Westinghouse Non-Proprietary Class 3

WCAP-15981-NP-A
Revision 0

September 2008

Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants

Westinghouse

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Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants

Robert J. Lutz, Jr.*
Risk Applications and Methods - I

September 2008

Reviewer: James D. Andrachek*
Regulatory Compliance and Plant Licensing

Approved: Melissa A. Lucci*, Acting Manager
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Work Performed under MUHP-3036 / PA-LSC-0072

*Electronically approved records are authenticated in the electronic document management system.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 28, 2008

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Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: FINAL SAFETY EVALUATION FOR PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG) TOPICAL REPORT (TR) WCAP-15981-NP, "POST ACCIDENT MONITORING INSTRUMENTATION RE-DEFINITION FOR WESTINGHOUSE NSSS [NUCLEAR STEAM SUPPLY SYSTEM] PLANTS" (TAC NO. MC4524)

Dear Mr. Bischoff:

By letter dated September 17, 2004, the PWROG (formerly the Westinghouse Owners Group) submitted TR WCAP-15981-NP, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," to the U.S. Nuclear Regulatory Commission (NRC) staff. The PWROG submitted supplemental information in response to the NRC's request for additional information by letters dated March 20, 2006, August 10, 2006, June 28, 2007, and August 22, 2007. The PWROG also provided supplemental information in handouts during a September 20, 2007, NRC public meeting. By letter dated October 22, 2007, an NRC draft safety evaluation (SE) regarding our approval of WCAP-15981-NP was provided for PWROG review and comments. By letter dated December 21, 2007, the PWROG commented on the draft SE. The NRC staff's disposition of PWROG's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that WCAP-15981-NP is acceptable for referencing in licensing applications for Westinghouse designed pressurized water reactors to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that PWROG publish an accepted version of this TR within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed final SE after the title page. Also, it must contain historical review information, including NRC requests for additional information and your responses. The accepted version shall include an "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, the PWROG and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

Ho K. Nieh, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Final SE

cc w/encl:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT (TR) WCAP-15981-NP

"POST ACCIDENT MONITORING INSTRUMENTATION RE-DEFINITION FOR

WESTINGHOUSE NSSS [NUCLEAR STEAM SUPPLY SYSTEM] PLANTS"

PRESSURIZED WATER REACTOR OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION AND BACKGROUND

By letter dated September 17, 2004 (Reference 1), the Pressurized Water Reactor Owners Group (PWROG) (formerly the Westinghouse Owners Group) submitted Topical Report (TR) WCAP-15981-NP, "Post Accident Monitoring Re-Definition for Westinghouse NSSS Plants," for U.S. Nuclear Regulatory Commission (NRC) staff review. In response to the NRC staff's requests for additional information by e-mails dated April 11, 2005, and May 10, 2006, the PWROG submitted supplemental material by letters dated March 20, 2006 (Reference 2), August 10, 2006 (Reference 3), June 28, 2007 (Reference 4), and August 22, 2007 (Reference 14). The PWROG also provided supplemental information in handouts during a September 20, 2007, NRC public meeting (Reference 15).

TR WCAP-15981-NP provides technical justification for identifying Post Accident Monitoring (PAM) instrumentation that should be included in the Technical Specifications (TS) for Westinghouse Nuclear Steam Supply System (NSSS) plants. In addition, TR WCAP-15981-NP provides a methodology to be used by licensees to reassess the PAM instrumentation that should be included in the plant-specific TS. TR WCAP-15981-NP was not submitted as a risk-informed application pursuant to Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated November 2002 (Reference 5), but uses probabilistic risk assessment (PRA) information as one element of the overall method to determine the instrumentation to be included in the PAM TS. Given that TR WCAP-15981-NP is not risk-informed, the NRC staff undertook a review of the methodology in order to provide perspectives into how PRA and other information (e.g., Emergency Operating Procedures (EOPs), Severe Accident Management Guidelines (SAMGs), and the Emergency Plan (EP)) would be used collectively to identify instrumentation for inclusion in the PAM TS.

Section 3.2 of this safety evaluation (SE) provides the results of the NRC staff's evaluation of the instrumentation that should be included in the PAM TS, and the instrumentation that can be relocated from the PAM TS, for Westinghouse NSSS plants. The NRC staff's evaluation of the alternate PAM instrumentation proposed in TR WCAP-15981-NP, is also provided in Section 3.2.21. Section 3.3 of this SE provides the results of the NRC staff's evaluation of the methodology to be used by licensees to reassess the PAM instrumentation that should be included in the plant-specific TS.

ENCLOSURE

2.0 REGULATORY EVALUATION

The primary purpose of PAM instrumentation is to display plant variables that provide information required by the control room operator during accident situations. This information provides the necessary support for the operator to take manual actions to initiate safety systems and other appropriate systems important to safety.

2.1 Applicable Regulations

Criterion 13, "Instrumentation and control," of Appendix A to Title 10 of Part 50 of the *Code of Federal Regulations* (10 CFR), requires operating reactor licensees to provide instrumentation to monitor variables and systems over their anticipated ranges for accident conditions as appropriate to ensure adequate safety.

Criterion 19, "Control room," of Appendix A of 10 CFR Part 50 requires operating reactor licensees to provide a control room from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions, including loss-of-coolant accidents.

The regulation at 10 CFR 50.36(c)(2)(ii)(C) requires that TS limiting conditions for operation (LCOs) of a nuclear reactor be established for a structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a design-basis accident (DBA) or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The regulation at 10 CFR 50.36(c)(2)(ii)(D) requires that TS LCOs of a nuclear reactor be established for a SSC which operating experience or PRA has shown to be significant to public health and safety.

2.2 Applicable Regulatory Criteria/Guidelines

RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," dated May 1983 (Reference 6), describes a method acceptable to the NRC staff for complying with the Commission's regulations to provide instrumentation for monitoring plant variables and systems during and after an accident.

RG 1.97 groups the monitored variables into five types. Each type separates the variables based on the general purpose (or function) of the variables. Individual variables may be monitored for multiple functions and therefore belong to multiple types.

- Type A variables provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety function for design basis events.

- Type B variables provide information to indicate whether plant safety functions are being accomplished.

- Type C variables provide information to indicate the potential for being breached or the actual breach of the barriers to fission product releases.

- Type D variables provide information to indicate the operation of individual safety systems and other systems important to safety.

- Type E variables provide information for use in determining the magnitude of a release of radioactive materials and continual assessment of such releases.

RG 1.97 provides design and qualification criteria separated into three categories that provide a graded approach depending on the importance to safety of the measurement of a specific variable. The categories, and the design and qualification criteria associated with each category, are described in RG 1.97. The monitoring of individual variables for multiple functions may result in an individual variable needing to meet multiple design and qualification criteria and as a result belonging to multiple categories.

- Category 1 provides for full qualification, redundancy, and continuous real-time display, and on-site Class 1E power sources.

- Category 2 provides for qualification but is less stringent in that it does not include seismic qualification, redundancy, or continuous display, and only a highly-reliable power source is needed.

- Category 3 provides for high-quality commercial-grade equipment and only offsite power is needed.

This mixture of type and category results in several instruments that need to meet multiple type and category combinations. In cases were a single variable needs to monitor multiple functions, some licensees have provided one set of instrumentation that meets the highest category criteria of the multiple functions for that variable.

2.3 Regulatory Criteria/Guidelines Applicable To PRA

General guidance for evaluating the technical basis of proposed risk-informed changes is provided in Chapter 19.0 of NUREG-0800, Standard Review Plan (SRP) (Reference 7). SRP Chapter 19, Appendix D, "Use of Risk Information in Review of Non-Risk-Informed License Amendment Requests," provides guidance to the NRC staff in determining if "special circumstances" exist for license amendment requests that are not risk-informed. Special circumstances would exist if, even though the application is in compliance with existing regulatory requirements, concerns associated with the application are identified regarding adequate protection of the public. Per the guidance of Appendix D, the NRC staff used elements of the risk-informed decisionmaking process described in RG 1.174 to focus the review.

Although the guidance presented in RG 1.174 does not constitute a definition of adequate protection, it does provide an appropriate set of guidelines that can be used in the initial process in determining the potential for "special circumstances" and in providing a basis for finding that there is reasonable assurance of adequate protection by compliance with the existing regulatory requirements. In addition, SRP Chapter 19 and RG 1.174 state that a risk-informed application should be evaluated to ensure that the proposed change(s) meet five key safety principles:

- The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.
- The proposed change is consistent with the defense-in-depth philosophy.
- The proposed change maintains sufficient safety margins.
- When the proposed changes increase risk, i.e., core damage frequency (CDF) or large early release frequency (LERF), the increases should be small and consistent with the intent of the Commission's Safety Goal Policy.
- The impact of the proposed change should be monitored using performance measurement strategies.

The quality of the PRA supporting the change must be compatible with the safety implications of the TS change being requested, and the degree to which the decision relies on the risk information. SRP Chapter 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 8), provides guidance for determining the technical adequacy of PRA results for risk-informed activities.

RG 1.174 and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" (Reference 9), provide specific guidance and acceptance guidelines for assessing the impact of licensing basis changes, including proposed permanent TS changes.

The NRC staff considered the above guidance in assessing the proposed TS changes, but did not perform an in-depth review of every item since TR WCAP-15981-NP was not risk-informed.

3.0 TECHNICAL EVALUATION

3.1 Description of the Proposed Change

PAM instrumentation provides information required by the control room operators during accident situations to (1) provide information required to permit the control room operators to take preplanned manual actions to accomplish safe plant shutdown; (2) determine whether the reactor trip, engineered safety features systems, and manually initiated safety systems, and other systems important to safety are performing their intended functions; and (3) provide information that enable the control room operators to determine the potential for causing a gross breach of the barriers to reactivity release and to determine if a gross breach of a barrier has occurred.

TS 3.3.3 of the Standard TSs for Westinghouse plants (NUREG-1431 [Reference 10]), contains a generic list of PAM instruments for Westinghouse NSSS plants, and also contains a reviewer's note that states that a plant should include all of its plant-specific RG 1.97 Type A instrumentation and Category 1 instrumentation in the PAM TS. The generic list of PAM instrumentation was developed in the late 1980's based on DBA requirements and generic insights from PRAs available at that time. TR WCAP-15981-NP states that the associated changes to NUREG-1431 (that have been reviewed by the NRC staff in TR WCAP-15981-NP) will be included in a Technical Specifications Task Force (TSTF) Traveler for submission to the NRC at a later date. It was the NRC staff's initial understanding that the TSTF would revise the generic list of PAM instrumentation on Table 3.3.3-1 of TS 3.3.3, "PAM Instrumentation," in NUREG-1431. The NRC staff believes that a revised list of PAM instrumentation on Table 3.3.3-1 of TS 3.3.3 could be misinterpreted by plants that do not plan to apply TR-WCAP-15981-NP. During a September 20, 2007, NRC public meeting (Reference 15), the PWROG clarified that instead of revising the generic list of PAM instrumentation on Table 3.3.3-1 of TS 3.3.3, that an additional Reviewer's Note to TS 3.3.3 would be added that would provide licensees with an option of revising their plant-specific PAM table to reflect the PAM instrumentation that satisfies Criteria 3 and/or 4 of 10 CFR 50.36 based on the methodology contained in TR WCAP-15981. The NRC staff is in agreement with this approach.

The PAM instrumentation that is currently included in TS 3.3.3 was selected based on application of the following criteria contained in 10 CFR Part 50.36(c)(2)(ii):

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
- Criterion 3: A SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
- Criterion 4: A SSC which operating experience or PRA has shown to be significant to public health and safety.

The fourth criterion was added to 10 CFR 50.36 in 1995, to reflect the insights gained from PRA. Such insights were not widely known or available at the time when TS 3.3.3 was issued in Revision 0 of NUREG-1431.

TR WCAP-15981-NP provides a methodology to be used by licensees to reassess the PAM instrumentation that should be included in the plant-specific PAM TS. The plant-specific implementation of the TR WCAP-15981-NP methodology requires a plant-specific evaluation of the instrumentation assumed or credited in the plant's: (1) DBAs, (2) EOPs, (3) PRA, (4) SAMGs, and (5) EP implementing procedures. It is noted in TR WCAP-15981-NP that the purpose of the PAM instrumentation is to provide the necessary indications in a post accident environment. Thus, in evaluating instrumentation for retention in the PAM TS, the methodology in TR WCAP-15981-NP focuses on instrumentation that satisfies Criteria 3 or 4 of 10 CFR 50.36(c)(2)(ii).

The NRC staff position on which RG 1.97 variables should be included in the PAM TS, as stated in a May 1988 letter from T. E. Murley to W. S. Wilgus (Reference 11), has been that all Type A variables and all non-Type A Category 1 variables should be included in the PAM TS.

TR WCAP-15981-NP provides recommended changes to the list of variables that should be included in the PAM TS. The basis for these changes is the PWROG's position that variables that satisfy 10 CFR 50.36(c)(2)(ii) Criterion 3 or 4 should be included in the PAM TS. The TR WCAP-15981-NP recommended that instrumentation that satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) should be classified as Type A instrumentation and instrumentation that satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) should be classified as Category 1 instrumentation. The TR WCAP-15981-NP also recommended that instrumentation that do not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) but are currently in PAM TS could be downgraded in category and relocated from the PAM TS to licensee controlled documents.

The PWROG performed an analysis in TR WCAP-15981-NP to reevaluate the PAM variables against Criterion 3 and 4 of 10 CFR 50.36(c)(2)(ii) based on how each variable is used in accident management at Westinghouse NSSS plants. Based on the results of that analysis, TR WCAP-15981-NP recommends type and/or category changes for several variables. In some cases, this includes a change to the category of a variable, a change to the type of a variable, or a change to both the type and category of a variable.

TR WCAP-15981-NP proposes that the following variables be included in the PAM TS:

- Neutron Flux (Power Range)
- Reactor Coolant System (RCS) Pressure
- Core Exit Temperature
- High Head Safety Injection Flow
- Refueling Water Storage Tank Level
- Containment Pressure
- Containment Isolation Valve Position
- Pressurizer Level
- Steam Generator Level (Wide Range)
- Steam Generator Pressure
- Auxiliary Feedwater Flow
- Containment Area Radiation (High Range)

TR WCAP-15981-NP proposes that the following variables be relocated from the PAM TS:

- Neutron Flux (Source Range)
- RCS Hot-Leg Temperature
- RCS Cold-Leg Temperature
- Reactor Vessel Water Level
- Subcooling Margin
- Containment Sump Water Level (Wide Range)
- Condensate Storage Tank Level

TR WCAP-15981-NP also proposes that the following other PAM variables, which are not usually included in the PAM TS (but which may be included in the PAM TSs of other Westinghouse NSSS plants that have not converted to NUREG-1431) should be relocated from the PAM TS:

- Containment Sump Water Level (Narrow Range)
- Containment Hydrogen
- Pressurizer Pressure
- RCS Radiation Level
- Steam Generator Level (Narrow Range)
- Pressurizer Power Operated Relief Valve (PORV) Position
- Pressurizer PORV Block Valve Position
- Pressurizer Safety Valve Position
- Radiation Effluent
- Auxiliary Feedwater Valve Position
- Boric Acid Tank Level
- Containment Enclosure Negative Pressure
- Residual Heat Removal Flow
- Spray Additive Tank Level
- Component Cooling Water Temperature
- Component Cooling Water Flow
- Service Water Temperature
- Service Water Flow

3.2 Deterministic Evaluation of the Proposed Changes

Sections 3.2.1 to 3.2.22 provide the NRC staff's evaluation of the proposed changes for each parameter identified in Section 3.1.

