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Subject: Pressurized Water Reactor Owners Group

Response to Additional Requests for Information 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," WOG-06-259, August 10, 2006.

8. NRC E-Mail, S. Peters (NRC) to T. Laubham (W), "WCAP-15981 Final Draft RAIs" March 5, 2007.

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9. NRC E-Mail, S. Peters (NRC) to C. DiMuzio (W), "WCAP-15981 RAIs ", May 3, 2007.

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).

Attachment 1 to this letter provides the responses to the RAIs in Ref. 8 and 9. Attachment 2 to this letter provides revisions, as mark-ups, to WCAP-15981 that address these RAIs, where changes to the WCAP were required to address the RAIs.

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, and this letter 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 feel free to call 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)
J. D. Andrachek
R. J. Lutz
C. B. Brinkman
J. A. Gresham
PMO

Attachment 1

Response to Request for Additional Information Regarding WCAP-15981, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants"

Request for Additional Information (RAIs)
[Received March 5, 2007 via an e-mail from S. Peters (NRC) to T. Laubham (W)]

1. Page 14 of WCAP-15981 indicated that all the design basis accidents other than a loss of coolant accident, steam line break and steam generator tube rupture events do not explicitly rely on operator actions. This statement may be not accurate.

   It should be noted that the design basis analysis (DBA) of an inadvertent safety injection (SI) actuation required operator action to terminate SI flow. This should be a DBA for pressurizer indication. Also, the DBA of a loss of normal feedwater for one Westinghouse plant indicated that auxiliary feedwater (AFW) was delivered automatically by two AFW pumps to two steam generators (SGs). Operator action was required to start a third AFW pump from two AFW pumps, delivering flow to third SG. The AFW indication would provide information to the operator to determine the operable AFW pump from the remaining two AFW pumps.

   In light of the observation described above, clarify the statement on page 14 referenced above and address its effect on the results of the post-accident monitoring (PAM) instrumentation redefinition discussed in WCAP-15981.

   Response:

   The operator action to terminate SI flow using the pressurizer level indication and RCS subcooling indication is currently identified as a DBA action in Table 8 of WCAP-15981.

   However, the Inadvertent Operation of the Emergency Core Cooling System During Power Operation design basis accident (DBA) analysis and the operator action to terminate safety injection (SI) flow for this event are not discussed in Section 4.1 of the WCAP. Section 4.1 of the WCAP will be revised to discuss the operator action to terminate safety injection (SI) flow that is assumed in the analysis of this event.

   It should be noted that the DBA analyses for some plants allow temporary pressurizer PORV or safety valve water relief by demonstrating that a more serious plant condition will not result following a spurious SI signal. For those plants, the operator action to terminate SI is not assumed for this event; however other actions may be credited (e.g., terminating normal charging flow and ensuring the PORVs are in automatic and the block valves are open).

   Section 4.1 of WCAP-15981 under the sub-heading of "Other Design Basis Accidents" will be revised to identify that termination of SI flow using the pressurizer level indication and RCS subcooling indication is an explicit operator action for the Inadvertent Operation of the Emergency Core Cooling System During Power Operation DBA for those plants that analyze the event to prevent pressurizer overfill.

   (The revision to Section 4.1 is provided in Attachment 2 of this letter.)

   The operator action to start a third auxiliary feedwater (AFW) pump for the loss of normal feedwater event is not typical for Westinghouse NSSS plants and is therefore, not discussed in Section 4.1 of the WCAP. The implementation guidance provided in Section 8 requires a plant-specific evaluation of the DBAs to identify the instrumentation used to cue operator actions to mitigate the accident. Therefore, plant-specific instrumentation used to cue operator actions in DBAs would be identified during the plant-specific implementation of the methodology described in this WCAP. Also note that the operator actions to maintain steam generator (SG) heat sink and prevent SG overfill using SG level as an indication are listed as a DBA action in Table 8 of the WCAP.

2. Section 5 of WCAP-15981 discussed redefinition of the PAM instruments in Table 9. The discussion did not include instruments such as AFW valve position, boric acid tank level, containment enclosure negative pressure, residual heat removal flow and spray additive tank level.

   Expand the Section 5 discussion to include the Table 9 instruments that were not already discussed for PAM instrumentation redefinition.

   Response:

   Section 5 of WCAP-15981 was expanded to include a discussion of the instruments identified above that were added to the revised Table 9 contained in Attachment 2 to WOG-06-104 (AFW valve position, boric acid tank level, containment enclosure negative pressure, residual heat removal flow and spray additive tank level).

   (The revisions to Section 5 are provided in Attachment 2 of this letter.)

3. WCAP-15981 redefined the PAM instrumentation and proposed to include in the Standard TS only the PAM instruments monitoring Category 1 variables, which were defined in RG 1.97 as key variables that most directly provide information on the accomplishment of safety functions.

   In satisfying the requirements of TMI Action Plan Item II.F.2, "Instrumentation for Detection of Inadequate Core Cooling," existing Westinghouse plants rely on reactor vessel water level (RVWL) system, as well as core exit thermocouples (CETs) and subcooling margin monitoring capacity to provide the operator with ability to monitor the coolant conditions and to appropriately take actions to assure core cooling during the approach to, and to recover from, the inadequate core cooling conditions.

   Page 31 of WCAP-15981 indicated that the RVWL indication was a backup to the CETs for identifying an inadequate core cooling (ICC) condition and was a Category 3 variable. Therefore, the Westinghouse Owners Group proposed to relocate the RVWL system from the Standard TS (NUREG-1431) to a license control document.

   It should be noted that the RVWL system together with CETs and subcooling monitors were designed to provide direct and reliable indications to the operator for the ICC identification and mitigation. Discuss why: (1) both the RVWL system and CET were not used together as Category 1 instruments and included in the Standard TS; and (2) the RVWL system was a backup to the CETs for detecting an ICC condition, instead of the CETs being a backup to the RVWL system. The discussion should include plant operating data from past many reactor years experience and emergency operating procedures considerations to support the preferred Instruments (RVWL system and CETs, RVWL system, or CETs) that would provide most direct, reliable and unambiguous indications for detecting an ICC condition as required by TMI Action Plan Item II.F.2.

   Response:

   Emergency Response Guideline (ERG) Critical Function Status Tree F-0.2 "Core Cooling" (Revision 2, April 2006) uses the core-exit thermocouples (CETs) as the primary measurement for diagnosing inadequate core cooling. The status tree first looks at the CET indications and if they are greater than 1200°F, then inadequate core cooling is diagnosed directly. If the CET indications are less than 1200°F, then RCS subcooling based on the CETs is used. If no subcooling exists, then the CETs are again used at the 700°F temperature level, in conjunction with Reactor Vessel Level Instrumentation System (RVLIS) indication, and the status of the reactor coolant pumps (RCPs), to determine the priority given to an inadequate core cooling response. An alternate method for diagnosing inadequate core cooling is also provided in the ERG Background Document for F-0.2 that also only uses the CETs.

Furthermore, the executive volume of the ERGs contains a discussion of issues related to the use of RVLIS for detecting inadequate core cooling conditions. Several instances are identified when the RVLIS may give an ambiguous indication. These include:

1) a break in the upper head,

2) periods of reactor vessel upper plenum injection,

3) periods of accumulator injection into a highly voided downcomer,

4) periods when the reactor vessel upper head behaves like a pressurizer, and

5) periods of void redistribution in the RCS.

Several additional instances are identified which may result in biased RVLIS indications. These include:

1) reverse flows in the reactor vessel, and

2) core blockage.

The ERGs further discuss that for larger RCS pipe breaks, the response of the RVLIS may be erratic, due to rapid pressure changes in the vessel, in the early portion of the blowdown. In this case, the RVLIS reading would only be useful for monitoring accident recovery when other corroborative indications (e.g., CETs and subcooling based on CETs) can also be observed. 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. For breaks in this range, the system conditions will change at a slow enough rate that the RVLIS indication will accurately trend with RCS inventory. In this case (smaller RCS breaks), other corroborative indications (e.g., CETs and subcooling based on CETs) can also be observed.

For the accident sequences that lead to inadequate core cooling, the accident analyses such as those described in Sections 5 and 6 of WCAP-14696-A, Revision 1 "Westinghouse Owners Group Core Damage Assessment Guidance," indicate that the need for operator action to mitigate these events can be based solely on the CET temperature indication. Other analyses of initiating events that lead to inadequate core cooling and core damage, such as those used as the technical basis for the Westinghouse Owners Group Severe Accident Management Guidance, show that the use of the CET temperature indication alone provides the most appropriate and timely information to the operators for the diagnosis and mitigation of these events.

Therefore, it can be concluded that the RVLIS indication is a secondary indication that can be an ambiguous indicator of an approach to inadequate core cooling and can therefore be relocated from the Technical Specifications to a licensee control document because it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) as discussed on page 31 in Section 5.1 of the WCAP.

4. Page 30 of WCAP-15981 indicated that the RCS hot leg temperature (wide range) indication was a backup to the CETs for indicating that the core cooling safety function was being accomplished, and thus, was classified as a Category 3 variable. It is also indicated that the RCS cold leg temperature (wide range) indication was a diagnostic indication, and thus, was classified as a Category 3 variable. Both RCS temperature monitors were proposed to relocate from the Standard TS to a licensee control document.

