (added per the WCAP revisions contained in Ref. 7) will be revised to include a new bulleted item, "Important operator actions based on a review of EOPs".

(The revision to Table 14 is provided in Attachment 4)

Attachment 4

Markup of WCAP-15981 Table 14

Table 14 Process to Determine Instrumentation to be Included in the PAM Technical Specification

Step	Description	Details
1	Identification of operator actions in the design basis accident analyses	• Operator actions based on a review of the design basis accident analyses o Operator actions for which no automatic actuation of equipment is provided
2	PRA technical adequacy	• Summary of PRA o Scope (Level 1, LERF, external events) o Peer reviews o Update history o PRA updating process • PRA reflects as-built, as-operated design o Recent plant modifications and operational changes not reflected in the PRA do not impact the plant-specific PAM implementation • PRA accident sequence and human reliability assessment is technically adequate for evaluating the risk associated with the PAM implementation • Peer review comments resolved or do not impact plant-specific PAM implementation (limited to accident sequence and human reliability elements)
3	Identification of important operator actions identified in the risk assessments	• Operator action Risk Achievement Worth (RAW) and Fussell-Vesely (FV) importance values for CDF and LERF from the quantitative internal events PRA • Important operator actions based on review of results from the external event assessments (qualitative or quantitative PRA)
4	Identification of important operator actions identified in the accident management	• Important operator actions based on a review of results from the E-Plan, including the EALs, CDA and ODCM • Important operator actions based on a review of the EOPs • Important operator actions based on a review of the SAMG
5	Identification of variables and associated instrumentation for the important operator actions identified in Steps 1, 3 and 4	• Identification of important operator actions to the variables and associated instrumentation that cue or verify the operator action • Identify minimum set of instrumentation to support important operator actions
6	Identification of instrumentation to be included or relocated from the PAM Technical Specification	• Focused evaluation of the adequacy of the PRA and HRA treatment of operator actions in the PRA associated with any variables (instrumentation) to be relocated from the PAM Technical Specification • Verify that any instrumentation proposed for relocation from the PAM Technical Specification does not cue an operator action important to risk for external initiating events • Identify appropriate changes to the Regulatory Guide 1.97 classifications to be consistent with the inclusion in, or relocation from, the PAM Technical Specification