3.2.1 Neutron Flux (Power Range)

RG 1.97 recommends that Type B Category 1 instrumentation be provided to monitor Neutron Flux from 10^{-6} to 100 percent full power to provide function detection and accomplishment of mitigation of the Reactivity Control function. TR WCAP-15981-NP recommends that the Neutron Flux (Power Range) portion of the RG 1.97 recommended range remain as a Type B Category 1 variable. Therefore, TR WCAP-15981-NP concluded that Neutron Flux (Power Range) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

3.2.2 Neutron Flux (Source Range)

RG 1.97 recommends that Type B Category 1 instrumentation be provided to monitor Neutron Flux from 10^{-6} to 100 percent full power to provide function detection and accomplishment of mitigation of the Reactivity Control function. TR WCAP-15981-NP recommends that the Neutron Flux (Source Range) portion of the RG 1.97 recommended range be reclassified as Type B Category 3 and be relocated from the PAM TS. The justification provided in TR WCAP-15981-NP is that Neutron Flux (Power Range) provides the most direct indication of the accomplishment of the Reactivity Control function. Neutron Flux (Source Range) provides verification of the automatic actuation of the RPS and is therefore a Type B variable and provides diagnostics of continued subcriticality during RCS cooldown and depressurization and should be reclassified as a Category 3 variable. Therefore, TR WCAP-15981-NP concluded that Neutron Flux (Source Range) does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and does not need to be included in the PAM TS.

However, TR WCAP-15981-NP does not discuss instrumentation to be used to provide an early indication of a return to criticality. Neutron Flux (Source Range) instrumentation provides this information. In a letter dated August 22, 2007 (Reference 14), the PWROG provided additional clarification to support the RAI responses documented in a letter dated June 28, 2007. The letter dated August 22, 2007 (Reference 14), states that RCS Boron concentration provides information to ensure adequate shutdown margin. However, TR WCAP-15981-NP has not proposed that RCS Boron Concentration be upgraded to a Category 1 variable in lieu of Neutron Flux (Source Range). Based on the information provided, the NRC staff does not agree with the proposed reclassification of Neutron Flux (Source Range) and concludes that Neutron Flux (Source Range) should be included in the PAM TS.

3.2.3 RCS Hot-Leg Water Temperature

RG 1.97 recommends that Type B Category 1 instrumentation be provided to monitor RCS Hot-Leg Water Temperature to provide function detection, accomplishment of mitigation, verification, and long-term surveillance of the Core Cooling function. TR WCAP-15981-NP recommends that RCS Hot-Leg Water Temperature be reclassified as Type B Category 3 and be relocated from the PAM TS. The justification provided in TR WCAP-15981-NP is that Core Exit Temperature provides the most direct indication of the accomplishment of the Core Cooling function. RCS Hot-Leg Water Temperature provides confirmatory information to indicate whether the Core Cooling function is being accomplished and is, therefore, a Type B variable and provides backup diagnostics to the Core Exit Temperature and High Head Safety Injection Flow indications and should be reclassified as a Category 3 variable. Therefore, TR WCAP-15981-NP concluded that RCS Hot-Leg Water Temperature does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and does not need to be included in the PAM TS.

Because High Head Safety Injection Flow is a second key variable for the accomplishment of the Core Cooling function and is included in the PAM TS (see Section 3.2.9), the NRC staff agrees that RCS Hot-Leg Water Temperature can be reclassified as a Type B Category 3 variable and does not need to be included in the PAM.

3.2.4 RCS Cold-Leg Water Temperature

RG 1.97 recommends that Type B Category 1 instrumentation be provided to monitor RCS Cold-Leg Water Temperature to provide function detection, accomplishment of mitigation, verification, and long-term surveillance of the Core Cooling function. TR WCAP-15981-NP recommends that RCS Cold-Leg Water Temperature be reclassified as Type B Category 3 and be relocated from the PAM TS. The justification provided in TR WCAP-15981-NP is that Core Exit Temperature provides the most direct indication of the accomplishment of the Core Cooling function. RCS Cold-Leg Water Temperature provides confirmatory information to indicate whether the Core Cooling function is being accomplished and is, therefore, a Type B variable and provides backup diagnostics to the Core Exit Temperature and High Head Safety Injection Flow indications and should be reclassified as a Category 3 variable. Therefore, TR WCAP-15981-NP concluded that RCS Cold-Leg Water Temperature does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and does not need to be included in the PAM TS.

The information provided by the PWROG during a September 20, 2007, NRC public meeting (Reference 15), indicated that the RCS cold-leg water temperature is used in the critical safety function (CSF) status trees of the Emergency Response Guidelines (ERG) for Westinghouse NSSS manufactured plants to direct the operators to a function restoration guideline, FR-P.1, which provides actions to avoid or limit pressurized thermal shock (PTS) to the reactor vessel. The generic probabilistic risk assessment (PRA) information provided in Appendix A to TR WCAP-15981-NP does not show that the PTS, and thus, the RCS cold-leg water temperature indication used by the operators to avoid the PTS, are risk significant. Therefore, the NRC staff agrees with the PWROG that based upon the generic assessment, the RCS cold-leg water temperature does not satisfy Criterion 4 of 10 CFR 50.36(c)(2) (ii) and need not be included in PAM TS. However, licensees will need to confirm whether this instrument should be retained in the plant-specific TS using the methodology in TR WCAP-15981-NP.

3.2.5 RCS Pressure

RG 1.97 recommends that Type B Category 1 instrumentation be provided to monitor RCS Pressure to provide function detection and accomplishment of mitigation, verification, and long-term surveillance of the Core Cooling function. TR WCAP-15981-NP recommends that RCS Pressure remains as a Type B Category 1 variable and also be classified as a Type A variable for the Core Cooling function. Therefore, TR WCAP-15981-NP concluded that RCS Pressure satisfies Criterion 3 and 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

RG 1.97 recommends that Type B Category 1 instrumentation be provided to monitor RCS Pressure to provide function detection and accomplishment of mitigation of the Maintaining RCS Integrity function. TR WCAP-15981-NP recommends that RCS Pressure remains as a Type B Category 1 variable. Therefore, TR WCAP-15981-NP concluded that RCS Pressure satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

RG 1.97 recommends that Type C Category 1 instrumentation be provided to monitor RCS Pressure to provide detection of potential for or actual breach, accomplishment of mitigation, and long-term surveillance of the Reactor Coolant Pressure Boundary function. TR WCAP-15981-NP recommends that RCS Pressure remains as a Type C Category 1 variable. Therefore, TR WCAP-15981-NP concluded that RCS Pressure satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

RG 1.97 recommends that Type C Category 1 instrumentation be provided to monitor RCS Pressure to provide detection of potential for breach and accomplishment of mitigation of the Containment function. The letter dated August 22, 2007 (Reference 14), provides justification that RCS Pressure above the normal operating pressure is not a threat to containment integrity and, therefore, RCS Pressure is not considered to be an indicator of a potential breach and accomplishment of mitigation of the Containment function. The NRC staff agrees with this conclusion. Therefore, RCS Pressure does not need to be considered a Type C Category 1 variable for the Containment function and does not need to be included in the PAM TS for the Containment function.

TR WCAP-15981-NP recommends that RCS Pressure also be classified as a Type D Category 1 key variable to monitor Primary Coolant System status. The letter dated August 22, 2007 (Reference 14), provides justification that the RCS Pressure used in conjunction with Core Exit Temperature provides indication that there are no large breaches of the reactor coolant pressure boundary and that the primary system function of removing decay heat to the steam generators is available. The NRC staff agrees with this conclusion. Therefore, RCS Pressure can be considered a Type D Category 1 key variable for Primary Coolant System status and should be included in the PAM TS for Primary Coolant System status.

3.2.6 Core Exit Temperature

RG 1.97 recommends that Type B Category 1 instrumentation be provided to monitor Core Exit Temperature to provide verification of the Core Cooling function. TR WCAP-15981-NP recommends that the Core Exit Temperature remains as a Type B Category 1 variable and also be classified as a Type A variable for the Core Cooling function. Therefore, TR WCAP-15981-NP concluded that Core Exit Temperature satisfies Criterion 3 and 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

RG 1.97 recommends that Type C Category 1 instrumentation be provided to monitor Core Exit Temperature to provide detection of potential for breach, accomplishment of mitigation, and long-term surveillance of the Fuel Cladding function. TR WCAP-15981-NP recommends that Core Exit Temperature remains as a Type C Category 1 variable. Therefore, TR WCAP-15981-NP concluded that Core Exit Temperature satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

TR WCAP-15981-NP recommends that Core Exit Temperature also be classified as a Type B Category 1 key variable for the Maintaining RCS Integrity function. The letter dated August 22, 2007 (Reference 14), provides justification that Core Exit Temperature in conjunction with RCS Pressure can identify when pressurized thermal shock conditions are being approached and would be an indicator for the Maintaining RCS Integrity function. The NRC staff agrees with this conclusion. Therefore, Core Exit Temperature can be considered a Type B Category 1 key variable for the Maintaining RCS Integrity function and should be included in the PAM TS for the Maintaining RCS Integrity function.

TR WCAP-15981-NP recommends that Core Exit Temperature also be classified as a Type C Category 1 key variable for the Reactor Coolant Pressure Boundary function. The letter dated August 22, 2007 (Reference 14), provides justification that the Core Exit Temperature used in conjunction with RCS Pressure provides indication that there are no large breaches of the reactor coolant pressure boundary and that the primary system function of removing decay heat to the steam generators is available. The NRC staff agrees with this conclusion. Therefore, Core Exit Temperature can be considered a Type C Category 1 key variable for the Reactor Coolant Pressure Boundary function and should be included in the PAM TS for the Reactor Coolant Pressure Boundary function.

TR WCAP-15981-NP recommends that Core Exit Temperature also be classified as a Type D Category 1 key variable to monitor Primary Coolant System status. The letter dated August 22, 2007 (Reference 14), provides justification that the Core Exit Temperature used in conjunction with RCS Pressure provides indication that there are no large breaches of the reactor coolant pressure boundary and that the primary system function of removing decay heat to the steam generators is available. The NRC staff agrees with this conclusion. Therefore, Core Exit

Temperature can be considered a Type D Category 1 key variable for the Primary Coolant System status.

3.2.7 Reactor Vessel Water Level

RG 1.97 recommends that Type B Category 1 instrumentation be provided to monitor Reactor Vessel Water Level (RVLIS), or Coolant Inventory, to provide verification and accomplishment of mitigation for the Core Cooling function. TR WCAP-15981-NP recommends that Reactor Vessel Water Level be reclassified as Type B Category 3 and be relocated from the PAM TS. The justification provided in TR WCAP-15981-NP is that Core Exit Temperature provides the most direct indication of the accomplishment of the Core Cooling function. Reactor Vessel Water Level provides information to indicate whether the Core Cooling function is being accomplished and is, therefore, a Type B variable and provides backup diagnostics to the Core Exit Temperature indication and should be reclassified as a Category 3 variable. Therefore, TR WCAP-15981-NP concluded that Reactor Vessel Water Level does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and does not need to be included in the PAM TS.

The information provided by the PWROG during a September 20, 2007, NRC public meeting (Reference 15) indicated that the RVLIS is used in the CSF status trees of the ERG for Westinghouse NSSS manufactured plants to direct the operators to the function restoration guidelines, including (1) FR-C.1 and FR-C.2 that provide actions to restore an adequate core cooling, and (2) FR-I.1 through FR-I.3 that provide actions to restore RCS inventory. The generic PRA information provided in Appendix A to TR WCAP-15981-NP does not show that the RVLIS used in FR-C.1 and FR-C.2 for core cooling and FR-I.1 through FR-I.3 for RCS inventory restoration is risk significant. Therefore, the NRC staff agrees with the PWROG that based upon the generic assessment, the RVLIS does not satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii) and need not be included in PAM TS. However, licensees will need to confirm whether this instrument should be retained in the plant-specific TS using the methodology in TR WCAP-15981-NP.

3.2.8 RCS Subcooling

RG 1.97 recommends that Type B Category 2 instrumentation be provided to monitor RCS Subcooling or Degrees of Subcooling to provide verification and analysis of plant conditions of the Core Cooling function. TR WCAP-15981-NP recommends that RCS Subcooling be reclassified as Type B Category 3 and be relocated from the PAM TS. The justification provided in TR WCAP-15981-NP is that Core Exit Temperature and RCS Pressure are inputs to RCS Subcooling and provide the most direct indication of the accomplishment of the Core Cooling function. RCS Subcooling provides information to indicate whether the Core Cooling function is being accomplished. Therefore, it is a Type B variable and is a backup to Core Exit Temperature and RCS Pressure and should be reclassified as a Category 3 variable. Therefore, TR WCAP-15981-NP concluded that RCS Subcooling does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and does not need to be included in the PAM TS. The NRC staff agrees with the conclusion that RCS Subcooling is not a key variable for the Core Cooling function and may be relocated from the PAM TS.

The letter dated August 22, 2007 (Reference 14) discusses the Core Exit Temperature and RCS Pressure instrumentation that are inputs to the RCS Subcooling indication. However, this discussion only provides justification as to why RCS Subcooling does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and, therefore, does not need to be classified as Category 1 and does not need to be included in the PAM TS. This discussion did not provide justification for reclassifying RCS Subcooling from Type B Category 2 to Type B Category 3. Therefore, the NRC staff does not have sufficient information to make a determination on the reclassification of RCS Subcooling. RCS Subcooling should remain classified as a Type B Category 2 variable.

3.2.9 High Head Safety Injection Flow

RG 1.97 recommends that Type D Category 2 instrumentation be provided to monitor High Head Safety Injection Flow or Flow in High Pressure Injection System to monitor the operation of the Safety Injection Systems. TR WCAP-15981-NP recommends that High Head Safety Injection Flow also be classified as a Type B Category 1 key variable for the Core Cooling function as it provides information for the verification of automatic actuation of safety injection and direct information to verify the operation of safety injection to maintain the inventory for the Core Cooling function. TR WCAP-15981-NP concluded that High Head Safety Injection Flow satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

3.2.10 Refueling Water Storage Tank Level

RG 1.97 recommends that Type D Category 2 instrumentation be provided to monitor Refueling Water Storage Tank Level to monitor the operation of the Safety Injection Systems. TR WCAP-15981-NP recommends that Refueling Water Storage Tank Level also be classified as a Type D Category 1 variable because it provides information on the accomplishment of Safety Injection System function. For plants with manual switchover to Emergency Core Cooling System (ECCS) recirculation, Refueling Water Storage Tank Level should be classified as a Type A variable. TR WCAP-15981-NP concluded that Refueling Water Storage Tank Level satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. TR WCAP-15981-NP also concluded that for plants with manual switchover to ECCS recirculation Refueling Water Storage Tank Level satisfies Criterion 3 of 10 CFR 50.35(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

TR WCAP-15981-NP recommends that Refueling Water Storage Tank Level also be classified as a Type B Category 1 key variable for the Core Cooling function. TR WCAP-15981-NP concluded that Refueling Water Storage Tank Level satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. However, TR WCAP-15981-NP does not discuss how Refueling Storage Tank Level instrumentation provides information concerning the Core Cooling function. Therefore, the NRC staff is unable to determine the applicability of Refueling Water Storage Tank Level to the Core Cooling function.

3.2.11 Containment Sump Water Level (Wide Range)

RG 1.97 recommends that Type B Category 1 instrumentation be provided to monitor Containment Sump Water Level (Wide Range) or Containment Water Level to provide function detection, accomplishment of mitigation, and verification of the Maintaining RCS Integrity function. TR WCAP-15981-NP recommends that Containment Sump Water Level (Wide Range) be reclassified as Type B Category 3 and be relocated from the PAM TS. The justification provided in TR WCAP-15981-NP is that Containment Sump Water Level (Wide Range) indication provides backup information to other key indicators for identifying the accomplishment of the Maintaining RCS Integrity function and is a Type B Category 3 variable. Therefore, TR WCAP-15981-NP concludes that Containment Sump Water Level (Wide Range) indication is a Type B variable and provides information on the status of ECCS recirculation flow delivery and should be reclassified as a Category 3 variable. Therefore, TR WCAP-15981-NP concluded that Containment Sump Water Level (Wide Range) does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and does not need to be included in the PAM TS.

The information provided in the letter dated August 22, 2007 (Reference 14), does not satisfactorily demonstrate that the Containment Sump Water Level (Wide Range) instrumentation used in the CSF status trees of the ERGs for Westinghouse NSSS plants does not meet Criterion 4 of 10 CFR 50.36(c)(2)(ii). Based on the information provided, the NRC staff does not agree with the proposed reclassification of Containment Sump Water Level (Wide Range) and concludes that Containment Sump Water Level (Wide Range) should be included in the PAM TS for the Maintaining RCS Integrity function.