   It should be noted that both RCS temperature monitors were used by the operator to carry out appropriate actions to effectively accomplish important safety functions for Westinghouse plants during post-accident conditions. For example, the RCS hot leg WR indication was used by the operator to verify adequate core cooling, RCS subcooling, RHR initiation conditions, and in conjunction with the RCS cold leg WR temperature indication, the effectiveness of RCS heat removal by the secondary system. The RCS hot leg temperature indication was also used by the operator to determine if safety injection flow could be reduced. As for the RCS cold leg temperature indication, it was used, in combination with the RCS hot leg temperature indication, to verify the effectiveness of RCS heat removal by the secondary system. The RCS cold leg temperature indication was used by the operator during the steam generator depressurization to assure that the depressurization did not impose a challenge to the integrity critical functions. The reviewer realized that functional diversity for determination of core cooling would include CETs, RVWL system, and subcooling monitors, and functional redundancy for determination of secondary heat sink would be provided by steam generator water level, AFW flow and CETs.

For the RCS integrity and heat removal safety functions, as well as the emergency operating procedures effectiveness, the RG 1.97 indicated that the measurement of a single key variable were not sufficient to assure the accomplishment of a given safety function. Where multiple variable were needed to assure the accomplishment of a giving safety function, it was essential that they each be considered key variable and measured with high quality instrumentation.

Address the consistency with the RG 1.97 guidance referenced above for the proposed relocation of the RCS hot leg and cold leg temperature instruments out of the standard TS.

Response:

The plant EOPs contain guidance for responding to reactor trip and/or safety injection initiation that addresses three critical stages of an event: recovery of critical safety functions; immediate accident diagnosis and mitigation to achieve a safe, stable state; and long term recovery to move from a safe, stable state to a cold shutdown condition. The critical safety function and the accident diagnosis and mitigation stages primarily consist of assuring adequate core cooling and an adequate heat sink. The plant EOPs direct the operators to monitor core temperature and RCS subcooling as the most appropriate means of diagnosing an approach to inadequate core cooling. The EOP background documents typically list the CETs as the preferred means of satisfying these requirements. For LOCAs, SI flow can also be used as a secondary indicator of adequate core cooling. Other indications, such as hot leg RTDs are available to provide additional information that these functions are being satisfied. It should also be noted that the measurement range of the hot leg RTDs (as well as cold leg RTDs) is very limited (e.g., ~600°F) and therefore the RTDs are not a useful indicator of inadequate core cooling. The most direct indications for ensuring an adequate heat sink are the CETs, SG Water Level and AFW Flow indications. The EOPs base the response to an inadequate heat sink (F-0.3, "Heat Sink") on the SG Water Level and AFW Flow indications. Other indications such as hot and cold leg RTDs and RCS subcooling monitors can be used as secondary indicators of adequate heat sink.

The PAM instrumentation that should be included in the plant Technical Specifications should only be the primary means of accident diagnosis and mitigation to achieve a safe, stable state, which would satisfy Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii). Backup instrumentation, while useful, should not be included in the plant Technical Specifications, since it does not satisfy Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii). Thus, the proposed relocation of the RCS hot leg and RCS cold leg temperature indications out of the Technical Specifications is consistent with the application of Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) to these indications.

Request for Additional Information (RAIs)
[Received May 3, 2007 via an e-mail from S. Peters (NRC) NRC to C. DiMuzio (W)]

Background

Regulatory Guide (RG) 1.97 provides an acceptable method for complying with the Commission's regulations to provide instrumentation to monitor plant variables and systems during and after an accident. Each operating reactor has been reviewed against the recommendations of RG 1.97.

The NRC staff recognizes that the goal of WCAP-15981 is to justify changes to the list of variables that each licensee includes in their post-accident monitoring (PAM) technical specifications (TSs). The basis for the justification in WCAP-15981 is to include variables that satisfy 10 CFR 50.36(c)(2)(ii) Criterion 3 or 4 in the PAM TSs. However, the NRC staff has the responsibility to ensure that changes to a licensee's PAM TSs are reviewed to determine if these changes also constitute changes in the licensee's commitments to RG 1.97. Therefore, any proposed change in type or category of a variable related to WCAP-15981 could be a deviation from RG 1.97 and would need to be reviewed as a potential generic deviation from RG 1.97.

Any changes in the type or category of instrumentation provided by a licensee would need to be submitted to the NRC staff for review as a potential deviation from the licensee's commitment to RG 1.97. However a change in a variable's Type A status could be done under the provisions of 10 CFR 50.59 and would not need to be submitted to the staff for review unless the change included a previously unreviewed deviation or a TS change. A change that involved a variable that is currently a Category 1, Type B, C, D, or E variable and is also being declared a Type A variable or is currently a Type A variable and is reverting back to only being a Category 1, Type B, C, D, or E variable could be done under the provisions of 10 CFR 50.59.

Response:

As discussed in the response to General Comment No. 2 from the Plant Systems Branch that is contained in Attachment 1 to WOG-06-104, the Regulatory Guide 1.97 reclassification of the instrumentation was performed to reflect how the instrumentation is currently utilized in accident management, as opposed to the classification identified when the original plant specific Regulatory Guide 1.97 evaluations were performed. For consistency, the Regulatory Guide 1.97 classification should be consistent with the instrumentation proposed to be included in the PAM Technical Specification. Criteria 3 and 4 of 10 CFR 50.36 were utilized to determine whether the instrumentation should be included in the PAM Technical Specification, not the Regulatory Guide 1.97 reclassification of the instrumentation.

Instrumentation that was classified in WCAP-15981 as Regulatory Guide 1.97 Type A satisfies 10 CFR 50.36(c)(2)(ii) Criterion 3. Instrumentation that provides primary information needed to permit the operators to take manual actions for which no automatic actions are provided to satisfy a DBA safety function (Type A definition) are part of the primary success path that functions to mitigate a DBA that either assumes a failure of or presents a challenge to the integrity of a fission product barrier (Criterion 3 definition). Non-Type A instrumentation that was classified in WCAP-15981 as Regulatory Guide 1.97 Category 1 satisfies 10 CFR 50.36(c)(2)(ii) Criterion 4. The non-Type A instrumentation that provides direct indication on the accomplishment of a safety function (Category 1 definition) are those which the probabilistic risk assessment has shown to be significant to public health and safety (Criterion 4 definition). All other instrumentation that has a lower Regulatory Guide 1.97 classification does not satisfy Criterion 3 or 4 of 10 CFR 50.36 and should not be included in the PAM Technical Specification.

Conversely, all instrumentation that does not meet either Criterion 3 or Criterion 4 of 10 CFR 50.36(c)(2)(ii) should not be classified as either a Regulatory Guide 1.97 Type A or Category 1 indication.

The reclassification of the instrumentation proposed to be included in the PAM Technical Specification was performed solely to determine whether it satisfied Criteria 3 and 4 of 10 CFR 50.36, not with respect to the classifications and categories of design and qualification criteria associated with Regulatory Guide 1.97. A licensee's commitments to Regulatory Guide 1.97 are not changed by the proposed changes to the PAM Technical Specification.

RAI 1 Part A.

The purpose of the May 16, 2005, RAI Question 1 was to ensure that, with the proposed WCAP-15981 changes, key Category 1 variables would remain that provide information for each Regulatory Guide 1.97, "Criteria For Accident Monitoring Instrumentation For Nuclear Power Plants," Category 1 function (i.e., Reactivity Control, Core Cooling, Maintaining Reactor Coolant System (RCS) Integrity, Maintaining Containment Integrity, Fuel Cladding, Reactor Coolant Pressure Boundary, Containment, Primary Coolant System Status, Secondary System Status, Auxiliary Feedwater (AFW) System Status, and Containment Radiation).

The PWROG should indicate which variables are considered the key variables for each Category 1 function listed in RG 1.97. If there is a determination that a RG 1.97 Category 1 function should be downgraded to Category 2 or 3, provide additional justification directly addressing the potential downgrade of the function.

Response:

As discussed in Section 5 and summarized in Table 10 of WCAP-15981, the following variables are considered key variables for each of the Category 1 functions listed in Regulatory Guide 1.97:

Reactivity Control – Power Range Neutron Flux Indication

Core Cooling – Core Exit Temperature, RCS Pressure (Wide Range) High Head Safety Injection Flow and RWST Level Indications

Maintaining Reactor Coolant System Integrity – RCS Pressure (Wide Range), Containment Pressure (Wide Range) and Core Exit Temperature Indications

Maintaining Containment Integrity – Containment Pressure (Wide Range) and Penetration Flow Path Containment Isolation Valve Position Indications

Fuel Cladding –Core Exit Temperature Indications

Reactor Coolant System Pressure Boundary – RCS Pressure (Wide Range), Core Exit Temperature, Pressurizer Level, Containment Pressure (Wide Range) and SG Level (Wide Range) Indications

Containment – Containment Pressure (Wide Range) Indication

Primary Coolant System Status – RCS Pressure (Wide Range), Pressurizer Level and Core Exit Temperature Indications

Secondary System Status – SG Level (Wide Range) and SG Pressure Indications

Auxiliary Feedwater or Emergency Feedwater System Status – AFW Flowrate Indication

Containment Radiation – Containment Area Radiation Indication

Table 11 of the WCAP summarizes the basis for reclassification of the current Category 1 variables contained in Technical Specification 3.3.3, "PAM Instrumentation," of NUREG-1431, that are proposed to be relocated to licensee controlled documents.