RG 1.97 also recommends that Type C Category 1 instrumentation be provided to monitor Containment Sump Water Level (Wide Range) to provide detection of breach, accomplishment of mitigation, verification, and long-term surveillance of the Reactor Coolant Pressure Boundary function. TR WCAP-15981-NP recommends that Containment Sump Water Level (Wide Range) be reclassified as Type C Category 3 and be relocated from the PAM TS. The justification provided in TR WCAP-15981-NP is that degradation of the RCS Pressure Boundary can more appropriately be indicated by RCS Pressure, Pressurizer Level, and Steam Generator Level (Wide Range). While 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. Therefore, TR WCAP-15981-NP concludes that Containment Sump Water Level (Wide Range) indication is a Type C variable and provides backup information to other primary indicators for identifying the accomplishment of the Reactor Coolant Pressure Integrity function and should be reclassified as a Category 3 variable. Therefore, TR WCAP-15981-NP concluded that Containment Sump Water Level (Wide Range) does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and does not need to be included in the PAM TS.

The information provided by the PWROG during a September 20, 2007, NRC public meeting (Reference 15) indicated that the containment sump water level (wide range) is used in the CSF status trees of the ERG for Westinghouse NSSS manufactured plants to direct the operators to a function restoration guideline, FR-Z.2, which provides actions for the operators to respond to containment flooding. The generic PRA information provided in Appendix A to TR WCAP-15981-NP does not show that prevention of the containment flooding, and thus, the containment sump water level indication used by the operators to prevent containment flooding from occurring are risk significant. Therefore, the NRC staff agrees with the PWROG that based on the generic assessment, the containment sump water level (wide range) does not satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii) and need not be included in PAM TS for the Reactor Coolant Pressure Boundary function. However, licensees will need to confirm whether this instrument should be retained in the plant-specific TS using the methodology in TR WCAP-15981-NP.

3.2.12 Containment Pressure

RG 1.97 recommends that Type B Category 1 instrumentation be provided to monitor Containment Pressure to provide function detection, accomplishment of mitigation, and verification of the Maintaining RCS Integrity function. TR WCAP-15981-NP recommends that RCS Pressure remains as a Type B Category 1 variable. Therefore, TR WCAP-15981-NP concluded that Containment Pressure satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

RG 1.97 recommends that Type B Category 1 instrumentation be provided to monitor Containment Pressure to provide function detection, accomplishment of mitigation, and verification of the Maintaining Containment Integrity function. TR WCAP-15981-NP recommends that Containment Pressure remains as a Type B Category 1 variable. Therefore, TR WCAP-15981-NP concluded that Containment Pressure satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

RG 1.97 recommends that Type C Category 1 instrumentation be provided to monitor Containment Pressure to provide detection of breach, accomplishment of mitigation, verification, and long-term surveillance of the Reactor Coolant Pressure Boundary function. TR WCAP-15981-NP recommends that Containment Pressure remains as a Type C Category 1 variable. Therefore, TR WCAP-15981-NP concluded that Containment Pressure satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

RG 1.97 recommends that Type C Category 1 instrumentation be provided to monitor Containment Pressure to provide detection of potential for or actual breach and accomplishment of mitigation of the Containment function. TR WCAP-15981-NP recommends that Containment Pressure remains as a Type C Category 1 variable. Therefore, TR WCAP-15981-NP concluded that Containment Pressure satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

3.2.13 Containment Isolation Valve Position

RG 1.97 recommends that Type B Category 1 instrumentation be provided to monitor Containment Isolation Valve Position to provide function detection and accomplishment of mitigation of the Maintaining Containment Integrity function. TR WCAP-15981-NP recommends that Containment Isolation Valve Position remains as a Type B Category 1 variable. Therefore, TR WCAP-15981-NP concluded that Containment Isolation Valve Position satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

3.2.14 Pressurizer Level

RG 1.97 recommends that Type D Category 1 instrumentation be provided to monitor Pressurizer Level to ensure proper operation of the pressurizer in the Primary Coolant System. TR WCAP-15981-NP recommends that Pressurizer Level remains as a Type D Category 1 variable and also be classified as a Type A variable to provide information to permit the operator to take actions to terminate safety injection. Therefore, TR WCAP-15981-NP concluded that Containment Pressure satisfies Criterion 3 and 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

TR WCAP-15981-NP recommends that Pressurizer Level also be classified as a Type C Category 1 key variable for the Reactor Coolant Pressure Boundary function. The letter dated August 22, 2007 (Reference 14), provides justification that Pressurizer Level provides indication for termination of safety injection for secondary side breaks and other non-LOCA events that result in automatic start of safety injection. The NRC staff agrees with this conclusion. Therefore, Pressurizer Level can be considered a Type C Category 1 key variable for the Reactor Coolant Pressure Boundary function and should be included in the PAM TS for the Reactor Coolant Pressure Boundary function.

3.2.15 Steam Generator Level (Wide Range)

RG 1.97 recommends that Type D Category 1 instrumentation be provided to monitor Steam Generator Level (Wide Range) to monitor operation of the Secondary System. TR WCAP-15981-NP recommends that Steam Generator Level (Wide Range) remains as a Type D Category 1 variable and also be classified as a Type A variable to provide information for operator action to maintain a heat sink. Therefore, TR WCAP-15981-NP concluded that Steam Generator Level (Wide Range) satisfies Criterion 3 and 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

TR WCAP-15981-NP recommends that Steam Generator Level (Wide Range) also be classified as a Type C Category 1 key variable for the Reactor Coolant Pressure Boundary function. The PWROG letter dated August 22, 2007 (Reference 14), provides justification that Steam Generator Level (Wide Range) provides information on the availability of a secondary side heat sink for core decay heat removal for accident sequences when RCS Pressure and Temperature are above the cut-in point for shutdown cooling using the residual heat removal system. The NRC staff agrees with this conclusion. Therefore, Steam Generator Level (Wide Range) can be considered a Type C Category 1 key variable for the Reactor Coolant Pressure Boundary function and should be included in the PAM TS for the Reactor Coolant Pressure Boundary function.

3.2.16 Steam Generator Pressure

RG 1.97 recommends that Type D Category 2 instrumentation be provided to monitor Steam Generator Pressure to monitor operation of the Secondary System. TR WCAP-15981-NP recommends that Steam Generator Pressure also be classified as a Type A Category 1 variable to provide information for operator action for steam generator tube rupture break flow termination. Therefore, TR WCAP-15981-NP concluded that Steam Generator Pressure satisfies Criterion 3 and 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

3.2.17 Auxiliary Feedwater Flow

RG 1.97 recommends that Type D Category 2 instrumentation be provided to monitor Auxiliary Feedwater Flow to monitor operation of the Auxiliary Feedwater System. TR WCAP-15981-NP recommends that Auxiliary Feedwater Flow also be classified as a Type B Category 1 variable. Auxiliary Feedwater Flow provides information on the verification of the automatic actuation of Auxiliary Feedwater and provides the direct verification of satisfying the heat sink function. Therefore, TR WCAP-15981-NP concluded that Auxiliary Feedwater Flow satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

3.2.18 Condensate Storage Tank Water Level

RG 1.97 recommends that Type D Category 1 instrumentation be provided to monitor Condensate Storage Tank Water Level to ensure water supply for the Auxiliary Feedwater System. TR WCAP-15981-NP recommends that Condensate Storage Tank Water Level be reclassified as a Type B Category 2 variable and a Type D Category 3 variable. Condensate Storage Tank Level provides information on whether the Steam Generator heat sink can be maintained from the condensate storage tank. It does not provide information on the operation of the Auxiliary Feedwater System which is provided by Auxiliary Feedwater Flow and Steam Generator Level (Wide Range). Therefore, TR WCAP-15981-NP concluded that Condensate Storage Tank Water Level does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and does not need to be included in the PAM TS. Because, Auxiliary Feedwater Flow provides the key information on the operation of the Auxiliary Feedwater System and is included in the PAM TS, the NRC staff agrees that Condensate Storage Tank Water Level can be reclassified as a Type B Category 2 and Type D Category 3 variables and does not need to be included in the PAM TS.

3.2.19 Containment Area Radiation (High Range)

RG 1.97 recommends that Type E Category 1 instrumentation be provided to monitor Containment Area Radiation (High Range) for detection of significant releases, release assessment, long-term surveillance, and emergency plan actuation for Containment Radiation. TR WCAP-15981-NP recommends that Containment Area Radiation (High Range) remains as a Type E Category 1 variable. Therefore, TR WCAP-15981-NP concluded that Containment Area Radiation (High Range) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

TR WCAP-15981-NP recommends that Containment Area Radiation (High Range) also be classified as a Type C Category 1 variable for the Reactor Coolant Pressure Boundary function as it provides key information to identify a fission product barrier challenge, detection of breach, and verification of the Reactor Coolant Pressure Boundary function. Therefore, TR WCAP-15981-NP concluded that Containment Area Radiation (High Range) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. The NRC staff agrees with this conclusion.

3.2.20 Other PAM Variables

TR WCAP-15981-NP provides justification for various other RG 1.97 Category 2 and Category 3 variables and non-RG 1.97 variables that do not need to be included in the PAM TS. These variables include Containment Sump Water Level (Narrow Range), Containment Hydrogen, Pressurizer Pressure, RCS Radiation Level, Steam Generator Level (Narrow Range), Pressurizer PORV Position, Pressurizer PORV Block Valve Position, Pressurizer Safety Valve Position, Radiation Effluent, Auxiliary Feedwater Valve Position, Boric Acid Tank Level, Containment Enclosure Negative Pressure, Residual Heat Removal Flow, Spray Additive Tank Level, Component Cooling Water Temperature, Component Cooling Water Flow, Service Water Temperature, and Service Water Flow. The NRC staff agrees that since these variables do not satisfy either Criterion 3 or Criterion 4 of 10 CFR 50.36(c)(2)(ii), they do not need to be included in the PAM TS. However, if for a plant-specific application one of these variables is classified as a Type A variable, that variable would satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS.

3.2.21 Proposed Alternate Instrumentation

Alternate instrumentation should meet the same RG 1.97 category as the primary instrumentation. RG 1.97 recommends two channels of Category 1 instrumentation for each Type A or Category 1 variable. TR WCAP-15981-NP recommends the use of alternate instrumentation for various PAM instrumentation.

3.2.21.1 Neutron Flux Power Range

TR WCAP-15981-NP proposed that Neutron Flux (Intermediate Range) or Neutron Flux (Source Range) and Rod Position Indication or Rod Bottom Lights be used as alternate instrumentation for Neutron Flux (Power Range). TR WCAP-15981-NP has not discussed the qualification of Neutron Flux (Intermediate Range), has proposed a down grade of Neutron Flux (Source Range), and Rod Position Indication and Rod Bottom Lights are Category 3. Therefore, based on the information provided the NRC staff does not agree with the proposed use of these alternate instrumentation on a generic basis.

3.2.21.2 High Head Safety Injection Flow

TR WCAP-15981-NP proposed that High Head Safety Injection Pump Amperage and High Head Safety Injection Pump Discharge or Header Pressure and Automatic Safety injection Valve Position be used as alternate instrumentation for High Head Safety injection Flow. TR WCAP-15981-NP has not discussed the qualification of the proposed alternate instrumentation. Therefore, based on the information provided the NRC staff does not agree with the proposed use of these alternate instrumentation on a generic basis.

3.2.21.3 Containment Area Radiation (High Range)

TR WCAP-15981-NP proposed that portable radiation instrumentation be used as alternate instrumentation in the event that both required channels of Containment Area Radiation (High Range) are unavailable. NUREG-1431 currently includes the initiation of action that requires a report that outlines the preplanned alternate method of monitoring the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the function to operable status. The selection of a preplanned alternate method of monitoring is plant specific. Therefore, the determination of the appropriateness of the use of portable radiation instrumentation should be performed on a plant specific basis.

3.2.21.4 Steam Generator Level (Wide Range)

TR WCAP-15981-NP proposed that a combination of Steam Generator Level (Narrow Range) and Auxiliary Feedwater Flow be used as alternate instrumentation for Steam Generator Level (Wide Range). The use of Steam Generator Level (Narrow Range) and/or Auxiliary Feedwater Flow as a alternate channel to Steam Generator Level (Wide Range) has been accepted previously for a limited number of plant specific applications based on the plant specific design. The use of these instruments as alternates should continue to be reviewed on a plant specific basis and should not be applied on a generic basis.

3.2.21.5 Auxiliary Feedwater Flow

TR WCAP-15981-NP proposed that (1) Auxiliary Feedwater Pump Amperage and Auxiliary Feedwater Pump Discharge Pressure or Flow Control Valve Position for motor driven auxiliary feedwater pumps and (2) Auxiliary Feedwater Pump Discharge Pressure or Steam Supply Valve Position and Flow Control Valve Position for turbine driven auxiliary feedwater pumps, as alternate instrumentation for Auxiliary Feedwater Flow. TR WCAP-15981-NP has not discussed the qualification of the proposed alternate instrumentation. Therefore, based on the information provided, the NRC staff does not agree with the proposed use of these alternate instrumentation on a generic basis.

3.2.21.6 Core Exit Temperature Channels

NUREG-1431 includes the number of required channels of Core Exit Temperature instruments as two required channels per quadrant with a channel consisting of two Core Exit Thermocouples. TR WCAP-15981-NP proposed that the number of required channels be changed to two with a limitation on which Core Exit Thermocouples should be excluded from being included in the TS. As part of this recommendation, TR WCAP-15981-NP referenced TR WCAP-14696-A (Reference 16). However, TR WCAP-15981-NP and TR WCAP-14696-A did not discuss the quadrants or how many channels should be required per quadrant. Therefore, based on the information provided, the NRC staff does not agree with the proposed change for the number of required channels for Core Exit Temperature in NUREG-1431.

3.2.22 Summary of the Deterministic Evaluation

The NRC staff agrees with the TR WCAP-15981-NP recommendation that the following variables should be included in the PAM TS for the functions indicated:

Variable	Function	Type/ Category
Neutron Flux (Power Range)	Reactivity Control	B1
RCS Pressure	Core Cooling	A1, B1
RCS Pressure	Maintaining RCS Integrity	B1
RCS Pressure	Reactor Coolant Pressure Boundary	C1

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Variable	Function	Type/ Category
RCS Pressure	Primary Coolant System Status	D1
Core Exit Temperature	Core Cooling	A1, B1
Core Exit Temperature	Fuel Cladding	C1
Core Exit Temperature	Maintaining RCS Integrity	B1
Core Exit Temperature	Reactor Coolant Pressure Boundary	C1
Core Exit Temperature	Primary Coolant System Status	D1
High Head Safety Injection Flow	Core Cooling	B1
Refueling Water Storage Tank Level	Safety Injection Systems Status	A1,D2
Containment Pressure	Maintaining RCS Integrity	B1
Containment Pressure	Maintaining Containment Integrity	B1
Containment Pressure	Reactor Coolant Pressure Boundary	C1
Containment Pressure	Containment	C1
Containment Isolation Valve Position	Maintaining Containment Integrity	B1
Pressurizer Level	Primary Coolant System Status	A1, D1
Pressurizer Level	Reactor Coolant Pressure Boundary	C1
Steam Generator Level (Wide Range)	Secondary System Status	A1, D1
Steam Generator Level (Wide Range)	Reactor Coolant Pressure Boundary	C1
Steam Generator Pressure	Secondary System Status	A1, D2
Auxiliary Feedwater Flow	Auxiliary Feedwater System Status	B1, D2
Containment Area Radiation (High Range)	Reactor Coolant Pressure Boundary	C1
Containment Area Radiation (High Range)	Containment Radiation	E1

The NRC staff agrees with the TR WCAP-15981-NP recommendation that the following variables can be relocated from the PAM TS for the functions indicated:

Variable	Function	Type/ Category
RCS Hot-Leg Water Temperature	Core Cooling	B3
RCS Cold-Leg Water Temperature	Core Cooling	B3
RCS Pressure	Containment	N/A
Reactor Vessel Water Level	Core Cooling	B3
Containment Sump Water Level	Reactor Coolant Pressure Boundary	C3
RCS Subcooling	Core Cooling	B2
Condensate Storage Tank Level	Auxiliary Feedwater System Status	B2, D3
Other PAM Variables	Various	Various

The NRC staff does not agree with the TR WCAP-15981-NP recommendation that the following variables can be relocated from the PAM TS for the functions indicated:

Variable	Function	Type/ Category
Neutron Flux (Source Range)	Reactivity Control	B1
Containment Sump Water Level	Maintaining RCS Integrity	B1

The NRC staff was unable to determine the applicability of the following variable for the function indicated and, therefore, does not agree with the TR WCAP-15981-NP recommendation that the following variable should be included in the PAM TS for the function indicated:

Variable	Function	Type/ Category
Refueling Water Storage Tank Level	Core Cooling	A1, D2

Attachment 1 to this SE provides a detailed list of each RG 1.97 variable and TR WCAP-15981-NP proposed changes that have been reviewed and accepted by the NRC staff in Section 3.2 of this SE.

3.3 Evaluation of the Proposed PRA Changes

TR WCAP-15981-NP was not submitted as a risk-informed application pursuant to RG 1.174, but uses PRA information as one element of the overall method to determine the instrumentation to be included in the PAM TS. Instrumentation associated with DBA response, as well as implementation of EOPs, SAMGs, and the plant's EP is also considered. Therefore, the methodology is potentially more prescriptive than a risk-informed approach.

3.3.1 Review Methodology

The NRC staff notes that the methodology provides a basis for assessing which instrumentation should be retained within the plant-specific PAM TS, and which instrumentation could be removed from the PAM TS. The methodology establishes a clear expectation that any instrumentation removed from the PAM TS would be relocated to licensee controlled documents. The methodology does not address or propose removal of such instrumentation from the plant. This constraint provides additional assurance that the risk implications of the methodology would be minimal, and that adequate protection would not be called into question as a result of implementation. Accordingly, treatment of TR WCAP-15981-NP as a non-risk-informed application is reasonable.

The NRC staff undertook a review of the methodology in order to provide perspectives into how PRA and other information would be used collectively to identify instrumentation for inclusion in the PAM TS. The NRC staff considered the guidance and key safety principles discussed in Section 2.3 of this SE for assessing the impact of proposed risk-informed changes, including proposed permanent TS changes, but did not perform an in-depth review of every item since TR WCAP-15981-NP is not risk-informed.

3.3.2 NRC Evaluation

The overall process to be used by licensees to identify the instrumentation to be included in the PAM TS is described in Section 8 and Table 14 of TR WCAP-15981-NP. The process requires a plant-specific determination of the plant parameters that are the basis for important operator actions to bring the plant to a safe stable state following an accident. This involves an evaluation of operator actions assumed or credited in the plant's DBA, EOPs, PRA, SAMG, and EP implementing procedures. Screening criteria for identifying operator actions and supporting instrumentation in each of these areas are provided in Section 3.2 and Table 5 of the TR. Instrumentation that does not satisfy Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) can be relocated to a licensee controlled document, following a focused evaluation to confirm the adequacy of the PRA and human reliability analysis (HRA) with regard to the treatment of the operator actions associated with that instrumentation.

Implementation is carried out through a 6-step process, as itemized below:

1. Identification of operator actions in the DBA analyses.
2. Verification of PRA technical adequacy.
3. Identification of important operator actions identified in the risk assessments.
4. Identification of important operator actions identified in accident management.
5. Identification of variables and associated instrumentation for the important operator actions identified in Steps 1, 3, and 4.

6. Identification of instrumentation to be included in or relocated from the PAM TS.

The focus of the NRC staff's review was on the process and guidance for: verification of PRA technical adequacy (Step 2), the use of the PRA to identify important operator actions (Step 3), and the focused evaluation of the adequacy of the HRA treatment of operator actions associated with any instrumentation to be relocated from the PAM TS (Step 6).

3.3.2.1 Verification of PRA Technical Adequacy

TR WCAP-15981-NP states that the licensee should ensure that the internal events PRA is technically adequate for this application, but that only a limited assessment is required since this is not a risk-informed application. The TR states that the assessment of PRA technical adequacy needs to consider the areas of the accident sequence analysis and the HRA to assure that the treatment of operator actions based on plant instrumentation is appropriate. It further states that the licensee should confirm that all operator actions potentially impacted by the subject instruments have been identified, that the treatment of these operator actions in the PRA is appropriate (including the human error probability values and dependencies), and that there are no peer review comments that can affect the conclusions regarding instrument importance. Table 14 also indicates that the licensee should confirm that the PRA reflects the as-built, as-operated design, and that any plant modifications and operational changes not reflected in the PRA do not impact the plant-specific PAM instrumentation application. As discussed in Section 3.3.2.2 below, for any PAM instrumentation that is proposed to be relocated from the PAM TS to a licensee controlled document, the process includes an additional, focused evaluation to confirm the adequacy of the HRA with regard to the treatment of the operator actions associated with that instrumentation.

The assessment of technical adequacy is limited to the internal events PRA. TR WCAP-15981-NP justifies this treatment on the basis of a review of the important operator actions from several Westinghouse NSSS plants with a fully quantified external events PRA that has shown that the important operator actions that are based on control room instrumentation in the external events PRA are the same as those already determined to be significant in the internal events PRA.

In the NRC staff's view, the guidance regarding PRA technical adequacy sufficiently addresses those aspects of the PRA most important to this application, specifically, the completeness of the PRA with regard to important operator actions, the adequacy of the HRA treatment of those actions, and the impact of any related peer review comments. The assessment of PRA technical adequacy is less than what might be expected if this were a risk-informed application in which the instrumentation to be retained within the PAM TS hinged on the use of PRA; but it is considered adequate given that the instrumentation importance in PRA is just one of several considerations in the methodology.

3.3.2.2 Use of PRA to Identify Important Operator Actions

TR WCAP-15981-NP provides a discussion of instrumentation importance in PRAs within Section 3.2 and Appendix A of the TR. Generic lists of operator actions with the highest mean Risk Achievement Worth (RAW) and Fussell-Vesely (FV) importances are provided in Appendix A based on a proprietary database of plant-specific PRA results for Westinghouse plants. A table relating specific PAM instrumentation to the important operator actions and applicable EOPs is also provided in Appendix A. The TR indicates that the instrumentation utilized for each operator action was identified by reviewing the detailed PRA models for several plants and confirming these results with an independent review of the generic Westinghouse Owners Group (WOG) Emergency Response Guidelines, upon which all of the WOG plant EOPs are based. The generic lists of important operator actions and associated instrumentation presented in TR WCAP-15981-NP are indicative of the types of operator actions and instrumentation that might be retained in the PAM TS, but would not be entirely applicable to any given plant. As described in the implementation guidance in Section 8, each licensee would need to confirm the specific instrumentation to be relocated from the plant-specific PAM TS based on a plant-specific implementation of the TR WCAP-15981-NP methodology.

TR WCAP-15981-NP states that the plant-specific RAW and FV importances are to be used to identify the risk important operator actions for both CDF and LERF. The guidance specifies that a RAW value greater than 2.0 or a FV value greater than 0.05 should be used to define the risk-important operator actions (for both CDF and LERF). Although the emphasis of the identification process is on operator actions in internal events, operator actions in external events are also considered. The guidance states that if a quantitative PRA for external events (e.g., fire and seismic initiating events) is available, the risk importance of operator actions can be identified in the same manner as in the internal events PRA. If only a qualitative external events risk assessment is available, the results of the assessment can also be used to identify important operator actions by identifying operator actions required for risk-important external events. As noted in Sections 3.1 and 8, the evaluation of external events would be limited to ensuring that instrumentation proposed to be relocated from the PAM TS is not used for important operator actions to respond to external initiating events. The requirement to verify that any instrumentation proposed for relocation from the PAM TS does not cue an operator action important to risk for external initiating events is provided in Step 6 of the implementation process (Table 14).

In the NRC staff's view, the guidance regarding the use of the PRA to identify important operator actions is reasonable. Consideration of operator actions important to CDF as well as LERF provides some assurance that both the core and containment barriers will not be significantly and adversely impacted by changes to the PAM TS. Use of both the RAW and FV importance measures provides additional confidence that the key operator actions from a risk point of view would be captured. The RAW importance metric provides a measure of the potential risk increase if instrument reliability is reduced as a result of its removal from the TS, and is the most relevant metric for preserving the existing level of safety. The FV importance metric provides a measure of the potential risk reduction if the associated operator actions are improved via training or procedure modifications, and is less relevant to this application. The specified screening criteria for the RAW metric (RAW > 2.0) is consistent with that established for individual basic events in NRC and industry guidance on risk-informing the special treatment requirements of 10 CFR Part 50 (i.e., NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," [Reference 12] which is endorsed by the NRC in RG 1.201 [Reference 13]). The NRC staff concludes that the guidance on the use of PRA to identify important operator actions is acceptable given that the instrumentation importance in PRA is just one of several considerations in the methodology.

3.3.2.3 Focused Evaluation of PRA/HRA for Instrumentation to be Relocated from the PAM TS

The final step of the implementation process (Step 6 in Table 14) is to identify the instrumentation that can be relocated from the PAM TS to licensee controlled documents. The guidance states that an instrumentation that does not satisfy Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) can be relocated to a licensee controlled document, following a focused evaluation to confirm the adequacy of the HRA with regard to the treatment of the operator actions associated with that instrumentation. The guidance also states that at this point, the external events risk assessment should be reviewed to determine that none of the instrumentation proposed to be relocated from the PAM TS supports a risk important operator action.

In concept, the preceding steps in the implementation process would provide assurance that potentially risk significant instrumentation is not removed from the PAM TS. The inclusion of this final verification, through its focus on the specific instrumentation proposed for relocation and on the treatment of the associated operator actions in the HRA, provides added assurance that risk significant instrumentation is not inadvertently relocated from the PAM TS. The NRC staff concludes that this guidance is sufficient to ensure that any instrumentation proposed to be relocated from the PAM TS will receive specific consideration for risk implications.

3.3.2.4 Conclusion Regarding Use of PRA in the Re-Definition of PAM Instrumentation

Based on the information contained in TR WCAP-15981-NP and in the RAI responses, the NRC staff concludes that:

- The guidance regarding PRA technical adequacy sufficiently addresses those aspects of the PRA most important to this application, specifically, the completeness of the PRA with regard to important operator actions, the adequacy of the PRA and HRA treatment of those actions, and the impact of any related peer review comments.

- The guidance regarding the use of the PRA to identify important operator actions is reasonable, specifically, the consideration of operator actions important to CDF as well as LERF, the use of both the RAW and FV importance measures, and the specified screening criteria for these metrics.

- The inclusion of a final verification step, through its focus on the specific instrumentation proposed for relocation and on the treatment of the associated operator actions in the HRA, provides added assurance that risk-significant instrumentation is not inadvertently relocated from the PAM TS.

Although the guidance on the use of PRA for this application may be less rigorous than what might be expected if this were a risk-informed application (in which the instrumentation to be retained within the PAM TS hinged on the use of PRA), it is considered adequate given that the instrumentation importance in PRA is just one of several considerations in the methodology, and that any instrumentation removed from the PAM TS would be relocated to licensee controlled documents and not removed from the plant. The latter constraint provides additional assurance that the risk implications of the methodology would be minimal, and that adequate protection would not be called into question as a result of implementation. Accordingly, treatment of TR WCAP-15981-NP as a non-risk-informed application is reasonable.

4.0 LIMITATIONS AND CONDITIONS

The NRC staff has placed the following conditions and limitations on use of TR WCAP-15981-NP:

- Licensees that submit license amendment requests (LARs) based on TR WCAP-15981-NP must confirm the applicability of this TR to their plant, complete all parts of the stated methodology, and provide the information identified in Section 4.1 below.

- TR WCAP-15981-NP provides justification for various Other RG 1.97 Category 2 and Category 3 variables and non-RG 1.97 variables that do not need to be included in the PAM TS. The NRC staff agrees that since these variables (as listed in Section 3.2.20 of this SE) do not satisfy either Criterion 3 or Criterion 4 of 10 CFR 50.36(c)(2)(ii), they do not need to be included in the PAM TS. However, if for a plant-specific application one of these variables is classified as a Type A variable, that variable would satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS.

- As discussed in Section 3.2.21 of this SE, TR WCAP-15981-NP recommends the use of generically applicable alternate instrumentation for various PAM instrumentation. The NRC staff does not agree with the proposed use of alternate instrumentation on a generic basis. The use of instruments as alternates should continue to be reviewed on a plant-specific basis. Specifically with regard to the Core Exit Temperature Channels, the NRC staff does not agree with the proposed change for the number of required channels for Core Exit Temperature in NUREG-1431.

4.1 Plant-Specific Items to be Submitted by Licensees

Licensees that submit an LAR based on TR WCAP-15981-NP need to submit the following plant-specific information:

1. A general description of the PRA, including the scope of the analyses, PRA update history (including version peer reviewed, version(s) in which peer review comments were addressed, and version used for PAM application), and the licensee's PRA updating and quality assurance process.

2. A description of the most relevant peer reviews, a characterization of the peer review findings, a summary of the status of resolution of the peer review comments, and a listing of all unresolved facts and observations that potentially impact the application of TR WCAP-15981-NP.

3. A conclusion regarding PRA quality assessment for the PAM TS application, and verification that the quality is acceptable for the application. This should include confirmation that the PRA reflects the as-built, as-operated design, and that any recent plant modifications and operational changes not reflected in the PRA do not impact the plant-specific PAM application; all peer review comments have

been resolved or don't impact plant-specific PAM application; the PRA and HRA is sufficiently complete and applicable for evaluating the risk associated with the PAM application.

4. Listings of the important operator actions identified based on RAW and FV importance values for CDF and for LERF, along with these values.

5. Additions to the list of important operator actions based on review of results from the plant-specific external event assessments, or verification that the plant-specific risk assessments do not result in identification of additional risk-significant operator actions or variables/instruments.

6. A listing of variables/instruments related to the important operator actions. This should indicate how each variable/instrument considered in the methodology application was related to or mapped to a PRA model element or operator action.

7. Summary tables showing important indications for accident management, and the context in which they are important (e.g., DBA analysis, DBA, EOPs, SAMGs, PRA, EP (similar to Tables 7 and 8 in TR WCAP-15981-NP)).

8. A summary table describing variables/instruments added to or relocated from the technical specifications, and the specific bases for each change.

9. For any variables/instruments to be deleted from the TSs based on their lack of risk significance, the results of the focused evaluation of the adequacy of the PRA and HRA treatment (or lack of treatment) of operator actions associated with those variables/instruments.

10. For any variables/instruments to be deleted from the TSs based on their lack of risk significance, a discussion of how the reliability and availability of these instruments will be monitored and assessed (e.g., under the maintenance rule, other licensee program, or performance measurement strategy).

11. Prior to changing the treatment requirements for instrumentation that is used in emergency action level classification, licensees must consider the impacts of the changes on the effectiveness of their emergency plans.

5.0 CONCLUSION

Based on the above evaluation, the NRC staff concludes that the proposed changes described in TR WCAP-15981-NP, as modified in this SE and summarized in Section 3.2.22, are acceptable for Westinghouse NSSS plants in accordance with the limitations and conditions in Section 4.0. Licensees that submit license amendment requests (LAR) based on TR WCAP-15981-NP must confirm the applicability of this TR to their plant, complete all parts of the stated methodology, and provide the information identified in Section 4.1. For those items where the NRC staff was unable to conclude that the proposed change was acceptable, the PWROG may submit additional information as a supplement to TR WCAP-15981-NP.

6.0 REFERENCES

1. Letter from F. P. Shiffley (WOG) to US NRC, "Transmittal of WCAP-15981-NP, Post Accident Monitoring Re-Definition for Westinghouse NSSS Plants," September 17, 2004 (Accession No. ML042660254).

2. Letter from F. P. Schiffley (PWROG) to US NRC, "Responses to the NRC Request for Additional Information Regarding the Review of WCAP-15981-NP, Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," March 20, 2006 (Accession No. ML060810243).