RAI 1 Part B.

The PWROG should discuss how Criterion 1 of 10 CFR 50.36(c)(2)(ii) is met for each variable (Neutron Flux source range, RCS Hot Leg Temperature, RCS Cold Leg Temperature, Reactor Vessel Water Level, Containment Sump Water Level wide range, and Condensate Storage Tank Level) in Table 11 of WCAP-15981 that is being proposed for removal from the TS.

Response:

10 CFR 50.36 (c)(2)(ii) Criterion 1 states "Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary." The instrumentation that satisfies Criterion 1 is contained in Technical Specification 3.4.15, "RCS Leakage Detection Instrumentation," of NUREG-1431. The LCO requirements in Technical Specification 3.4.15 are not impacted by the changes proposed to Technical Specification 3.3.3 of NUREG-1431.

Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," identifies acceptable methods for selecting RCS leakage detection systems. Therefore Criterion 1 of 10 CFR 50.36 (c)(2)(ii) is not impacted by the variables identified in Table 11 of the WCAP that are proposed to be relocated from Technical Specification 3.3.3 of NUREG-1431.

RAI 1 Part C.

Based on the responses to the May 16, 2005, RAIs and a review of WCAP-15981, it became apparent to the NRC staff that the PWROG was recommending changes in RG 1.97 type classifications for a number of RG 1.97 variables. The NRC staff has identified the following change recommendations where further information is needed:

a. RG 1.97 recommends Type B Category 1 Neutron Flux source range instrumentation for function detection and accomplishment of mitigation of **Reactivity Control**. WCAP-15981 recommends that Neutron Flux source range be classified as Type B Category 3.

   Provide justification and include the variable(s) that would provide information for function detection and accomplishment of mitigation of **Reactivity Control**.

   Response:

   As discussed on page 29 in Section 5.1 of the WCAP, the Power Range Neutron Flux indication provides the most direct indication of the accomplishment of the Reactivity Control safety function. The Source Range Neutron Flux indication provides the verification of the automatic actuation of the Reactor Protection System (RPS) and is therefore, a Type B variable. In addition, the Source Range Neutron Flux indication provides diagnostics of <u>continued</u> subcriticality during RCS cooldown and depressurization and is therefore, a Category 3 variable.

b. RG 1.97 recommends Type B, Category 1 RCS Hot Leg Temperature instrumentation for function detection, accomplishment of mitigation, verification, long term surveillance for **Core Cooling**. WCAP-15981 recommends that RCS Hot Leg Temperature be classified as Type B, Category 3.

Provide justification and include the variables that would provide information for function detection, accomplishment of mitigation, verification, and long term surveillance of **Core Cooling**.

Response:

As discussed on page 36 in Section 5.1 of the WCAP, the Core Exit Temperature indication provides the most direct indication of the accomplishment of the Core Cooling function. The RCS Hot Leg Temperature indication provides confirmatory information to indicate whether the Core Cooling safety function is being accomplished and is therefore, a Type B variable. In addition, the RCS Hot Leg Temperature indication provides backup diagnostics to the Core Exit Temperature and High Head SI Flow indications and is therefore, a Category 3 variable.

c. RG 1.97 recommends Type B, Category 1 RCS Cold Leg Temperature instrumentation for function detection, accomplishment of mitigation, verification, and long term surveillance for **Core Cooling**. WCAP-15981 recommends that RCS Cold Leg Temperature be classified as Type B, Category 3.

Provide justification and include the variables that would provide information for function detection, accomplishment of mitigation, verification, and long term surveillance of **Core Cooling**.

Response:

As discussed on page 36 in Section 5.1 of the WCAP, the Core Exit Temperature indication provides the most direct indication of the accomplishment of the Core Cooling function. The RCS Cold Leg Temperature indication provides confirmatory information to indicate whether the Core Cooling safety function is being accomplished and is therefore, a Type B variable. In addition, RCS Cold Leg Temperature indication provides backup diagnostics to the Core Exit Temperature and High Head SI Flow indications and is therefore, a Category 3 variable.

d. RG 1.97 recommends Type B, Category 1 RCS Pressure wide range instrumentation for function detection, accomplishment of mitigation, verification, and long term surveillance of **Core Cooling** and function detection and accomplishment of mitigation for **Maintaining RCS Integrity** and Type C, Category 1 instrumentation to provide detection of potential or actual breach, accomplishment of mitigation, and long term surveillance of **Reactor Coolant Pressure Boundary**. WCAP-15981 recommends that RCS Pressure wide range be classified as Type A, Category 1.

Although RCS Pressure wide range would be classified to be Type A, Category 1, would it also remain as a Type B, Category 1 variable and a Type C, Category 1 variable? If not, provide justification and include the variables that would provide key information for function detection, accomplishment of mitigation, verification, and long term surveillance of **Core Cooling**; function detection and accomplishment of mitigation for **Maintaining RCS Integrity**; detection of potential or actual breach, accomplishment of mitigation, and long term surveillance of the **Reactor Coolant Pressure Boundary**.

Response:

For the purposes of determining the instrumentation that should be included in the Technical Specifications, i.e., that satisfies Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii), the "highest" Regulatory Guide 1.97 Type (A) and Category (1) was identified, since this instrumentation satisfies either Criterion 3 or 4. The instrumentation may also be classified as other Types and Categories, however that information is not necessary to determine whether it should be included in the Technical Specifications, only the highest Regulatory Guide 1.97 Type (A) and Category (1) is needed for this determination.

The RCS Pressure Wide Range indication is a Type A, Category 1 variable, and therefore it satisfies Criteria 3 and 4 of 10 CFR 50.36 (c)(2)(ii).

The RCS Pressure Wide Range indication is also a Type B, Category 1 variable for the Core Cooling and the Maintaining RCS Integrity safety functions and a Type C, Category 1 variable for the Reactor Coolant Pressure Boundary safety function.

e. RG 1.97 recommends Type B, Category 1 Reactor Vessel Water Level or Coolant Inventory instrumentation for verification and accomplishment of mitigation of **Core Cooling.** WCAP-15981 recommends that Reactor Vessel Water Level be classified as Type B, Category 3.

   Provide justification and include the variables that would provide information for verification and accomplishment of mitigation of **Core Cooling.**

   Response:

   As discussed on page 36 in Section 5.1 of the WCAP, Core Exit Temperature indication provides the most direct indication of the accomplishment of the Core Cooling safety function. The Reactor Vessel Level Instrumentation System (RVLIS) indication provides confirmatory information to indicate whether the Core Cooling safety function is being accomplished and is therefore, a Type B variable. The issues associated with the use of the RVLIS indication are discussed further in the response to RAI No. 3 of the March 5, 2007 RAIs. RVLIS provides backup diagnostics to the Core Exit Temperature indication and is therefore, a Category 3 variable.

f. RG 1.97 recommends Type B, Category 1 Containment Sump Water Level wide range instrumentation for function detection, accomplishment of mitigation, and verification of **Maintaining RCS Integrity** and Type C, Category 1 Containment Sump Water Level wide range instrumentation for detection of breach, accomplishment of mitigation, verification, and long term surveillance of **Reactor Coolant Pressure Integrity**. WCAP-15981 recommends that Containment Sump Water Level wide range be classified as Type B, Category 2.

   Provide justification and include the variables that would provide information for function detection, accomplishment of mitigation and verification of **Maintaining RCS Integrity**. Also provide justification and include the variables that would provide information for detection of breach, accomplishment of mitigation, verification, and long term surveillance of **Reactor Coolant Pressure Boundary**.

   Response:

   As discussed in Section 5.1 of the WCAP, the Core Exit Temperature (see page 36 of the WCAP) and RCS Pressure Wide Range (see page 31 of the WCAP) indications provide information of the accomplishment of the Maintaining RCS Integrity safety function following an accident. The RCS Pressure Wide Range, Pressurizer Level (see page 34 of the WCAP) and SG Water Level Wide Range (see page 35 of the WCAP) indications also provide information of the accomplishment of the Reactor Coolant Pressure Boundary safety function. The Containment Sump Water Level Wide Range indication (see page 37 of the WCAP) provides information to indicate whether the Core Cooling safety function can be accomplished when RWST switchover to recirculation occurs. While the Containment Sump Water Level Wide Range indication can provide a direct indication of the potential degradation of the RCS pressure boundary, it is not the only indication or the most direct indication that can be used for this diagnosis. Degradation of the RCS pressure boundary can more appropriately be indicated by the SI Flow and Pressurizer Level indications. Therefore, Containment Sump Water Level Wide Range indication is a Type B variable. In addition, the Containment Sump Water Level Wide Range indication provides information on the status of ECC recirculation flow delivery and is therefore, a Category 2 variable.