3. Letter from F. P. Schiffley (PWROG) to US NRC, "Additional Revisions to WCAP-15981-NP, Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," August 10, 2006 (Accession No. ML062270035).

4. Letter from F. P. Schiffley (PWROG) to US NRC, "Response to Additional Requests for Information for WCAP-15981-NP, Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," June 28, 2007 (Accession No. ML071840028).

5. U.S. Nuclear Regulatory Commission,"An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, Revision 1, November 2002 (Accession No. ML023240437).

6. U.S. Nuclear Regulatory Commission, RG 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," NRC Office of Nuclear Regulatory Research, May 1983 (Accession No. ML003740282).

7. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," June 1987.

8. U.S. Nuclear Regulatory Commission, NUREG-0800, Chapter 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, June 2007.

9. U.S. Nuclear Regulatory Commission, RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," August 1998 (Accession No. ML003740176).

10. NUREG-1431, Rev. 3, "Standard Technical Specification Westinghouse Plants," dated June 2004 (Accession No. ML041830612).

11. Letter from T. E. Murley (NRC) to W. S. Wilgus (B&W Owners Group), "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," dated May 1988.

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12. NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline, July 2005 (Accession No. ML052910035).

13. U.S. Nuclear Regulatory Commission, RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According To Their Safety Significance," May 2006 (Accession No. ML061090627).

14. Letter from F. P. Schiffley (PWROG) to US NRC, "Responses to the NRC Request for Clarification of June 28, 2007, RAI Responses for WCAP-15981-NP, Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," August 22, 2007 (Accession No. ML072360096).

15. Letter from T.M. Mensah (NRC) to S.L. Rosenberg (NRC), "Summary of the September 20, 2007, Category 2 Public Meeting with the Pressurized Water Reactor Owners Group (PWROG) Concerning Topical Report (TR) WCAP-15981-NP, "Post Accident Monitoring Instrumentation Re-definition for Westinghouse NSSS [NUCLEAR STEAM SUPPLY SYSTEM] Plants," (Accession No. ML072750513).

16. Letter from Louis F. Liberatori Jr., (WOG) to US NRC, "Transmittal of Approved Topical Report: WCAP-14696-A, Rev. 1 (Non-Proprietary), "Westinghouse Owners Group Core Damage Assessment Guidance," November 1999," (Accession Nos. ML993480494 and ML993490267).

Attachments: 1. RG 1.97 Variables and WCAP-15981-NP Proposed Changes
2. Resolution of Comments

Principle Contributors: Barry Marcus
Robert Palla
Summer Sun

Date: February 18, 2008

RG 1.97 Variables and TR WCAP-15981-NP Proposed Changes								
Variable	RG 1.97 Function	RG 1.97 Purpose	Current Type and Category	Current TS Inclusion	Proposed Alternate Variable	Proposed Type and Category	Proposed TS Inclusion	Accepted By NRC
Any Type A Variable	Manual Action	No automatic control	A1	Yes	-	A1	Yes	Yes
Neutron Flux (Power Range)	Reactivity Control	Function detection, Accomplishment of mitigation	B1	Yes	-	B1	Yes	Yes
Neutron Flux (Source Range)	Reactivity Control	Function detection, Accomplishment of mitigation	B1	Yes	Neutron Flux (Power Range)	B3	No	No
RCS Hot-Leg Water Temperature	Core Cooling	Function detection, Accomplishment of mitigation, Verification, Long-term surveillance	B1	Yes	Core Exit Temperature, High Head SI	B3	No	Yes
RCS Cold-Leg Water Temperature	Core Cooling	Function detection, Accomplishment of mitigation, Verification, Long-term surveillance	B1	Yes	Core Exit Temperature	B3	No	Yes

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RG 1.97 Variables and TR WCAP-15981-NP Proposed Changes								
Variable	RG 1.97 Function	RG 1.97 Purpose	Current Type and Category	Current TS Inclusion	Proposed Alternate Variable	Proposed Type and Category	Proposed TS Inclusion	Accepted By NRC
RCS Pressure	Core Cooling	Function detection, Accomplishment of mitigation, Verification, Long-term surveillance	B1	Yes	-	A1, B1	Yes	Yes
RCS Pressure	Maintaining RCS Integrity	Function detection, Accomplishment of mitigation	B1	Yes	-	B1	Yes	Yes
RCS Pressure	Reactor Coolant Pressure Boundary	Detection of potential for or actual breach, Accomplishment of mitigation, Long-term surveillance	C1	Yes	-	C1	Yes	Yes

RG 1.97 Variables and TR WCAP-15981-NP Proposed Changes								
Variable	RG 1.97 Function	RG 1.97 Purpose	Current Type and Category	Current TS Inclusion	Proposed Alternate Variable	Proposed Type and Category	Proposed TS Inclusion	Accepted By NRC
RCS Pressure	Containment	Detection of potential for breach, Accomplishment of mitigation	C1	Yes	Containment Pressure	N/A	No	Yes
RCS Pressure	Primary Coolant System	-	-	Yes	-	D1	Yes	Yes
Core Exit Temperature	Core Cooling	Verification	B1	Yes	-	A1, B1	Yes	Yes
Core Exit Temperature	Fuel Cladding	Detection of potential for breach, Accomplishment of mitigation, Long-term surveillance	C1	Yes	-	C1	Yes	Yes
Core Exit Temperature	Maintaining RCS Integrity	-	-	Yes	-	B1	Yes	Yes

RG 1.97 Variables and TR WCAP-15981-NP Proposed Changes

Variable	RG 1.97 Function	RG 1.97 Purpose	Current Type and Category	Current TS Inclusion	Proposed Alternate Variable	Proposed Type and Category	Proposed TS Inclusion	Accepted By NRC
Core Exit Temperature	Reactor Coolant Pressure Boundary	-	-	Yes	-	C1	Yes	Yes
Core Exit Temperature	Primary Coolant System	-	-	Yes	-	D1	Yes	Yes
Reactor Vessel Water Level	Core Cooling	Verification, Accomplishment of mitigation	B1	Yes	Core Exit Temperature	B3	No	Yes
RCS Subcooling	Core Cooling	Verification and analysis of plant conditions	B2	No	-	B3	No	No
High Head Safety Injection Flow	Core Cooling	-	-	No	-	B1, D1	Yes	Yes

RG 1.97 Variables and TR WCAP-15981-NP Proposed Changes								
Variable	RG 1.97 Function	RG 1.97 Purpose	Current Type and Category	Current TS Inclusion	Proposed Alternate Variable	Proposed Type and Category	Proposed TS Inclusion	Accepted By NRC
Refueling Water Storage Tank Level	Safety Injection Systems	To monitor operation	D2	No	-	A1[*1], D1, D2	Yes	Yes
Refueling Water Storage Tank Level	Core Cooling	-	-	No	-	B1	Yes	No
Containment Sump Water Level (Wide Range)	Maintaining RCS Integrity	Function detection, Accomplishment of mitigation, Verification	B1	Yes	RCS Pressure	B2	No	No

ATTACHMENT

RG 1.97 Variables and TR WCAP-15981-NP Proposed Changes								
Variable	RG 1.97 Function	RG 1.97 Purpose	Current Type and Category	Current TS Inclusion	Proposed Alternate Variable	Proposed Type and Category	Proposed TS Inclusion	Accepted By NRC
Containment Sump Water Level (Wide Range)	Reactor Coolant Pressure Boundary	Detection of breach, Accomplishment of mitigation, Verification, Long-term-surveillance	C1	Yes	RCS Pressure, Pressurizer Level, Steam Generator Level (Wide Range)	C3	No	Yes
Containment Pressure	Maintaining RCS Integrity	Function detection, Accomplishment of mitigation, Verification	B1	Yes	-	B1	Yes	Yes
Containment Pressure	Maintaining Containment Integrity	Function detection, Accomplishment of mitigation, Verification	B1	Yes	-	B1	Yes	Yes

RG 1.97 Variables and TR WCAP-15981-NP Proposed Changes								
Variable	RG 1.97 Function	RG 1.97 Purpose	Current Type and Category	Current TS Inclusion	Proposed Alternate Variable	Proposed Type and Category	Proposed TS Inclusion	Accepted By NRC
Containment Pressure	Reactor Coolant Pressure Boundary	Detection of breach, Accomplishment of mitigation, Verification, Long-term surveillance	C1	Yes	-	C1	Yes	Yes
Containment Pressure	Containment	Detection of potential for or actual breach, Accomplishment of mitigation	C1	Yes	-	C1	Yes	Yes
Containment Isolation Valve Position	Maintaining Containment Integrity	Accomplishment of isolation	B1	Yes	-	B1	Yes	Yes

RG 1.97 Variables and TR WCAP-15981-NP Proposed Changes								
Variable	RG 1.97 Function	RG 1.97 Purpose	Current Type and Category	Current TS Inclusion	Proposed Alternate Variable	Proposed Type and Category	Proposed TS Inclusion	Accepted By NRC
Pressurizer Level	Primary Coolant System	To ensure proper operation of the pressurizer	D1	Yes	-	A1, D1	Yes	Yes
Pressurizer Level	Reactor Coolant Pressure Boundary	-	-	Yes	-	C1	Yes	Yes
Steam Generator Level (Wide Range)	Secondary System	To monitor operation	D1	Yes	-	A1, D1	Yes	Yes
Steam Generator Level (Wide Range)	Reactor Coolant Pressure Boundary	-	-	Yes	-	C1	Yes	Yes
Steam Generator Pressure	Secondary System	To monitor operation	D2	No	-	A1, D2	Yes	Yes

RG 1.97 Variables and TR WCAP-15981-NP Proposed Changes								
Variable	RG 1.97 Function	RG 1.97 Purpose	Current Type and Category	Current TS Inclusion	Proposed Alternate Variable	Proposed Type and Category	Proposed TS Inclusion	Accepted By NRC
Auxiliary Feedwater Flow	Auxiliary Feedwater System	To monitor operation	D2	Yes	-	B1, D2	Yes	Yes
Condensate Storage Tank Water Level	Auxiliary Feedwater System	To ensure water supply for auxiliary feedwater	D1	Yes	Auxiliary Feedwater Flow, Steam Generator Level (Wide Range)	B2, D3	No	Yes

RG 1.97 Variables and TR WCAP-15981-NP Proposed Changes								
Variable	RG 1.97 Function	RG 1.97 Purpose	Current Type and Category	Current TS Inclusion	Proposed Alternate Variable	Proposed Type and Category	Proposed TS Inclusion	Accepted By NRC
Containment Area Radiation (High Range)	Containment Radiation	Detection of significant releases, Release assessment, Long-term surveillance, Emergency plan actuation	E1	Yes	-	E1	Yes	Yes
Containment Area Radiation (High Range)	Reactor Coolant Pressure Boundary	-	-	No	-	C1	Yes	Yes

[1] If switchover to ECCS recirculation is based on RWST Level indication rather than the RWST level alarm, RWST Level indication should be classified as a Type A variable rather than a Type D variable.

RESOLUTION OF PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG)

COMMENTS ON DRAFT SAFETY EVALUATION (SE) FOR TOPICAL REPORT (TR)

WCAP-15981-NP, "POST ACCIDENT MONITORING INSTRUMENTATION RE-DEFINITION

FOR WESTINGHOUSE NSSS [NUCLEAR STEAM SUPPLY SYSTEM PLANTS]

(TAC NO. MC4524)

By letter dated December 21, 2007, the PWROG provided ten comments on the draft SE for TR WCAP-15981-NP. The following are the NRC staff's resolution of these comments:

1. Page 8, Section 3.2.2, Lines 4 though 8:

"However, TR WCAP-15981 ... should be included in the PAM TS."

PWROG Comment:

The response to item 4 on page 2 of Attachment 3 to OG-07-376 (Reference 1) discusses that the primary indication of a potential approach to recriticality is provided by the measurement of the reactor coolant boron concentration by sampling, as opposed to the use of instrumentation to determine the boron concentration. Additionally, the Source Range Neutron Flux reactor trip function is contained in Tech Spec 3.3.1, "RTS Instrumentation," which requires the indication to be operable when this portion of the Nuclear Instrumentation System (NIS) provides useful information (i.e., below P-6 when rod withdrawal is imminent or occurring). Additional information supporting this comment was presented during the September 20, 2007 public meeting between the PWROG and the NRC staff (slide 17 in the meeting handout, draft SE Reference 15).

During the September 20, 2007 public meeting, the NRC staff did not indicate that they disagreed with this conclusion. As a consequence, the PWROG was not provided an opportunity to provide additional information that may have satisfied the NRC staff and changed the staff's conclusion.

The NRC staff should identify the basis of their rejection of the TR WCAP-15981-NP conclusion so that licensees will have an opportunity to provide additional information in plant specific license Amendment Requests.

NRC Response:

Neutron Flux (Source Range)

Regulatory Guide (RG) 1.97 recommends that Type B Category 1 instrumentation be provided to monitor Neutron Flux to provide function detection and accomplishment of mitigation of the Reactivity Control function. Neutron Flux (Source Range) monitors the lower portion of the

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range. WCAP-15981-NP recommends that Neutron Flux (Source Range) be reclassified as Type B Category 3. In response to the NRC staff's Request for Additional Information (RAI) concerning providing an early indication of a return to criticality, the PWROG discussed monitoring boron concentration. The information provided does not support the determination that the monitoring of boron concentration via non-Category 1 instrumentation would provide adequate information about a potential return to criticality. Therefore, the NRC staff finds it unacceptable to reclassify Neutron Flux (Source Range) on a generic basis.

2. Page 12, Section 3.2.10, Lines 27 through 33:

> "TR WCAP-15981-NP recommends that Refueling Water Storage Tank Level also be classified as a Type B Category 1 key variable for the Core Cooling function. TR WCAP-15981-NP concluded that Refueling Water Storage Tank Level satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and should be included in the PAM TS. However, TR WCAP-15981-NP does not discuss how Refueling Storage Tank [(RWST)] Level instrumentation provides information concerning the Core Cooling function. Therefore, the NRC staff is unable to determine the applicability of Refueling Water Storage Tank Level to the Core Cooling function."

PWROG Comment:

> The response to item 1 on page 1 of Attachment 3 to OG-07-376 (Reference 1) provides the discussion of how Refueling Water Storage Tank Level instrumentation provides information concerning the Core Cooling function. Therefore, we believe that the NRC staff should be able to make a determination of the applicability of the Refueling Water Storage Tank Level to the Core Cooling function. Note that while this response to item 1 in Attachment 3 of Reference 1 does not explicitly refer to the Core Cooling function, the response directly discusses the role of the Refueling Water Storage Tank Level in the accomplishment of the Core Cooling function.

NRC Response:

RG 1.97 recommends that Type D Category 2 instrumentation be provided to monitor RWST Level to monitor the operation of the Safety Injection System. WCAP-15981-NP recommends that RWST Level also be classified as a Type D Category 1 variable because it provides information on the accomplishment of the Safety Injection System function and concludes that RWST Level satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) and, therefore, should be included in the PAM TS. WCAP-15981-NP also recommends that for plants with manual switchover to Emergency Core Cooling System (ECCS) recirculation, RWST Level should be classified as a Type A variable and concludes that for plants with manual switchover to ECCS recirculation, RWST Level also satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) and, therefore, should be included in the PAM TS. The NRC staff agrees with these conclusions.

WCAP-15981-NP additionally recommends that RWST level also be classified as a Type B Category 1 variable for the Core Cooling Function. The NRC staff's RAI clarification question provided by email dated July 31, 2007, asked the PWROG to, "Provide a discussion of how RWST Level provides key information for the Core Cooling function."

The PWROG response on August 22, 2007, was:

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

This response addressed how RWST Level is considered to be Type D Category 1 and Type A variables. However, it did not provide information on how RWST Level fulfills the Core Cooling function which would be a Type B variable. Therefore, sufficient information to verify the applicability of RWST Level to the Core Cooling function was not provided.