The Containment Sump Water Level Wide Range indication only provides backup information to other primary indictors for identifying the accomplishment of the Maintaining RCS Integrity safety function, and is a Type B, Category 3 variable. Also, the Containment Sump Water Level Wide Range indication only provides backup information to other primary indictors for identifying the accomplishment of the Reactor Coolant Pressure Integrity safety function and is a Type C, Category 3 variable.

g. RG 1.97 recommends Type C, Category 1 Core Exit Temperature instrumentation for detection of potential for breach, accomplishment of mitigation, and long term surveillance of **Fuel Cladding**. WCAP-15981 recommends that Core Exit Temperature be classified as Type A, Category 1.

   Although Core Exit Temperature would be classified as Type A, Category 1, would it also remain as a Type C, Category 1 variable? If not, provide justification and include the variables that would provide key information for detection of potential for breach, accomplishment of mitigation, and long term surveillance of **Fuel Cladding**.

   Response:

   For the purposes of determining the instrumentation that should be included in the Technical Specifications, i.e., that satisfies Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii), the "highest" Regulatory Guide 1.97 Type (A) and Category (1) was identified, since this instrumentation satisfies either Criterion 3 or 4. The instrumentation may also be classified as other Types and Categories, however that information is not necessary to determine whether it should be included in the Technical Specifications, only the highest Regulatory Guide 1.97 Type (A) and Category (1) is needed for this determination.

   The Core Exit Temperature indication is a Type A, Category 1 variable and therefore satisfies Criteria 3 and 4 of 10 CFR 50.36 (c)(2)(ii).

   The Core Exit Temperature indication is also a Type C, Category 1 variable because it provides information for identifying the Fuel Cladding safety function (as well as a Type B, Category 3 variable because it satisfies the Core Cooling safety function) as discussed in Regulatory Guide 1.97.

h. RG 1.97 recommends Type C, Category 1 Containment Pressure wide range instrumentation for detection of potential for or actual breach, and accomplishment of mitigation of the **Containment**. WCAP-15981 recommends that Containment Pressure wide range be classified as Type A, Category 1.

   Although Containment Pressure wide range would be classified as Type A, Category 1, would it also remain as a Type C, Category 1 variable? If not, provide justification and include the variables that would provide key information for detection of potential for or actual breach and accomplishment of mitigation of the **Containment**.

   RG 1.97 also recommends Type B, Category 1 Containment Pressure instrumentation for function detection, accomplishment of mitigation, and verification of **Maintaining RCS Integrity**; function detection, accomplishment of mitigation, and verification of **Maintaining Containment Integrity**; and Type C, Category 1 Containment Pressure instrumentation to provide detection of breach, accomplishment of mitigation, verification, and long term surveillance of the **Reactor Coolant Pressure Boundary**.

   Confirm that Containment Pressure would remain as a Type B, Category 1 variable for function detection, accomplishment of mitigation, and verification of **Maintaining RCS Integrity**, as a

Type B, Category 1 variable for function detection, accomplishment of mitigation, and verification of **Maintaining Containment Integrity**, and as a Type C, Category 1 variable for detection of breach, accomplishment of mitigation, verification, and long term surveillance of the **Reactor Coolant Pressure Boundary**.

Response:

In WCAP-15981, the Containment Pressure Wide Range indication is classified as a Type B, Category 1 variable for identifying the accomplishment of the Maintaining RCS Integrity safety function; a Type B, Category 1 variable for identifying the accomplishment of the Maintaining Containment Integrity safety function; and a Type C, Category 1 variable for identifying the accomplishment of the Reactor Coolant Pressure Boundary safety function.

i. RG 1.97 recommends Type D, Category 2 High Head SI Flow instrumentation to monitor operation of **Safety Injection** Systems. WCAP-15981 recommends that High Head SI Flow be classified as Type B, Category 1.

Although High Head SI Flow would be classified as Type B, Category 1, would it also remain as a Type D, Category 2 variable? If not, provide justification and include the variables, along with the category classifications, that would provide key information on the operation of the **Safety Injection Systems**.

Response:

For the purposes of determining the instrumentation that should be included in the Technical Specifications, i.e., that satisfies Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii), the "highest" Regulatory Guide 1.97 Type (A) and Category (1) was identified, since this instrumentation satisfies either Criterion 3 or 4. The instrumentation may also be classified as other Types and Categories, however that information is not necessary to determine whether it should be included in the Technical Specifications, only the highest Regulatory Guide 1.97 Type (A) and Category (1) is needed for this determination.

The High Head SI Flow indication is a Regulatory Guide 1.97 Type B, Category 1 variable and therefore satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The High Head SI Flow indication is also a Type D, Category 2 variable because it provides information to identify the accomplishment of the Safety Injection Systems safety function.

j. RG 1.97 recommends Type D, Category 2 Refueling Water Storage Tank (RWST) Level instrumentation to monitor operation of the **Safety Injection Systems**. WCAP-15981 recommends that RWST Level be classified as Type D, Category 1. For plants where switchover to recirculation is based on RWST level indication rather than RWST Level alarm, WCAP-15981 is recommending that RWST Level be classified as Type A, Category 1.

Although RWST Level would be classified as a Type A, Category 1 variable for some plants, would it also remain as a Type D, Category 1 variable? If not, provide justification and include the variables, along with the category classifications, that would provide key information on the operation of the **Safety Injection Systems**.

Response:

For the purposes of determining the instrumentation that should be included in the Technical Specifications, i.e., that satisfies Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii), the "highest" Regulatory Guide 1.97 Type (A) and Category (1) was identified, since this instrumentation satisfies either Criterion 3 or 4. The instrumentation may also be classified as other Types and Categories, however that information is not necessary to determine whether it should be included in the Technical Specifications, only the highest Regulatory Guide 1.97 Type (A) and Category (1) is needed for this determination.

The RWST Level indication is a Regulatory Guide 1.97 Type A, Category 1 variable for some plants, and therefore satisfies Criteria 3 and 4 of 10 CFR 50.36 (c)(2)(ii), as discussed in the RAI above.

For other plants, RWST Level indication is a Type D, Category 1 variable because it provides information on the accomplishment of the Safety Injection Systems safety function.

The RWST Level indication is also a Type D, Category 2 variable because it provides indication of operation of the Safety Injection Systems safety function.

k. RG 1.97 recommends Type D, Category 1 Pressurizer Level instrumentation to ensure proper operation of the pressurizer in the **Primary Coolant System**. WCAP-15981 recommends that Pressurizer Level be classified as Type A, Category 1.

   Although Pressurizer Level would be classified as Type A, Category 1, would it also remain as a Type D, Category 1 variable? If not, provide justification and include the variables that would provide the key information to ensure proper operation of the pressurizer in the **Primary Coolant System**.

   Response:

   For the purposes of determining the instrumentation that should be included in the Technical Specifications, i.e., that satisfies Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii), the "highest" Regulatory Guide 1.97 Type (A) and Category (1) was identified, since this instrumentation satisfies either Criterion 3 or 4. The instrumentation may also be classified as other Types and Categories, however that information is not necessary to determine whether it should be included in the Technical Specifications, only the highest Regulatory Guide 1.97 Type (A) and Category (1) is needed for this determination.

   The Pressurizer Level indication is a Type A, Category 1 variable and therefore satisfies Criteria 3 and 4 of 10 CFR 50.36 (c)(2)(ii).

   The Pressurizer Level indication is also a Type D, Category 1 variable based on providing information to ensure proper operation of the pressurizer in the Primary Coolant System safety function.

l RG 1.97 recommends that Type D, Category 1 Steam Generator Water Level wide range instrumentation monitor operation of the **Secondary System**. WCAP-15981 recommends that Steam Generator Water Level wide range be classified as Type A, Category 1.

   Although Steam Generator Water Level wide range would be classified as Type A, Category 1, would it also remain as a Type D, Category 1 variable? If not, provide justification and include the variables that would provide the key information to monitor the operation of the **Secondary System**.

   Response:

   For the purposes of determining the instrumentation that should be included in the Technical Specifications, i.e., that satisfies Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii), the "highest" Regulatory Guide 1.97 Type (A) and Category (1) was identified, since this instrumentation satisfies either Criterion 3 or 4. The instrumentation may also be classified as other Types and Categories, however that information is not necessary to determine whether it should be included in the Technical Specifications, only the highest Regulatory Guide 1.97 Type (A) and Category (1) is needed for this determination.

The SG Water Level Wide Range indication is a Type A, Category 1 variable and therefore satisfies Criteria 3 and 4 of 10 CFR 50.36 (c)(2)(ii).

The SG Water Level Wide Range indication is also a Type D, Category 1 variable because it provides information to monitor the operation of the Secondary System safety function.

m. RG 1.97 recommends that Type D, Category 2 Steam Generator Pressure instrumentation monitor operation of the **Secondary System**. WCAP-15981 recommends that Steam Generator Pressure be classified as Type A, Category 1.

   Although Steam Generator Pressure would be classified as Type A, Category 1, would it also remain as a Type D, Category 2 variable? If not, provide justification and include the variables, along with the category classifications, that would provide backup information for the **Secondary System**.

   Response:

   For the purposes of determining the instrumentation that should be included in the Technical Specifications, i.e., that satisfies Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii), the "highest" Regulatory Guide 1.97 Type (A) and Category (1) was identified, since this instrumentation satisfies either Criterion 3 or 4. The instrumentation may also be classified as other Types and Categories, however that information is not necessary to determine whether it should be included in the Technical Specifications, only the highest Regulatory Guide 1.97 Type (A) and Category (1) is needed for this determination.

   SG Pressure indication is a Type A, Category 1 variable and therefore satisfies Criteria 3 and 4 of 10 CFR 50.36 (c)(2)(ii).

   SG Pressure indication is also a Type D, Category 2 variable because it provides information on the Secondary System safety function.

n. RG 1.97 recommends Type D, Category 2 AFW Flow instrumentation to monitor the operation of the **AFW System**. WCAP-15981 recommends that AFW Flow be classified as Type B, Category 1.

   Although AFW Flow would be classified as Type B, Category 1, would it also remain as a Type D, Category 2 variable? If not, provide justification and include the variables that would provide backup information for monitoring the operation of the **AFW System**.

   Response:

   For the purposes of determining the instrumentation that should be included in the Technical Specifications, i.e., that satisfies Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii), the "highest" Regulatory Guide 1.97 Type (A) and Category (1) was identified, since this instrumentation satisfies either Criterion 3 or 4. The instrumentation may also be classified as other Types and Categories, however that information is not necessary to determine whether it should be included in the Technical Specifications, only the highest Regulatory Guide 1.97 Type (A) and Category (1) is needed for this determination.

   The AFW Flow indication is a Type B, Category 1 variable and therefore satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

   The AFW Flow indication is also a Type D, Category 2 variable because it provides information for monitoring the operation of the AFW System safety function.

o. RG 1.97 recommends Type D, Category 1 Condensate Storage Tank Level instrumentation to ensure water supply for AFW in the **AFW System**. WCAP-15981 recommends that Condensate Storage Tank Level be classified as Type B, Category 2.

Although Condensate Storage Tank Level would be classified as Type B, Category 2, would it also remain as a Type D, Category 1 variable? If not, provide justification and include the variables that would provide information to ensure water supply for AFW in the **AFW System**.

Response:

As discussed on page 35 in Section 5.1 of the WCAP, the Condensate Storage Tank Level indication provides information on whether the SG heat sink can be maintained from this source and is therefore, a Type B variable. The Condensate Storage Tank Level indication does not provide information to indicate the operation of the AFW System safety function, which is provided by AFW Flowrate (see page 36 of the WCAP) and SG Level Wide Range indications (see page 35 of the WCAP). Therefore, Condensate Storage Tank Level indication is not considered to be a Type D variable for the SG heat sink function. In addition, the Condensate Storage Tank Level indication provides information indicating long term AFW System safety function operating status and is therefore, a Category 2 variable. The key variables that provide the indications of the accomplishment of the heat sink safety function are the SG Level Wide Range (see page 35 of the WCAP) and AFW Flow (see page 36 of the WCAP) indications. Therefore, the Condensate Storage Tank Level indication is not considered to be a Category 1 variable.

The Condensate Storage Tank Level indication is also a Type D, Category 3 variable because it provides information to indicate the operation of the auxiliary feedwater system and is a backup variable for monitoring the operation of the AFW System safety function.

p. RG 1.97 recommends Type E, Category 1 Containment Area Radiation high range instrumentation for detection of significant releases, release assessment, long term surveillance, and emergency plan actuation for **Containment Radiation**. WCAP-15981 recommends that Containment Area Radiation high range be classified as Type C, Category 1.

Although Containment Area Radiation high range would be classified as Type C, Category 1, would it also remain as a Type E, Category 1 variable? If not, provide justification and include the variables that would provide the key information to ensure detection of significant releases, release assessment, long term surveillance, and emergency plan actuation for **Containment Radiation**.

Response:

For the purposes of determining the instrumentation that should be included in the Technical Specifications, i.e., that satisfies Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii), the "highest" Regulatory Guide 1.97 Type (A) and Category (1) was identified, since this instrumentation satisfies either Criterion 3 or 4. The instrumentation may also be classified as other Types and Categories, however that information is not necessary to determine whether it should be included in the Technical Specifications, only the highest Regulatory Guide 1.97 Type (A) and Category (1) is needed for this determination.

The Containment Area Radiation High Range indication is a Type C, Category 1 and therefore satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The Containment Area Radiation High Range indication is also a Type C, Category 3 variable because it indicates operation of the Reactor Coolant Pressure Boundary safety function.

The Containment Area Radiation High Range indication is also a Type E, Category 1 variable because it provides key information to ensure a release assessment for the Containment Radiation safety function.

2. In response to the May 16, 2005, RAI Question 4, the PWROG stated that Neutron Flux source range instrumentation does not meet either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii). However, the PWROG stated, "In the longer term, the potential for recriticality is only a concern during RCS depressurization to cold shutdown conditions for accident sequences where significant borated water has not been injected into the RCS. In this case, the EOPs instruct the operators to determine the required RCS shutdown boron concentration and then borate the RCS to the required level before proceeding with RCS cooldown and depressurization." This appears to argue that Boron Concentration information is important to the operator and therefore, Boron Concentration would satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii). Furthermore, the response to RAI Question 4 does not indicate what instrumentation would be used to detect a return to criticality. Provide information on the instrumentation used to detect a return to criticality.

   Based on the argument presented either Neutron Flux source range or Boron Concentration meet Criterion 4 of 10 CFR 50.36 (c)(2)(ii) and should be included in the PAM TS. Address these issues. This question requests information similar to the information requested in item 1k above.

   Response:

   The EOPs instruct the operators to determine the required RCS shutdown boron concentration and then borate the RCS to the required level before proceeding with RCS cooldown and depressurization. The EOPs recommend that the determination of the boron concentration be determined from RCS samplings, as opposed to boron concentration indications. The RCS boron concentration is considered to be the primary method used to determine the potential for a return to criticality during RCS cooldown and depressurization following an accident. The EOPs direct the operator to verify the boron concentration by an RCS sample. Additionally, boron concentration is typically not indicated in the control room, or if indicated in the control room, it is not used in the EOP decision making. Therefore, neither the Neutron Flux Source Range indication, as discussed in the May 16, 2005 RAI 4 response referred to above, nor the RCS boron concentration satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii).

3. In response to May 16, 2005, RAI Question 7, the WOG stated that AFW Flow and Steam Generator Level instrumentation provide information for diagnosing issues related to the performance of the AFW system. The primary diagnosis used in the EOPs for inadequate AFW performance is AFW flow. The secondary symptom used in the EOPs to diagnose inadequate AFW performance is decreasing Steam Generator Level. Condensate Storage Tank Level is not used in the diagnosis of inadequate AFW performance.

   WCAP-15981 recommends that AFW Flow be reclassified as a Type B, Category 1 variable, but does not specify which RG 1.97 safety system function the AFW Flow performs. WCAP-15981 also recommends that since Condensate Storage Tank Level does not satisfy Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii), Condensate Storage Tank Level be reclassified as a Type B, Category 2 variable.

   However, the PWROG has not addressed which Type D variables provide key information concerning AFW Flow system status. What variables provide AFW System Status? Is WCAP-15981 recommending that, in addition to the Type B classification of AFW flow and Condensate Storage Tank Level, it should also be classified as a Type D variable? Would this recommendation include AFW Flow as the Type D, Category 1 key variable for the AFW System Status and Condensate Storage Tank Level as a Type D, Category 2 variable? This question requests information that is similar to information requested in items 1n and 1o above.

Response:

For the purposes of determining the instrumentation that should be included in the Technical Specifications, i.e., that satisfies Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii), the "highest" Regulatory Guide 1.97 Type (A) and Category (1) was identified, since this instrumentation satisfies either Criterion 3 or 4. The instrumentation may also be classified as other Types and Categories, however that information is not necessary to determine whether it should be included in the Technical Specifications, only the highest Regulatory Guide 1.97 Type (A) and Category (1) is needed for this determination.

The Condensate Storage Tank Level indication does not satisfy either Criterion 3 or Criterion 4 of 10 CFR 50.36(c)(2)(ii) and therefore should not be included in the Technical Specifications. Based on the discussion on page 35 in Section 5.1 of the WCAP, this indication should be a Type B, Category 2.

As discussed above in response to RAI No. 1 Part C, paragraph (o), the Condensate Storage Tank Level indication is also a Type D, Category 3 variable.

The AFW Flow indication is a Type B, Category 1 variable as discussed on pages 36 and 37 in Section 5.1 of the WCAP, and therefore satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The AFW Flow indication also provides information on AFW System safety function, and is also a Type D, Category 2 variable.