3. <u>Page 13, Section 3.2.11, Lines 1 through 7 and 33 through 37</u>:

> "The information provided in the letter dated August 22, 2007 (Reference 14), does not satisfactorily demonstrate that the Containment Sump Water Level (Wide Range) instrumentation used in the CSF status trees of the ERGs for Westinghouse NSSS plants does not meet Criterion 4 of 10 CFR 50.36(c)(2)(ii). Based on the information provided, the NRC staff does not agree with the proposed reclassification of Containment Sump Water Level (Wide Range) and concludes that Containment Sump Water Level (Wide Range) should be included in the PAM TS."
>
> "Therefore, the NRC staff agrees with the PWROG that based on the generic assessment, the Containment Sump Water Level (Wide Range) does not satisfy Criterion 4 of 10 CFR 50.36(c)(2) (ii) and need not be included in PAM TS. However, licensees will need to confirm whether this instrument should be retained in the plant-specific TS using the methodology in TR WCAP-15981-NP."

<u>PWROG Comment:</u>

> During the September 20, 2007 public meeting between the PWROG and the NRC staff (slide 21 in the meeting handout, Reference 15 in the draft SE) the basis for the TR WCAP-15981-NP recommendation that the Containment Sump Water Level (Wide Range) instrumentation could be relocated from the PAM TS was discussed with respect to its use in the Emergency Response Guideline Containment Critical Safety Function (CSF) Status Tree. While the NRC staff agreed with this position (as discussed in lines 33 through 37 in the draft SE), the NRC staff did not indicate that they disagreed with the basis for relocation of the Containment Sump Water Level (Wide Range) from the PAM TS from the perspective of indicating function detection, accomplishment of mitigation, and verification of the Maintaining RCS Integrity as discussed in the response to RAI 1, part C. f on page 10 of Attachment 1 to OG-07-292 (Reference 2). As a consequence,

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the PWROG was not provided an opportunity to provide additional information that may have satisfied the NRC staff and changed the staff's conclusion.

As a result, the draft SE offers two different conclusions regarding relocation of the same indication, i.e., Containment Sump Water Level (Wide Range) from the PAM TS (i.e., lines 1 through 7 versus lines 33 through 37). The PWROG is concerned with the inconsistency in the NRC staff's position with regard to relocating the Containment Sump Water Level (Wide Range) from the PAM TS.

Also, with respect to the NRC staff position stated in lines 1-7, the basis for the rejection of the TR WCAP-15981-NP conclusion should be identified so that licensees will have an opportunity to provide additional information in plant specific License Amendment Requests.

NRC Response:

RG 1.97 recommends that Type B Category 1 instrumentation be provided to monitor Containment Water Level (Wide Range) to provide function detection, accomplishment of mitigation, and verification of the Maintaining RCS Integrity function. RG 1.97 also recommends that Type C Category 1 instrumentation be provided to monitor Containment Water Level to provide detection of breach, accomplishment of mitigation, verification, and long-term surveillance of the Reactor Coolant Pressure Boundary function.

WCAP-15981-NP recommends that Containment Sump Water Level (Wide Range) be reclassified as Type B Category 3 and Type C Category 3. The NRC staff agrees with the conclusion that it would be acceptable to reclassify Containment Sump Water Level (Wide Range) for the Reactor Coolant Pressure Boundary function to Type C Category 3. However, the information provided in the August 22, 2007, letter did not satisfactorily demonstrate that Containment Sump Water Level (Wide Range) instrumentation, used in the CSF status trees of the ERGs for Westinghouse NSSS plants,for the Maintaining RCS Integrity function, does not meet Criterion 4 of 10 CFR 50.36(c)(2)(ii). Therefore, sufficient information to support reclassification of the Containment Sump Water Level (Wide Range) for the Maintaining RCS Integrity function was not provided.

Thus, the NRC staff's conclusions are consistent. The NRC staff has found that PWROG provided sufficient information to support relocation of the Containment Sump Water Level (Wide Range) for the Reactor Coolant Pressure Boundary function but not for the Maintaining RCS Integrity function. The statements on Page 13 of the NRC staff's SE have been modified to clarify these conclusions.

4. Page 17, Section 3.2.21, Lines 1 through 48, and Page 18 Lines 1 through 8:

"Proposed Alternate Instrumentation .... instrumentation on a generic basis."

PWROG Comment:

Except for Containment Area Radiation (High Range) which is currently addressed in the Bases for TS 3.3.3 of NUREG-1431, the NRC staff has not generically approved the

PWROG recommendations for the use of alternate indications in the event that the instrumentation in the PAM TS is inoperable. The basis for this NRC staff conclusion is tied to the apparent lack of qualification of the alternate instrumentation.
The use of alternate instrumentation was not included in any NRC Requests for Additional Information, nor were any issues raised during the public meeting held on September 20, 2007 (Reference 15 in the draft SE).
The PWROG requests that this discussion be deleted from the draft SE so that the use of alternate instrumentation and the justification for its use can be included in the TSTF that incorporates the other changes proposed by WCAP-15981-NP and approved by the NRC.

NRC Response:

WCAP-15981-NP recommended the use of alternate instrumentation for various PAM functions (Neutron Flux Power Range, High Head Safety Injection Flow, Containment Area High Range Radiation, Steam Generator Wide Range Level, and Auxiliary Feedwater Flow). These variables are discussed in the NRC staff's SE in Sections 3.2.21.1 through 3.2.21.5. RG 1.97 recommends two channels of Category 1 instrumentation for each Type A or Category 1 variable. Therefore, alternate instrumentation should also meet the Category 1 criteria.

For Neutron Flux Power Range, High Head Safety Injection Flow, and Auxiliary Feedwater Flow, WCAP-15981-NP did not discuss the qualification of the proposed alternate instrumentation. Therefore, sufficient information to verify that the proposed alternate instrumentation meets the Category 1 criteria was not provided. Based on the information provided, the NRC staff does not agree with the use of the proposed alternate instrumentation on a generic basis.

For Containment Area High Range Radiation, WCAP-15981-NP proposed that portable radiation instrumentation be used as alternate instrumentation in the event that both required channels of Containment Area High Range Radiation are unavailable. NUREG-1431 currently includes the initiation of an action that requires a report that outlines the preplanned alternate method of monitoring the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the function to operable status. This selection of a preplanned alternate method of monitoring is plant specific. Therefore, the appropriateness of the use of portable radiation instrumentation should be performed on a plant specific basis.

For Steam Generator Wide Range Level WCAP-15981-NP proposed that a combination of Steam Generator Narrow Range Level and Auxiliary Feedwater Flow be used as alternate instrumentation. The use of Steam Generator Narrow Range Level and/or Auxiliary Feedwater Flow as an alternate channel to Steam Generator Wide Range Level has been accepted previously for a limited number of plant specific applications based on the plant specific design. Therefore, the use of these alternates should continue to be reviewed on a plant specific basis.

5. Page 18 Lines 10 through 20:

"Core Exit Temperature Channels ...... for Core Exit Temperature in NUREG-1431."

PWROG Comment:

The proposed change to the number of required Core Exit Temperature channels was not included in any NRC Requests for Additional Information, nor were any issues raised during the public meeting held on September 20, 2007 (Reference 15 in the draft SE).

The PWROG requests that this discussion be deleted from the draft SE so that the proposed change to the number of required Core Exit Temperature channels and the justification for the change can be included in the TSTF that incorporates the other changes proposed by WCAP-15981-NP and approved by the NRC.

NRC Response:

NUREG-1431 includes the number of required channels of Core Exit Temperature instruments as two required channels per quadrant with a channel consisting of two Core Exit Thermocouples. WCAP-15981-NP recommended that the number of required channels be changed to two. The PWROG has not provided appropriate justification for this change. Therefore, a change in the number of channels should not be included in a TSTF until after appropriate justification has been provided and agreed upon by the NRC staff.

6. Page 20, Section 3.2.22, Lines 5 through 10:

"The NRC staff does not agree with the TR WCAP-15981-NP recommendation that the following variables can be relocated from the PAM TS for the function indicated:

Variable	Function	Type/Category
Neutron Flux (Source Range)	Reactivity Control	B1
Containment Sump Water Level	Maintaining RCS Integrity	B1"

PWROG Comment:

See Comment 1 and 3 above.

NRC Response:

No change is necessary. See responses to Comments 1 and 3 above.

7. Page 20, Section 3.2.22, Lines 12 through 18:

"The NRC staff was unable to determine the applicability of the following variable for the function indicated and, therefore, does not agree with the TR WCAP-15981-NP recommendation that the following variable should be included in the PAM TS for the function indicated:

Variable	Function	Type/Category
Refueling Water Storage Tank Level	Core Cooling	AI, D2

PWROG Comment:

See Comment 2 above.

NRC Response:

The tables in the NRC staff's SE Section 3.2.22 summarize, in a tabular form, the NRC staff's conclusions in Sections 3.2.1 through 3.2.21. Since the NRC staff's conclusions have not changed, no changes to the tables are necessary. See response to Comment 2 above.

8. Page 25, Section 4.0, Lines 9 through 15:

"As discussed in Section 3.2.21 .... for Core Exit Temperature in NUREG-1431."

PWROG Comment:

See Comment 5 above.

NRC Response:

No change is necessary. See response to Comment 5 above.

9. Page 26, Section 5.0, Lines 29 through 31:

"For those items where .... as a revision to TR WCAP-15981-NP."

PWROG Comment:

Please revise this sentence to:

"For those items where the NRC staff was unable to conclude that the proposed change was acceptable, the PWROG may submit those changes and the justification for the changes in the TSTF that incorporates the changes proposed by WCAP-15981-NP and approved by the NRC."

NRC Response:

The NRC staff's conclusion in the Draft SE included a statement that for items where the staff was unable to conclude that the proposed change was acceptable that the PWROG may submit additional information as a revision to WCAP-15981-NP.

The PWROG's proposed that changes along with justifications for proposed changes that have not been agreed upon be included in a TSTF. This approach would be inappropriate. A TSTF regarding WCAP-15981-NP should only address changes that have been approved by the NRC. Any additional proposed changes along with appropriate justification should be presented to the NRC staff for a full review.

10. Draft SE Attachment, pages 1 (source range) and 4 (RWST level and containment sump water level):

PWROG Comment:

See comments 1, 2, and 3 above.

NRC Response:

The Draft SE included an Attachment, "RG 1.97 Variables and TR WCAP-15981-NP Proposed Changes." This attachment listed, in a tabular format, each proposed change and the results of the NRC staff's review. The PWROG requested changes to the table to reflect the proposals made in Comments 1, 2, and 3. No change is necessary. See responses to Comments 1, 2, and 3 above.

PWR Owners Group

Member Participation* for Project / Task MUHP-3036 / PA-LSC-0072

Utility Member	Plant Site(s)	Participant	
		Yes	No
AmerenUE	Callaway (W)	X	
American Electric Power	D.C. Cook 1&2 (W)	X	
Arizona Public Service	Palo Verde Unit 1, 2, & 3 (CE)		X
Constellation Energy Group	Calvert Cliffs 1 & 2 (CE)		X
Constellation Energy Group	Ginna (W)	X	
Dominion Connecticut	Millstone 2 (CE)		X
Dominion Connecticut	Millstone 3 (W)	X	
Dominion Kewaunee	Kewaunee (W)	X	
Dominion VA	North Anna 1 & 2, Surry 1 & 2 (W)	X	
Duke Energy	Catawba 1 & 2, McGuire 1 & 2 (W), Oconee 1, 2, 3 (B&W)	X	 X
Entergy	Palisades (CE)		X
Entergy Nuclear Northeast	Indian Point 2 & 3 (W)	X	
Entergy Operations South	Arkansas 2, Waterford 3 (CE), Arkansas 1 (B&W)		X X
Exelon Generation Co. LLC	Braidwood 1 & 2, Byron 1 & 2 (W), TMI 1 (B&W)	X	 X
FirstEnergy Nuclear Operating Co	Beaver Valley 1 & 2 (W), Davis-Besse (B&W)	X	 X
Florida Power & Light Group	St. Lucie 1 & 2 (CE)		X
Florida Power & Light Group	Turkey Point 3 & 4, Seabrook (W)	X	
Florida Power & Light Group	Pt. Beach 1&2 (W)	X	
Luminant Power	Comanche Peak 1 & 2 (W)	X	
Nuclear Management Company	Prairie Island 1&2	X	
Omaha Public Power District	Fort Calhoun (CE)		X
Pacific Gas & Electric	Diablo Canyon 1 & 2 (W)	X	
Progress Energy	Robinson 2, Shearon Harris (W), Crystal River 3 (B&W)	X	 X

PSEG - Nuclear	Salem 1 & 2 (W)	X	
Southern California Edison	SONGS 2 & 3 (CE)		X
South Carolina Electric & Gas	V.C. Summer (W)	X	
So. Texas Project Nuclear Operating Co.	South Texas Project 1 & 2 (W)	X	
Southern Nuclear Operating Co.	Farley 1 & 2, Vogtle 1 & 2 (W)	X	
Tennessee Valley Authority	Sequoyah 1 & 2, Watts Bar (W)	X	
Wolf Creek Nuclear Operating Co.	Wolf Creek (W)	X	

* **Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.**

PWR Owners Group

International Member Participation* for Project / Task MUHP-3036 / PA-LSC-0072

Utility Member	Plant Site(s)	Participant	
		Yes	No
British Energy	Sizewell B	X	
Electrabel (Belgian Utilities)	Doel 1, 2 & 4, Tihange 1 & 3	X	
Hokkaido	Tomari 1 & 2 (MHI)		X
Japan Atomic Power Company	Tsuruga 2 (MHI)		X
Kansai Electric Co., LTD	Mihama 1, Ohi 1, 2, Takahama 1 (W)	X	
Korea Hydro & Nuclear Power Corp.	Kori 1, 2, 3 & 4 Yonggwang 1 & 2 (W)	X X	
Korea Hydro & Nuclear Power Corp.	Yonggwang 3, 4, 5 & 6 Ulchin 3, 4 , 5 & 6(CE)		X X
Kyushu	Genkai 1, 2, 3 & 4, Sendai 1 & 2 (MHI)		X
Nukleama Electrarna KRSKO	Krsko (W)	X	
Nordostschweizerische Kraftwerke AG (NOK)	Beznau 1 & 2 (W)	X	
Ringhals AB	Ringhals 2, 3 & 4 (W)	X	
Shikoku	Ikata 1, 2 & 3 (MHI)		X
Spanish Utilities	Asco 1 & 2, Vandellos 2, Almaraz 1 & 2 (W)	X	
Taiwan Power Co.	Maanshan 1 & 2 (W)	X	
Electricite de France	54 Units	X	

* **This is a list of participants in this project as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending documents to participants not listed above.**

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LIST OF ACRONYMS

AC	Alternating Current
AFW	Auxiliary Feedwater
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BAT	Boric Acid Tank
CA	Computational Aid (in SAMG)
CAL	Channel Calibrations
CCW	Component Cooling Water
CDA	Core Damage Assessment
CDF	Core Damage Frequency
CET	Core Exit Thermocouples
CFR	Code of Federal Regulations
COT	Channel Operational Tests
CSF	Critical Safety Function
CST	Condensate Storage Tank
DBA	Design Basis Accidents
DC	Direct Current
E-Plan	Emergency Plan
EAL	Emergency Action Level
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedures
EPIP	Emergency Plan Implementing Procedures
ERG	Emergency Response Guidelines
FRG	Functional Restoration Guidelines
F-V	Fussell-Vesely
HPME	High Pressure Melt Ejection
HRA	Human Reliability Analysis
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination for External Events
ISLOCA	Interfacing System Loss of Coolant Accident
LCD	Licensee Controlled Document
LCO	Limiting Conditions for Operation
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
ODCM	Offsite Dose Calculation Manual
PAM	Post Accident Monitoring
PASS	Post Accident Sampling System
PORV	Power Operated Relief Valve (refers to pressurizer)
PRA	Probabilistic Risk Assessment
PSV	Pressurizer Safety Valves

LIST OF ACRONYMS (cont.)

PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
RAW	Risk Achievement Worth
RCCA	Rod Control Cluster Assembly
RCS	Reactor Coolant System
RHR	Residual Heat Removal
ROP	Reactor Oversight Process
RPS	Reactor Protection System
RRW	Risk Reduction Worth
RTD	Resistance Temperature Detector
RWST	Refueling Water Storage Tank
SAMG	Severe Accident Management Guidance
SAT	Spray Additive Tank
SDP	Significance Determination Process
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SLB	Steam Line Break
SMA	Seismic Margins Analysis
SSC	Systems, Structures and Components
STS	Standard Technical Specifications
SW	Service Water
TDAFW	Turbine Driven Auxiliary Feedwater
UFSAR	Updated Final Safety Analysis Report
WOG	Westinghouse Owners Group

EXECUTIVE SUMMARY

The Post Accident Monitoring (PAM) instrumentation contained in NUREG-1431, Revision 3, (Reference 1), "Standard Technical Specifications Westinghouse Plants," as well as the PAM instrumentation contained in plant specific Technical Specifications for licensees of Westinghouse Nuclear Steam Supply Systems (NSSS) that have not converted to NUREG-1431 were reviewed to: 1) determine which instrumentation is important to safety following an accident and should be retained in the PAM Technical Specification, 2) determine which instrumentation which is important to safety that is not included and should be included in the PAM Technical Specification, and, 3) determine which instrumentation can be relocated from the Technical Specifications to Licensee Controlled Documents (LCDs), as well as the appropriate Regulatory Guide 1.97 classification of the relocated instrumentation.

The PAM instrumentation was included in the Technical Specifications to ensure that the instrumentation required by the operators to respond to an accident and bring the plant to a safe stable state is operable if required during an accident. The inclusion of PAM instrumentation functions in NUREG-1431 was determined based on the Technical Specification Criteria contained in 10 CFR 50.36 (c)(2)(ii), which requires that a technical specification Limiting Condition for Operation (LCO) of a nuclear reactor must be established for each item meeting one or more of the four criteria presented in the regulation. The four criteria ensure that Systems, Structures and Components (SSCs) are available to bring the plant to a safe stable condition following an accident. There are two broad classes of instrumentation that satisfy one of these criterion: those that provide input to automatic actuation of safety systems (e.g., the Reactor Protection System and the Engineered Safety Feature Actuation System), and those that provide an indication in the control room for manual operator actions. The scope of this report only covers that instrumentation and control room indication that would be required to permit an effective operator response to design basis or beyond design basis accidents to maintain the plant in a safe condition. These indications are designated as the PAM instrumentation.

The instrumentation that should be included in the PAM Technical Specification is those that satisfy either Criterion 3 or Criterion 4 of 10 CFR 50.36 (c)(2)(ii):

- Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident (DBA) or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- Criterion 4: A structure, system, or component which operating experience or Probabilistic Risk Assessment (PRA) has shown to be significant to public health and safety.

Technical Specification 3.3.3, "PAM Instrumentation," in NUREG-1431 contains a reviewer's note that states that a plant should include all Regulatory Guide 1.97 Type A and all Regulatory Guide 1.97 Category 1, non-Type A instrumentation in the PAM Technical Specification. The list of generic PAM functions identified in Technical Specification 3.3.3 was developed in the late 1980's based on DBA assumptions and generic insights from PRAs available at that time.

Regulatory Guide 1.97 (Reference 2) Type A variables provide primary information needed to permit the operators to take specified manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBAs. Regulatory Guide 1.97

Category 1 variables provide information on the accomplishment of a safety function. The definition and categorization of variables in Regulatory Guide 1.97 was developed shortly after the Three Mile Island Unit 2 accident, based on information available at that time. Significant new information is now available to permit a re-evaluation of the PAM instrumentation that should be included in the Technical Specifications, and the appropriate Regulatory Guide 1.97 classification of the PAM instrumentation proposed to be relocated from the Technical Specifications. The re-classification of the relocated PAM instrumentation is consistent with the 50.44 rulemaking that relocated the hydrogen monitors from the Technical Specifications, and re-classified them from Regulatory Guide 1.97 Category 1 to Category 3.

This report documents the results of an assessment that was performed to determine the significance of the instrumentation that was previously identified to be included in the PAM instrumentation Technical Specifications for Westinghouse NSSS plants. The assessment considered the following safety elements: DBAs, PRA, Emergency Operating Procedures (EOPs), Severe Accident Management Guidance (SAMG), and Emergency Plan Implementing Procedures (EPIP).

This assessment resulted in a revised, generic set of PAM instrumentation recommended to be included in the plant Technical Specifications for Westinghouse NSSS plants. The PAM instrumentation recommended for Technical Specification 3.3.3 of NUREG-1431 is:

- Power Range Neutron Flux
- Steam Generator Pressure
- Refueling Water Storage Tank Level
- High Head Safety Injection Flow
- Reactor Coolant System Pressure (Wide Range)
- Containment Pressure (Wide Range)
- Penetration Flow Path Containment Isolation Valve Position
- Containment Area Radiation (High Range)
- Pressurizer Level
- Steam Generator Water Level (Wide Range)
- Core Exit Temperature
- Auxiliary Feedwater Flow

The remainder of the PAM instrumentation contained in NUREG-1431 and designated as Regulatory Guide 1.97 Type A or Category 1 has been determined to have a lower safety importance and can be relocated from the Technical Specifications to Licensee Controlled Documents. Any Regulatory Guide 1.97 instrumentation that is not Type A or Category 1 can also be relocated from the Technical Specifications. In addition, for those licensees that have not converted to NUREG-1431, any instrumentation that is contained in plant specific Technical Specifications and that is not classified as Regulatory Guide 1.97 instrumentation, can be relocated from the Technical Specifications.

1 PURPOSE

The objective of this report is to develop a methodology, which is based on how the PAM instrumentation is currently used in accident management, that can be used to review the PAM instrumentation currently included in the Technical Specifications to: 1) determine which PAM instrumentation is important to safety following an accident, considering both design basis and beyond design basis accidents, that should be retained in the PAM Technical Specification, 2) determine which instrumentation which is important to safety that is not included that should be included in the PAM Technical Specification, and 3) to determine which PAM instrumentation can be relocated from the Technical Specifications to LCDs, as well as the appropriate Regulatory Guide 1.97 classification of the relocated instrumentation.

Including the PAM instrumentation that is currently used in accident management consistent with the 10 CFR 50.36 criteria will allow the operators to focus on the PAM instrumentation that is most important to plant safety, as opposed to the PAM instrumentation that is less important plant safety, which is currently included in the PAM Technical Specification, thus providing a safety benefit.

2 BACKGROUND

The primary purpose of the PAM instrumentation is to display plant variables that provide information required by the operators during an accident. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions during DBAs.

The PAM instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess the plant status and behavior following an accident.

The availability of the PAM instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments were originally identified by addressing the recommendations of Regulatory Guide 1.97 as required by Supplement 1 to NUREG-0737.

Regulatory Guide 1.97 Type A variables provide the primary information required for the operator to take specific manual actions for which no automatic control is provided, and that are required for safety systems to accomplish their safety functions as assumed in the DBA analyses.

In addition to Type A variables, Regulatory Guide 1.97 identified Category 1 variables as significant to safety. Regulatory Guide 1.97 Category 1 variables were provided to determine whether other systems important to safety are performing their intended functions.

Typically, Regulatory Guide 1.97 Type A variables are also Category 1 variables. However, not all Category 1 variables are also classified as Type A.

Technical Specification 3.3.3 in NUREG-1431 contains the generic list of PAM instrumentation for Westinghouse NSSS plants, and also contains a reviewer's note that states that a plant should include all of their Regulatory Guide 1.97 Type A and all of their Regulatory Guide 1.97 Category 1 instrumentation in the PAM Technical Specification. This generic list of PAM instrumentation was developed in the late 1980's based on DBA requirements and generic insights from PRAs available at that time.

The PAM instrumentation was included in the Technical Specifications to ensure that instrumentation required by the operators to respond to an accident and bring the plant to a safe stable state is operable during an accident. The PAM instrumentation that is currently included in Technical Specification 3.3.3 of NUREG-1431 was determined to be appropriate for control by Technical Specifications based on the application of the criteria contained in 10 CFR Part 50.36 (c)(2)(ii) as identified below:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The fourth criterion was added to 10 CFR 50.36 in 1995 to reflect the insights gained from PRA studies. As discussed below, the PAM instrumentation contained in Technical Specification 3.3.3 of NUREG-1431 is based primarily on the first three criteria of 10 CFR 50.36. Insights from PRA studies were not widely known or available at the time when Technical Specification 3.3.3 was issued in Revision 0 of NUREG-1431.

The purpose of the PAM instrumentation is to function in a post accident environment to provide indications necessary for the operators to take manual actions to mitigate the consequences of an accident, or indications that have been determined to be risk significant. Therefore, only Criteria 3 and 4 of 10 CFR 50.36 (c)(2)(ii) are applicable when evaluating instruments for retention in the PAM Technical Specification.

The original basis for determining the instrumentation to be included in Technical Specification of NUREG-1431 is defined in WCAP-11618 (Reference 3). WCAP-11618 was submitted to the Nuclear Regulatory Commission (NRC) in November 1987 and identified the PAM Instrumentation that satisfied 10 CFR 50.36 (c)(2)(ii) Criterion 3. The justification for satisfying Criterion 3 as discussed in WCAP-11618 is as follows:

"Specific Accident Monitoring Instrumentation provides the operator with the information needed to perform the required manual actions to bring the plant to a stable condition following an accident. This instrumentation is part of the primary success path which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Specific Accident Monitoring Instrumentation satisfies criterion 3."

Therefore, WCAP-11618 limited the content of proposed Technical Specification 3.3.3 to Regulatory Guide 1.97 Type A instruments. Non-Type A Category 1 instrumentation was not identified as satisfying any of the criteria for inclusion in the Technical Specifications.

The NRC letter to the Owners Groups (Reference 4), which documented the review of WCAP-11618 stated that PAM Instrumentation satisfies the definition of Type A variables in Regulatory Guide 1.97, and meets Criterion 3. The NRC justification for retaining Type A variables states: "Type A variables provide primary information (i.e., information that is essential for the direct accomplishment of the specified manual actions (including long-term recovery actions) for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBAs or transients)." It also discusses that since only Type A variables meet Criterion 3, the Standard Technical Specifications (STS) should contain a narrative statement that indicates that individual plant Technical Specifications should contain a list of PAM Instrumentation that includes Type A variables.

However, regarding non-Type A Category 1 variables, the 1988 NRC letter stated that: "the staff is unable to confirm the Owners Groups' conclusion that Category 1 Post-Accident Monitoring Instrumentation is not of prime importance in limiting risk (Criterion 4). Recent PRAs have shown the risk significance of operator recovery actions which would require knowledge of Category 1 variables. Furthermore, recent severe accident studies have shown significant potential for risk reduction from accident management. The Owners Groups' should develop further risk-based justification in support of relocating any or all Category 1 variables from the Standard Technical Specifications." The Owners Groups participating in the development of the NUREG-1431 chose not to evaluate the inclusion of Regulatory Guide 1.97 Non-Type A, Category 1 instrumentation in the PAM Technical Specification at that time. Therefore, Technical Specification 3.3.3 was issued with the requirement that all plant specific Regulatory Guide 1.97 Type A, and all plant specific Regulatory Guide 1.97 Category 1 instrumentation be included in the PAM Technical Specification.

This report was developed to specifically address the NRC request to further evaluate the inclusion of Regulatory Guide 1.97 Category 1 variables in the PAM Technical Specification. In addition, this report provides a generic methodology for developing a technical basis for relocating certain Post Accident Monitoring instruments from the Technical Specifications. The conclusions contained in this report are based on generic risk insights (i.e., evaluations against 10 CFR 50.36 (c)(2)(ii) Criterion 4) and a re-evaluation of the overall basis for Accident Monitoring instrumentation with respect to the first three Criteria of 10 CFR 50.36 (c)(2)(ii). This report also includes the consideration of the reliance on the instrumentation not specifically evaluated when the list of PAM instrumentation was originally developed in NUREG-1431. These additional considerations include instrumentation required to mitigate the consequences of beyond design basis accidents, such as those that are important for Severe Accident Management (e.g., SAMG), and offsite emergency radiological protection actions (e.g., Emergency Action Level (EAL) declarations and offsite dose calculations).

The purpose of the PAM instrumentation is to provide a reliable means of monitoring plant variables and systems following an accident (Reference 2). These indications of plant variables are required by the operators during accident situations to (Reference 2):

- Permit the operator to take pre-planned manual actions to accomplish safe plant shutdown,
- Determine whether systems important to safety are performing their intended functions, and
- Enable the determination of the potential for a gross breach of the barriers to radioactivity release.

In addition, there are other indications of plant variables that provide information on the operation of systems important to safety to the operators during an accident to:

- Permit operators to make appropriate decisions on the use of systems, and

- Permit the early determination of the need to initiation offsite emergency radiological protective actions and estimate the magnitude of the threat.

The indications of plant variables important to safety, according to the above criteria, are classified in Regulatory Guide 1.97 according to the definitions in Table 1.

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.

Table 3 PAM Instrumentation Contained in NUREG-1431	
Power Range Neutron Flux	Penetration Flow Path Containment Isolation Valve Position
Source Range Neutron Flux	Containment Area Radiation (High Range)
Reactor Coolant System Hot Leg Temperature	Pressurizer Level
Reactor Coolant System Cold Leg Temperature	Steam Generator Water Level (Wide Range)
Reactor Coolant System Pressure (Wide Range)	Condensate Storage Tank Level
Reactor Vessel Water Level	Core Exit Temperature (Quadrants 1-4)
Containment Sump Water Level (Wide Range)	Auxiliary Feedwater Flow
Containment Pressure (Wide Range)	

Some instrumentation not contained in Technical Specification 3.3.3 of NUREG-1431 is contained in the PAM Technical Specifications of other Westinghouse NSSS plants. In most cases, these plants have not converted to NUREG-1431. The additional PAM instrumentation included in the Technical Specifications for these plants are identified in Table 4.

Table 4 PAM Instrumentation Not Contained in NUREG-1431	
Auxiliary Feedwater (AFW) Valve Position	RCS Subcooling Margin
Boric Acid Tank (BAT) Level	Residual Heat Removal (RHR) Flow
Condenser Air Ejector (High Range)	Refueling Water Storage Tank (RWST) Level
Containment Enclosure Negative Pressure	Pressurizer Safety Valve Position
Containment Sump Water Level (Narrow Range)	Spray Additive Tank (SAT) Level
Containment Pressure (Narrow Range)	Spent Fuel Pool Exhaust Radiation (High Range)
Containment Water Level (Wide Range)	Steam Generator Blowdown Radiation
Intermediate Range Neutron Flux	Steam Generator Pressure
Plant Vent Stack (High Range)	Steam Generator Water Level (Narrow Range)
Pressurizer Pressure	Steam Line Radiation
Power Operated Relief Valve (PORV) Position	Turbine Driven Auxiliary Feedwater (TDAFW) Pump Exhaust Radiation
PORV Block Valve Position	High Head Safety Injection Flow

3 RE-DEFINING PAM INSTRUMENTATION REQUIREMENTS

The definition and categorization of variables in Regulatory Guide 1.97 was developed shortly after the Three Mile Island Unit 2 accident, based on information available at that time. Significant new information is now available to permit a re-evaluation of the Regulatory Guide 1.97 classification and the definition of PAM instrumentation to be included in the Technical Specifications.

3.1 BASIS

The definition and categorization of variables in Regulatory Guide 1.97 was developed shortly after the Three Mile Island Unit 2 accident in 1979. There have been two important occurrences since original development of Regulatory Guide 1.97:

- Each licensee now has an integrated safety assessment of their plant in the form of a PRA that allows the determination of the importance to safety of components and systems, and
- Significant severe accident research has been completed that provides evidence that the uncertainties associated with PRA results are sufficiently well understood such that the PRA can be used as input to regulatory decisions.