4. Note (1) of Tables 10 and 11 of WCAP-15981 read, "Only the highest RG 1.97 classification is shown in this table." RCS Pressure wide range, Core Exit Temperature, Containment Pressure wide range, RWST Level, Pressurizer Level, Steam Generator Pressure, AFW Flow, and Containment Area Radiation high range are listed in Table 10 of WCAP-15981 as currently classified as Type A, Category 1. RCS Hot Leg Temperature, RCS Cold Leg Temperature, Containment Sump Water Level wide range, and Condensate Storage Tank Level are listed in Table 11 of WCAP-15981 as currently classified as Type A, Category 1. Listing only the highest classification is misleading. Tables 10 and 11 should list all type and category information applicable to each variable in that table.

Response:

For the purposes of determining the instrumentation that should be included in the Technical Specifications, i.e., that satisfies Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii), the "highest" Regulatory Guide 1.97 Type (A) and Category (1) was identified, since this instrumentation satisfies either Criterion 3 or 4. The instrumentation may also be classified as other Types and Categories, however that information is not necessary to determine whether it should be included in the Technical Specifications, only the highest Regulatory Guide 1.97 Type (A) and Category (1) is needed for this determination. Therefore not all Regulatory Guide 1.97 Types and Categories for each instrument were identified in the WCAP, since this information was not needed to determine the instrumentation that should be included in the PAM Technical Specification, and the WCAP was not revised to include this information.

The associated RAI 1 Part C responses that identified the additional Regulatory Guide 1.97 Types and Categories for each of the instruments discussed above are identified below.

Power Range Neutron Flux – See RAI 1 Part C (a)

RCS Hot Leg Temperature – See RAI 1 Part C (b)

RCS Cold Leg Temperature – See RAI 1 Part C (c)

RCS Pressure (Wide Range) – See RAI 1 Part C (d)

Reactor Vessel Water Level – See RAI 1 Part C (e)

Containment Sump Water Level (Wide Range) – See RAI 1 Part C (f)

Core Exit Temperature – See RAI 1 Part C (g)

Containment Pressure (Wide Range) – See RAI 1 Part C (h)

High Head SI Flow – See RAI 1 Part C (i)

RWST Level – See RAI 1 Part C (j)

Pressurizer Level – See RAI 1 Part C (k)

Steam Generator Water Level (Wide Range) – See RAI 1 Part C (l)

Steam Generator Pressure – See RAI 1 Part C (m)

AFW Flow – See RAI 1 Part C (n) and RAI 3

Condensate Storage Tank Level – See RAI 1 Part C (o) and RAI 3

Containment Area Radiation (High Range) – See RAI 1 Part C (p)

5. The PWROG indicated in Table 11 of WCAP-15981 that the source range neutron flux indication provides verification of automatic actuation of RPS, and diagnostics of continued subcriticality during RCS cooldown and depressurization. Therefore, the PWROG reclassified the source range neutron flux indication as a B3 variable, claimed that it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii), and proposed to remove it from the TS.

In SRP Section 15.4.6, the NRC staff requires that at least 15-minutes be available from the time the operator is made aware of an unplanned boron dilution event to the time a total loss of shutdown margin occurs during power operation, startup, hot standby, hot shutdown and cold shutdown. A warning time of 30 minutes is required during refueling.

Discuss how the source range neutron flux instrument was used for Westinghouse plants to meet the SRP 15.4.6 guidance, and justify that the source range neutron flux indication and alarm do not satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii), which states that a TS LCO is required for a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Response:

The PAM Technical Specification is only applicable in Modes 1, 2, and 3 (i.e., only to accidents occurring in Modes 1, 2, and 3). Therefore the boron dilution analysis will only be discussed for those modes.

SRP Section 15.4.6 addresses the analysis of unplanned boron dilution events (BDE). In the analysis of a BDE in Mode 1 with the rods in automatic control typically credits an alarm to alert the operator that an unplanned boron dilution is occurring. The analysis of a BDE in Mode 1 with the rods in manual control typically credits a reactor trip to mitigate the event and alert the operator that an unplanned boron dilution is occurring. The analysis of a BDE in Mode 2 typically credits a reactor trip to mitigate the event and alert the operator that an unplanned boron dilution is occurring. The analysis of a BDE in Mode 3 typically credits an alarm to alert the operator that an unplanned boron dilution is occurring. Therefore the boron dilution analyses performed in Modes 1, 2, and 3 either credit an alarm or reactor trip to alert the operator that an unplanned boron dilution is occurring.

Additionally, unless the operator would be stationed at and continuously monitoring the Source Range Neutron Flux indication, which would be highly unlikely due to the other required normal control room activities, the indication would not be very useful in detecting an unplanned boron dilution.

Therefore, the Source Range Neutron Flux indication does not satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) and therefore should not be included in the PAM Technical Specification. Applications of the Source Range Neutron Flux instrumentation for Technical Specification 3.3.1, "RTS Instrumentation," and, if applicable, Technical Specification 3.3.9, "Boron Dilution Protection System," are not addressed by WCAP-15981, since WCAP-15981 only addresses Technical Specification 3.3.3.

Attachment 2

Mark-Ups for WCAP-15981, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants"

Page 5a/b:	RAI #1 May 2007
Page 14a/b:	RAI #1 March 2007
Page 41a/b/c :	RAI #2 March 2007
Page 52:	Editorial Change
Page 56a1:	Telephone discussion between Bob Palla (NRC) and Bob Lutz (W) on June 26, 2007
Page 56b & 56b1:	Telephone discussion between Bob Palla (NRC) and Bob Lutz (W) on June 26, 2007
Pages 57 & 58:	Editorial Changes

Table 1	Regulatory Guide 1.97 Classification Criteria by Type
Type	**Definition (paraphrased)**
A	Provide primary information needed to permit the operators to take specified manual actions for which no automatic action is provided and that are required for safety systems to accomplish their safety functions for design basis accidents; it does not include those variables that are associated with contingency actions that may also be identified in written procedures
B	Provide information to indicate whether plant safety functions are being accomplished
C	Provide information to indicate the potential for breach of fission product barriers
D	Provide information to indicate the operation of individual safety systems and other systems important to safety
E	Provide information to determine the magnitude of fission product releases

In addition to these criteria for classifying instrumentation important to safety, Regulatory Guide 1.97 provides a categorization that represents a graded approach to requirements depending on the relative importance to safety for a particular indication. The categorization is identified in Table 2.

Table 2	Regulatory Guide 1.97 Classification Criteria by Category
Category	**Definition (paraphrased)**
1	Key variables that most directly provide information on the accomplishment of a safety function
2	Variables indicating system operating status
3	Backup and diagnostic indications

Instrumentation that was classified in WCAP-19581 as Regulatory Guide 1.97 Type A satisfies 10 CFR 50.36(c)(2)(ii) Criterion 3. Instrumentation that provides primary information needed to permit the operators to take manual actions for which no automatic actions are provided to satisfy a DBA safety function (Type A definition) are also part of the primary success path that functions to mitigate a DBA that either assumes a failure of or presents a challenge to the integrity of a fission product barrier (Criterion 3 definition). Non-Type A instrumentation that was classified in WCAP-15981 as Regulatory Guide 1.97 Category 1 satisfies 10 CFR 50.36(c)(2)(ii) Criterion 4. The non-Type A instrumentation that provides direct indication on the accomplishment of a safety function (Category 1 definition) are those which the probabilistic risk assessment has shown to be significant to public health and safety (Criterion 4 definition). All other instrumentation that has a lower Regulatory Guide 1.97 classification does not satisfy Criterion 3 or 4 of 10 CFR 50.36 and should not be included in the PAM Technical Specification. Conversely, all instrumentation that does not meet either Criterion 3 or Criterion 4 of 10 CFR 50.36(c)(2)(ii) should not be classified as either a Regulatory Guide 1.97 Type A or Category 1 indication.

The reclassification of the instrumentation proposed to be included in the PAM Technical Specification was performed solely to determine whether it satisfied Criteria 3 and 4 of 10 CFR 50.36; not with respect to the classifications and categories of design and qualification criteria associated with Regulatory Guide 1.97. A licensee's commitments to Regulatory Guide 1.97 are not changed by the proposed changes to the PAM Technical Specification.

2.1 WESTINGHOUSE NSSS PLANT ACCIDENT MONITORING INSTRUMENTATION

Technical Specification 3.3.3 "PAM Instrumentation" in NUREG-1431 provides assurance that those display variables that provide information required by the operators during accident situations are available. This information provides the necessary support for the operator to take manual actions for which no automatic action is provided and that are required for safety systems to accomplish their safety functions.

- These essential instruments are identified by licensee documents addressing the recommendations of Regulatory Guide 1.97. Instrumentation for Regulatory Guide 1.97 Type A and Category 1 variables are included Technical Specification 3.3.3 in NUREG-1431. With the exception of the Reactor Coolant System (RCS) Radiation for which there is no instrumentation available for direct measurement, these Regulatory Guide 1.97 Category 1 instruments were included in Technical Specification 3.3.3 in NUREG-1431 based on the NRC 1988 conclusion that these instruments may be important in limiting risk, based on a limited perspective of available PRA results. The instrumentation included in Technical Specification 3.3.3 is identified in Table 3.

- Pressurizer level and RCS subcooling are used to control/terminate Safety Injection (SI) flow during the depressurization to assure that the pressurizer is not overfilled.

Steam Line Break

In the event of a Steam Line Break (SLB), the DBA analyses assume that the operators will terminate SI. While the EOPs also direct the operators to terminate AFW to the faulted SG to minimize an overcooling condition in the RCS, this is typically not part of the response modeled in the design basis analyses. Termination of SI prevents a pressurizer overfill event which would result in the opening of a pressurizer relief valve. Overfilling the pressurizer and opening the relief valve may result in a stuck open relief valve condition since the valves are not designed for water relief. The primary diagnosis of a SLB condition is based on SG pressures. Comparison of SG steam flow between the SGs, and SG water level can also be used to diagnose a SLB accident. Termination of SI is based on a combination of pressurizer level and RCS subcooling.

Inadvertent Operation of the Emergency Core Cooling System During Power Operation

In the event of an Operation of the Emergency Core Cooling System During Power Operation, the DBA analyses assume that the operators will terminate SI according to the plant EOPs, if the plant specific analysis is performed to prevent filling the pressurizer. Termination of SI prevents a pressurizer overfill event which could result in temporary water relief via the pressurizer safety valves (PSVs). Overfilling the pressurizer and water relief via the PSVs may result in a stuck open PSV if the valves are not designed for water relief. It should be noted that the DBA analyses for some plants allow temporary pressurizer PORV or PSV water relief by demonstrating that a more serious plant condition will not result following an inadvertent operation of the emergency core cooling system during power operation. For those plants, the operator action to terminate SI to prevent pressurizer overfilling is not assumed for this event. The SI termination criteria in the EOPs for this DBA, and any other event with SI operating, is based on a combination of the pressurizer level and RCS subcooling indications.

Other Design Basis Accidents

All of the remaining DBA analyses typically do not rely on explicit operator actions. However, inherent in all of these remaining DBA analyses are two operator actions to establish and maintain long term core cooling: controlling AFW flow to maintain a heat sink and prevent SG overfill, and termination of SI to prevent pressurizer overfill. The control of the AFW flow to prevent SG overfill is based on SG level indication. Termination of SI to prevent pressurizer overfill is based on a combination of pressurizer level and RCS subcooling, which is determined from RCS pressure and RCS temperature.

4.2 PROBABILISTIC RISK ASSESSMENT ANALYSIS

PRAs represent a methodology for assessing the outcome of all credible accident sequences. The PRA covers the credible range of accident initiating events, possible equipment failures, and possible operator actions. Unlike design basis analyses, the PRA assesses the consequences of combinations of equipment failures and failures of operator actions. The impact of instrumentation on the accident outcome is modeled in the availability of the automatic actuation systems, as well as in the success of operator actions.

The PRA is particularly useful in assessing the importance of components relative to one another since the PRA is an integrated model that treats all accident initiators and sequences with a common set of assumptions and input data. One of the useful results from a PRA is the importance ranking and the standard importance measures. These results can be used to determine if reduced levels of requirements on various components will significantly impact the overall results, expressed in terms of risk. Typically, the risk important components are those that are required to establish and maintain a long term stable state for high probability accident sequences. On the other hand, those components that are required to establish and maintain a long term stable state for low probability accident sequences and those components that have backup alternative components to accomplish the same function will typically have a lower importance.

The operator actions modeled in the PRA are based on best estimate time windows available to complete various actions to bring the plant to a safe stable state and account for errors in diagnosing both the

Pressurizer PORV Block Valve Position Indication

The PORV Block Valve Limit Switch Position Indication provides information to the control room operators on the position of the pressurizer PORV block valves. It could be used to diagnose the availability of the pressurizer PORVs for use in depressurizing the RCS or to indicate the isolation of a stuck open PORV (LOCA) at lower RCS pressures. Since the PORV Block Valve Limit Switch Position Indication does not provide an indication for operator actions for which no automatic control is provided and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in PAM Technical Specification

PORV Block Valve Limit Switch Position Indication provides information to indicate the status of the pressurizer PORV block valves which are used to isolate the PORVs in the event of excessive PORV leakage, and is therefore a Type D variable. The PORV Block Valve Limit Switch Position Indication provides information on the status of the pressurizer PORV Block Valves for RCS integrity and is therefore a Category 2 variable.

Pressurizer Safety Valve Position Indication

The Pressurizer Safety Valve Position Indication provides information to the control room operators on the position of the pressurizer safety valves. It could be used to diagnose high RCS pressure or a stuck open safety valve (LOCA) at lower RCS pressures. Since the Position Indicator does not provide an indication for operator actions for which no automatic control is provided and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in PAM Technical Specification.

Pressurizer Safety Valve Limit Switch Position Indication provides information to indicate the operation (i.e., position) of the pressurizer safety valves, which are one means to prevent RCS overpressurization, and is therefore a Type D variable. The Pressurizer Safety Valve Limit Switch Position Indication provides information on the status of the pressurizer safety valves for RCS integrity and is therefore a Category 2 variable.

Radiation Effluent Monitors

Some plant specific Technical Specifications for plants that have not converted to NUREG-1431 may include effluent radiation monitors which are identified as PAM instrumentation in the Radiation Monitoring Instrumentation Technical Specification. These radiation monitor indications would typically only be used in the EALs and the OCDM. Since it is expected that the appropriate EAL level will already be specified based on other in-plant instrumentation, these indications are not expected to be safety significant and should not included in the Technical Specifications. Further, requirements for effluent radiation instrumentation for plants that have converted to NUREG-1431 can be relocated to LCDs and are not PAM instrumentation.

AFW Valve Position Indication

The AFW Valve Position Indication provides information to the control room operators on the position of the AFW flow control valves. The AFW flow control valves are adjusted by the operators to maintain steam generator level in the desired range following a reactor trip event. Steam generator level is maintained based on the Steam Generator Water Level Wide Range indication and not the AFW flow control valve position. The AFW Valve Position Indication would only provide useful information to the operators if the steam generator level were behaving in an uncontrolled manner. In addition, the EOPs do not provide guidance for the operators to use the AFW Valve Position Indication for any operator action; all EOP operator actions are cued from either the Steam Generator Water Level Wide Range indication or the AFW Flow indication, both of which are discussed previously. Since the AFW Valve Position Indication does not provide an indication for operator actions for which no automatic control is provided and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in the PAM Technical Specification.

AFW Valve Position Indication provides information to indicate the operation (i.e., position) of the AFW flow control valves, which are used to control steam generator level, and is therefore a Type D variable. The AFW Valve Position Indication provides information on the status of the AFW flow control valves for SG level and is therefore a Category 2 variable.

Boric Acid Tank Level

The Boric Acid Tank Level Indication provides information to the control room operators on the quantity of borated water available for RCS boration. There are two accident classes where boric acid tank level indication might be useful to the operators. The first is for ATWS events, where emergency boration is used and the charging pumps are aligned to take suction from the boric acid tank and the RWST. The second is for non-LOCA events, where the operator chooses to go to cold shutdown and RCS boration is required. For the design basis accident analyses, there are no assumed operator actions based on boric acid tank level. In addition, there are no EOP operator actions cued from the boric acid tank level. In both cases, controlling the boric acid tank level is a long term operator action and local indication is available. Since the Boric Acid Tank Level Indication does not provide an indication for operator actions for which no automatic control is provided and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in the PAM Technical Specification.

The Boric Acid Tank Level control room indication provides information to indicate whether emergency boration can be maintained, and is therefore a Type D variable. The Boric Acid Tank Level control room indication provides information for the long term operating status of boration and is therefore a Category 2 variable.

Containment Enclosure Negative Pressure

The Containment Enclosure Negative Pressure Indication provides information to the control room operators on the operation of the enclosure building exhaust and filtration system. The enclosure building exhaust and filtration system ensures that any leakage from the primary containment is captured and processed through filters thereby reducing the potential releases to the environment. Operation of the enclosure building exhaust and filtration system is credited in the design basis accident analysis for plants with this design feature. Operation of the enclosure building exhaust and filtration system is automatically initiated following a design basis accident. Fission product reduction by the enclosure building exhaust and filtration system is not typically modeled in risk assessments. Since the enclosure building exhaust and filtration system is automatically initiated for design basis accidents and it is not important from a risk perspective, the Containment Enclosure Negative Pressure Indication does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in the PAM Technical Specification.

The Containment Enclosure Negative Pressure control room indication provides information to indicate proper operation of the enclosure building exhaust and filtration system, and is therefore a Type D variable. The Containment Enclosure Negative Pressure control room indication provides information for the operating status of the enclosure building exhaust and filtration system and is therefore a Category 2 variable.

Residual Heat Removal Flow

The Residual Heat Removal (RHR) Flow Indication provides information to the control room operators on the flow through the RHR system that acts as the heat sink for post-accident decay heat removal for accidents involving a breach of the RCS. For accidents in which the RCS is intact, the accident is mitigated using decay heat removal via the SGs and RHR is only used during long term recovery. Following transfer to hot or cold leg ECCS recirculation, the EOPs provide guidance to ensure proper operation of the recirculation function using the SI flow indication. The EOPs do not specify the use of RHR flow as an indicator of successful operation of the system for decay heat removal. Since the RHR flow indication is not used for design basis accidents and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in the PAM Technical Specification.

The RHR Flow control room indication provides information to indicate proper operation of the RHR system, and is therefore a Type D variable. The RHR Flow control room indication provides information for the operating status of the RHR system and is therefore a Category 2 variable.

Spray Additive Tank Level

The Spray Additive Tank Level Indication provides information to the control room operators on the injection of spray additive (e.g. NaOH) available for fission product control and containment sump pH adjustment. The injection of the spray additive to the containment spray flow is passively accomplished with an eductor and there are no operator actions based on the Spray Additive Tank Level Indication. Since the Spray Additive Tank Level Indication does not provide an indication for operator actions for which no automatic control is provided and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in the PAM Technical Specification.

The Spray Additive Tank Level control room indication provides information to indicate proper operation of the spray additive for fission product and containment sump pH control, and is therefore a Type D variable. The Spray Additive Tank Level control room indication provides information for the long term operating status of fission product and containment sump pH control and is therefore a Category 2 variable.

Core Exit Temperature

Any of the CETs can provide the required information for operator actions related to RCS subcooling when the core is covered with water. The risk importance of the CETs is associated with the operator actions to respond to inadequate core cooling conditions from the PRA and from the Emergency Plan notifications of plant conditions that may influence offsite emergency radiological protective actions. An inadequate core cooling condition is assumed in the WOG ERGs if the highest reading CETs are indicating greater than 1200 degrees F. The peripheral rows of CETs are excluded from consideration of inadequate core cooling in the WOG ERGs. The WOG ERG (Reference 11) Background Document for FR-0.2 identifies that the CETs in the outer two rows of assemblies should be excluded from determinations of inadequate core cooling because they can receive significant cooling from SG drainage due to refluxing. The ERG Background Document also identifies that RCS hot leg temperature indications are not recommended for use in determining an inadequate core cooling condition, since the RCS hot leg temperature reacts significantly slower than the core exit temperature to uncovery of the core for some scenarios. The major reason is that the water draining from the SGs to the core can affect the RCS hot leg temperature indication.

For the CDA, the core heatup assessment in WCAP-14696-A, Revision 1 (Reference 16) (pages 5-1 through 5-7) shows that there is a radial temperature gradient in the core during core heatup due to inadequate core cooling. For the purpose of timely diagnosis of an inadequate core cooling condition, the central core exit thermocouple locations provide the most timely indications. The assessment in WCAP-14696-A, Rev. 1 also shows that non-central core exit thermocouple locations can provide a rapid indication of inadequate core cooling if the thermocouple locations in the outer-most assemblies are not used. For example, a comparison of WCAP-14696-A, Rev. 1 Figures 2b and 2c (and 3b vs. 3c) shows that there would be a delay of less than 5 minutes in the diagnosis of inadequate core cooling between the use of the central and non-central/non-peripheral CET locations. Thus, the minimum CET locations to provide information for risk significant operator actions in the EOPs and SAMG are not limited to the most central locations. Two CETs provide adequate feedback based on the relative uniformity of a core heatup during an inadequate core cooling episode.

The conditions at the RCS hot leg RTDs would represent the bulk temperature of the fluid flow from the core under inadequate core cooling conditions. The bulk temperature of the fluid at the RCS hot leg RTD locations would also be significantly reduced from the fluid conditions at the exit of the core, since there would be significant heat loses to structures in the upper core plenum region and the RCS piping between the reactor vessel and the RTD location during the initial phases of the an accident with inadequate core cooling. Also, since the upper indicated range of the RCS hot leg RTDs is 700 degrees F, they may be indicating off-scale high shortly after the "centrally located" CETs indicate an inadequate core cooling condition.

In defining the non-acceptable locations of the CETs in the PAM Technical Specification, the three outer rows were chosen based on the information in WCAP-14696-A, Rev.1, as opposed two outer rows from the ERG basis to provide additional margin for the inadequate core cooling indication (see Figure 1 for clarification). Based on the information in WCAP-14696-A, Rev. 1 and the discussion above, the required number of CET channels proposed to be included in the PAM Technical Specification is two. The recommendation of the required number of CET channels of two, and the exclusion of the CETs in the three outer rows are applicable to all two, three, and four loop Westinghouse NSSS plants.

The RAW and F-V risk importance measures are used to identify the risk important operator actions for both core damage frequency and large early release frequency. For consideration of external events (e.g., fire and seismic initiating events), if a quantitative PRA is available, the risk importance of operator actions can be identified as in the internal events PRA. For qualitative external events risk assessments, the results of the assessments can also be used to identify important operator actions by identifying operator actions required for risk important external events. As noted in Section 3.1 of this report, the evaluation of external initiating events should be limited to ensuring that instrumentation proposed to be relocated from the PAM Technical Specifications is not used for important operator actions to respond to those external initiating events. As discussed in Appendix A of this report, the risk important operator actions are expected to be identified from the at-power, internal events PRA. The risk important operator actions can be identified from the RAW and F-V values. As discussed in Section A.4 of this report, a RAW value greater than 2.0 or an F-V value greater than 0.05 should be used to define risk important operator actions for both core damage frequency and large early release frequency.

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The instrumentation required to support operator actions from the SAMG and the E-Plan would be identified separately since neither the SAMG nor the E-Plan is typically modeled in the PRA using the criteria in Table 5 of this report. This is shown as Step 4 in Table 14. As discussed in earlier in this report, the instrumentation used to support critical SAMG operator actions are those that identify challenges to the containment fission product boundaries. From the E-Plan, the Core Damage Assessment is important because it is used to project offsite doses from an accident and instrumentation used to provide the core damage assessment or the dose projections are important. The EALs are important because they support notification of the offsite authorities and provide a uniform method of ranking the severity of the accident; only the instrumentation that supports the declaration of a General Emergency is considered to be risk important. The generic determination for the Core Damage Assessment determination in this report was based on the use of the approved methodology in WCAP-14696-A. If a licensee has used a different methodology then an assessment of the key indications that support the core damage assessment should be performed based on the actual methodology used.

The next step in the process (Step 5 in Table 14) is to identify the instrumentation associated with the important design basis, PRA or accident management operator actions. This step establishes the relationship between the instrument and the associated operator actions. This would typically involve the use of the plant emergency procedures to identify any instrumentation that provides a cue for initiating these actions, as well as instrumentation used to confirm that the operator action has been successfully completed. Also included in this step is the identification of the minimum set of instrumentation that supports the important actions identified in the previous steps. In some cases, such as steam generator level, some actions can be cued from more than one variable (PAM function), while others can only be cued from a specific variable (PAM function). This step would therefore focus on the minimum set required to support the key operator actions.

Also as part of Step 5, the specific instrumentation for some operator actions that are important to risk can not be identified (as discussed in Appendix A). Examples include operator actions to restore AC power for a station blackout event, operator actions to restore service water, component cooling water and instrument air. In these cases, there are a wide range of symptoms indicating the need for the operator action and no specific instrumentation is relied upon to cue these actions. Therefore, when sufficient cues exist from multiple sources to prompt operator actions that are important to risk, no instrumentation needs to be identified for inclusion in the PAM Technical Specification.

The final step (Step 6 in Table 14) is to identify the instrumentation that can be relocated from the PAM technical specifications to licensee controlled documents. Any instrumentation that does not satisfy Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii), as determined using the process discussed in detail in this report, is a candidate for relocation from the PAM Technical Specification to a licensee controlled document. At this point, an evaluation of the HRA treatment of operator actions in the PRA associated with any variables (instrumentation) proposed to be relocated from the PAM Technical Specification should be performed to ensure that these variables are not important to risk. Also, the external initiating events risk assessment should be reviewed to determine that none of the instrumentation proposed to be relocated from the PAM Technical Specification supports a risk important operator action. The PAM Technical Specification requirements proposed to be relocated by this change will be relocated to licensee controlled documents that are incorporated by reference in the Updated Final Safety Analysis Report (UFSAR) and therefore, all changes to the relocated instrumentation requirements will be controlled by the 10 CFR 50.59 process.

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The proposed change revises the Regulatory Guide 1.97 instrumentation contained in the PAM Technical Specification to be consistent with the technical basis for accident monitoring instrumentation identified in WCAP-15981. This change includes evaluating the current Regulatory Guide 1.97 classification of the affected instrumentation with respect to its function as a post accident monitoring instrument based on WCAP-15981. The results of the WCAP-15981 evaluations performed are for the sole purpose of determining the most appropriate instrumentation to be included in the PAM Technical Specification. The current plant specific response to Regulatory Guide 1.97 (including the instrument type and category classifications) will not be changed as a result of the plant specific implementation of this change. Therefore, there are no changes to the plant specific response to Regulatory Guide 1.97 or the plant design associated with the plant specific implementation of this change.

(Note: only changes based on June 25, 2007 telephone discussion are shown here; all previous changes from PWROG letter OG-06-249 are incorporated into text)

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 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 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

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Note: only changes based on June 25, 2007 telephone discussion are shown here; all previous changes from PWROG letter OG-06-249 are incorporated into text)

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