The use of PRA methodologies in regulatory decision-making is well documented over the past decade:

- Generic Letter 88-20 required all licensees to perform an Individual Plant Examination (IPE) to identify vulnerabilities in the plant design and operation that could result in unacceptably high risk, as measured by the Core Damage Frequency (CDF) and Large Early Release Frequency (LERF).
- Licensees have maintained the IPEs and converted them to more detailed PRA for use in changing burdensome regulatory requirements and improving plant reliability.
- The NRC is now using the results of PRA studies for a number of regulatory functions, including:
  - The Reactor Oversight Process (ROP) and the Significance Determination Process (SDP), which use PRA as an input in determining the safety importance of components and systems,
  - The Maintenance Rule (10 CFR 50.65), which requires the use of PRA results in determining the safety importance of components,
  - Regulatory Guide 1.174 (Reference 5), which uses PRA results, combined with deterministic analysis results and engineering judgment, to make decisions related to changes to the plant licensing basis, and
  - The rulemaking to revise 10 CFR 50.44, which relied heavily on PRA results to determine the safety importance of systems and components to measure and control post accident hydrogen in the containment.

Since the results from the licensee's PRA will be one of several inputs used to identify the plant specific instrumentation that satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii), the acceptable PRA scope and technical adequacy required to support this application must be considered. PRA technical adequacy is addressed through the PRA peer reviews and self assessments using a variety of guidance, including the American Society of Mechanical Engineers (ASME) PRA Standard (Reference 7), Nuclear Energy Institute (NEI) PRA Peer Review Process Guidance (Reference 8) and/or Regulatory Guide 1.200 (Reference 9). The purpose of these references is to assure that the PRA is of sufficient technical robustness to be used in regulatory applications. As stated in Section 2.2.3 of Regulatory Guide 1.174 Revision 1:

> *"The scope, level of detail, and technical acceptability of the PRA are to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process. The more emphasis that is put on the risk insights and on PRA results in the decision making process, the more requirements that have to be placed on the PRA, in terms of both scope and how well the risk and the change in risk is assessed.*
>
> *"Conversely, emphasis on the PRA scope, level of detail, and technical acceptability can be reduced if a proposed change to the LB results in a risk decrease or is very small, or if the decision could be based mostly on traditional engineering arguments, or if compensating measures are proposed such that it can be convincingly argued that the change is very small."*

The identification of the instrumentation that satisfies Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii) that should be included in the PAM Technical Specification is based on the operator importances from the DBA analyses, the PRA and severe accident management and emergency plan actions from the SAMG and EPIP. As shown in Table 7 for the generic results, most of the instrumentation that were found to satisfy Criterion 3 or 4 of 10CFR50.36(c)(2)(ii) for inclusion in the PAM Technical Specification are based on several of the inputs and not solely the PRA results. In addition, the PRA is only used as a risk ranking tool (e.g., use of relative risk importances rather than delta-CDF and delta-LERF). Therefore, the technical adequacy of the PRA does not need to be at the highest levels for this application.

The important features of the PRA that can impact its use in the identification of the instrumentation that satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii) are the completeness and technical adequacy of the operator actions to prevent core damage or mitigate the consequences of LERF sequences. The role of instrumentation in risk is found through the importance of operator actions modeled in the PRA that are cued from instrumentation. Therefore, it is important to assure that the operator actions that could prevent core damage or mitigate LERF are included in the PRA model and are appropriately modeled. The risk importance of instrumentation will have the greatest sensitivity to this aspect of the PRA. Of secondary importance are the technical adequacy and completeness of the accident sequence identification and quantification and the technical adequacy of the data used in the PRA for initiating event frequencies and equipment reliability.

The more extensive PRA technical adequacy requirements contained in References 7 and 9, while assuring a more robust PRA, are not required for this application, since the determination of the PAM Technical Specification instrumentation does not rely solely on the PRA and the CDF and LERF values determined from the PRA model.

The PRA scope necessary to assure that important risk insights are included in the determination of the PAM instrumentation should include at least an at-power PRA for internal initiating events that considers

CDF (a Level 1 PRA), as well as early fission product releases (a LERF assessment). A qualitative assessment of late containment failures, and core damage risks from seismic, fire and other important external initiating events should also be performed for any PAM instrument proposed to be relocated from the Technical Specifications. The qualitative assessment of external events risks will generally result in a more conservative approach in determining the safety significance of components, compared to a quantitative PRA assessment. A review of the important operator actions from several Westinghouse NSSS plants with a fully quantified external events PRA, as discussed in Appendix A, has shown that the important operator actions that are based on control room instrumentation in the external events PRA are the same as those already determined to be significant from the internal events PRA.

3.2 METHODOLOGY

The overall methodology used for assessing the importance of instrumentation to be included in the PAM Technical Specification is similar to the methodology (Reference 10) developed and used in the successful elimination of Post Accident Sampling System (PASS) requirements that specifically addressed offsite emergency radiological protection aspects important to safety.

Although the approach used in this report uses the results of PRA assessments, it is not a risk-informed application in accordance with Regulatory Guide 1.174. Rather than focusing on the five elements of a risk informed approach as specified in Regulatory Guide 1.174, this approach directly assesses the importance of instrumentation with respect to the Criteria of 10 CFR 50.36 (c)(2)(ii). This direct assessment uses the plant DBA analyses, PRA, EOPs, SAMG and EPIP as the basis for assigning importance to the instrumentation. Therefore, the methodology used in this report is more prescriptive than a risk informed approach.

PAM instrumentation is intended to provide indications of plant parameters that are the basis for important operator actions to bring the plant to a safe stable state in the event of an accident. The information available to make this determination includes:

- Design Basis Accidents – While most DBAs rely on instrumentation that provides a signal to automatically initiate systems and components to bring the plant to a safe stable state, there are also several key operator actions assumed in the DBA analyses.
- Probabilistic Risk Assessment – The PRA models a number of operator actions to bring the plant to a safe stable state and prevent core damage.
- Emergency Operating Procedures – The EOPs provide guidance for the operator response to an accident, based on instrumentation indications of plant parameters. The EOPs are the basis for the PRA and DBA operator action modeling.

- Severe Accident Management Guidance – The SAMG provides guidance for the operator response to mitigate the consequences of severe core damage, including protecting fission product boundaries. The SAMG operator actions are based on instrumentation indications of key plant parameters.

- Emergency Plan and Emergency Plan Implementing Procedures – The EPIPs provide guidance for making decisions regarding offsite radiological protective actions based on the indications of plant parameters for several key instruments.

The following screening criteria have been developed for assessing the importance to safety of the PAM instrumentation, as described below and summarized in Table 5.

Table 5 Screening Criteria for Assessing PAM Instrumentation

Area	Criteria
Design Basis Accidents	Is credit taken for operator actions in the DBA analyses documented in the Updated Final Safety Analysis Report (UFSAR) based on instrumentation indications? Instrumentation that supports these operator actions satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).
Probabilistic Risk Assessment	Is credit taken for operator actions in the PRA for a <u>high risk significant</u> function based on instrumentation indications? A high risk significance is defined from CDF and LERF Risk Achievement and Risk Reduction metrics. Instrumentation that supports these operator actions satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).
Emergency Operating Procedures	No screening criteria; importance of EOP measures is included in the DBA and PRA assessments.
Severe Accident Management Guidance	Does the instrumentation provide an indication that would result in operator actions to prevent failure of a fission product barrier that could produce a "large early release" or a "large late release"? Instrumentation that supports these operator actions satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).
Emergency Plan Implementing Procedures	Does the instrumentation provide a <u>risk significant indication</u> used to classify an accident according to the appropriate EAL? Only those criteria that would result in the declaration of a General Emergency condition are considered risk significant. Instrumentation that supports these operator actions satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).
	Does the instrumentation provide a primary indication used to assess the severity of potential fission product releases according to the Offsite Dose Calculation Manual (ODCM)? Instrumentation that supports these operator actions satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).
	Does the instrumentation provide a primary indication of the degree of core damage for the Core Damage Assessment (CDA) from which offsite radiological protection actions might be taken? Instrumentation that supports these operator actions satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

Screening criteria have not been developed for determining the instrumentation that are utilized in the EOPs. The EOPs identify a wide range of instrumentation that are the basis for operator actions. While some of the instrumentation may be important in the DBA and PRA accident analyses, a larger portion of the instrumentation is used to verify plant conditions and the success of EOP prescribed actions and is therefore not of high safety significance. Further, the operator actions in the DBA and PRA analyses are based on the instrumentation specified in the EOPs. Therefore, the screening criteria for the DBA and PRA will identify the importance of the instrumentation utilized in the EOPs.

Although this is not a risk-informed application, some of the basic elements of Regulatory Guide 1.174 have been addressed in this application. In particular, this report provides:

- Reason for Proposed Change – The reason for change touches on each of the identified categories: the change improves operational safety by including certain key risk significant instruments in the Technical Specifications that were not previously included, the change enhances the consistency of risk basis in regulatory requirements by providing a sound technical basis for satisfying Criterion 4 of 50.36 (c)(2)(ii), and the change reduces unnecessary regulatory burdens by removing certain instruments form the PAM Technical Specifications that do not directly impact safety.

- Defense in Depth – Defense in Depth has been considered to ensure that a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved by maintaining those instruments in the PAM Technical Specification that are important for preventing core damage, maintaining containment integrity and implementing offsite emergency planning activities. Also, redundancy, independence, and diversity are maintained by identifying those instruments that can be used as back-ups to the instruments included in the PAM Technical Specifications.

- Safety Margins – Safety margins are maintained by ensuring that the instrumentation used to support operator actions credited in the design basis accident analyses are controlled by the PAM Technical Specifications.

- Risk Impact – The risk impact of instrumentation to support operator actions is considered by using the available risk assessment tools, as discussed in Appendix A, including the at-power PRA, the fire and seismic PRA assessments, the Level 2 PRA containment integrity assessment, the Severe Accident Management Guidance and the Site Emergency Plan. Risk importance measures were used to identify instrumentation that supports risk significant operator actions in the Level 1 PRA. The assessments from the other risk assessment tools (e.g., Level 2 PRA, SAMG, E-Plan) were more qualitative, but provide the key insights regarding the importance of instrumentation in preventing or mitigating risk significant conditions.

- Instrumentation that is relocated from the PAM Technical Specifications to LCDs will still be monitored for availability and subject to appropriate corrective action where appropriate.

Therefore, it is concluded that an appropriate process has been used to consider the re-definition of the plant instrumentation that should be included in the PAM Technical Specification and a re-classification of the PAM instrumentation proposed to be relocated from the Technical Specifications, similar to the re-classification of the hydrogen monitors from Category 1 to Category 3 in the 50.44 rulemaking.

The methodology for determining the PAM instrumentation that should be included in the Technical Specifications, and the PAM instrumentation that can be relocated from the Technical Specifications was based on generic DBA, PRA, EOP, SAMG, and EPIP information for Westinghouse NSSS plants. Therefore, implementation of this methodology on a plant specific basis requires the confirmation of the generic conclusions contained in the WCAP by reviewing the plant-specific DBA analyses, PRA, EOP, SAMG, and EPIP.

4 INSTRUMENTATION ASSESSMENT

This section provides the results of an assessment of the use and importance of instrumentation in the DBA analyses, the PRA, the EOPs, the SAMG, and the E-Plan/EPIPs.

4.1 DESIGN BASIS ACCIDENT ANALYSIS

While many of the DBAs are analyzed assuming the automatic actuation of systems and components, several of the DBAs also assume operator actions. The operator actions modeled in the DBA analyses are based on conservative time windows available for action, but are based on reliable instrumentation indications to diagnose the need for such actions. The DBAs that typically assume operator actions in the safety analyses are discussed below. The instrumentation indications upon which the operator actions are based would therefore satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

Loss of Coolant Accidents

In the event of a Loss of Coolant Accident (LOCA), the design basis analyses assume that the operator takes the appropriate actions for the transfer to Emergency Core Cooling (ECC) recirculation based on RWST level. In some older plant designs, the transfer to ECC recirculation consists entirely of manual actions by the operators, whereas in the newer plant designs, most or all of the required actions are automatic. There are no other short term operator actions assumed in the design basis LOCA analyses. In all cases, the operators would be alerted to the RWST inventory decrease to a point where transfer to the recirculation mode is required by a control room alarm and/or by the initiation of those automatic actions associated with switchover to ECC recirculation. The RWST level would typically only be used to confirm the initiation of the transfer to recirculation, based on either automatic actions or a switchover alarm. In some older plant designs, RWST level alone, without alarm indication, is the basis for cueing the operator initiation of switchover to ECC recirculation.

At a specified time after the initiation of the accident (e.g., 4 to 20 hours depending on plant specific analyses), the design basis analysis assumes that a switchover to hot leg recirculation is required to limit the potential for boron build-up in the reactor vessel. This is performed manually by the operators. However, the cue to perform hot leg recirculation switchover is based on time as opposed to any plant variables.

Additionally, the radiological dose analysis for the LOCA is typically based on continued operation of containment spray after the RWST has been emptied. For many large dry containment plants in the WOG fleet, an assumed operator action to transfer containment spray to the recirculation mode when the RWST is nearly empty (based on a low-low level alarm) is embedded in the design basis analyses. Failure to transfer containment spray pump suction from the injection mode to the recirculation mode could result in damage to the containment spray pumps. For plant designs that use the containment spray pumps for spray recirculation, the failure to switch to containment spray recirculation could impact the containment pressure response assumed in the design basis analyses. Thus, the RWST level indication could be an important indication for a DBA in which no automatic control is provided.

Steam Generator Tube Rupture

In the event of a Steam Generator Tube Rupture (SGTR), the design basis analyses assume that the operators will diagnose the occurrence of an SGTR accident, isolate the secondary side of the ruptured Steam Generator (SG), terminate AFW flow to the ruptured SG to prevent SG overfill, and initiate cooldown and depressurization of the RCS to terminate the loss of reactor coolant inventory through the ruptured tube. The SGTR accident requires operator actions for which no automatic control is provided to bring the plant to a safe stable state. The specific operator actions typically assumed in the DBA analysis are:

- Identify and isolate ruptured SG – based on SG level (diagnose rupture) and SG pressure (identify stuck open relief valve which affects the recovery strategy),
- Operator action to initiate cooldown using the intact SGs – based on ruptured SG pressure (cooldown target) and RCS pressure and temperature (maintain subcooling),
- Operator action to initiate RCS depressurization using the pressurizer spray, auxiliary spray or PORVs – based on RCS pressure, RCS temperature, pressurizer level and RCS subcooling, and
- Operator action to initiate SI termination – based on SI termination criteria of RCS pressure, RCS temperature and pressurizer level.

The primary diagnosis of the design basis SGTR event is based on comparing water levels in the SGs using the SG level indication. Secondary indications of a SGTR include steam line, condenser air ejector and SG blowdown radiation levels. Since the RCS radioactivity levels are very low in the plants that are currently operating, and are nearly undetectable at the beginning of a fuel cycle, the radiation monitors are less reliable than SG water levels for the diagnosis of a SGTR. Following a reactor trip caused by decreasing RCS pressure (due to the inventory loss through the ruptured SG tube), the AFW flow to each SG would be approximately equal. Because of the additional mass addition to the affected SG through the ruptured tube, the SG levels would quickly indicate which SG was affected. Isolation of the ruptured SG secondary side and termination of AFW flow to the ruptured SG do not depend on any instrumentation. Operator actions to cooldown and depressurize the RCS depend on several different instrumentation indications:

- SG Level indication is used to control AFW flow to the non-ruptured SGs during SG depressurization to assure that adequate level is maintained in the intact SGs as the SGs are depressurized,
- RCS Temperature indication is used during the depressurization to control the rate of RCS cooldown,
- RCS Pressure and SG pressure are used during the depressurization to determine when the pressures are equalized, which indicates that reactor coolant loss through the ruptured SG tube has been terminated,

- 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 accident condition and the need for operator actions. A portion of the operator error assessment is based on the availability and accuracy of the instrumentation indication that is the basis for the operator action. Operator actions that are important for accident prevention and accident mitigation are modeled in the PRA. Therefore, if a parameter indication (i.e. instrument) does not support an operator action modeled in the PRA, then it can be assumed to be of very low risk significance. The methodology for the treatment of instrumentation in the operator error assessment varies from plant to plant, but is typically included in the model in an explicit manner.

Key PRA results obtained from a survey of all Westinghouse NSSS plants are available in a composite PRA database. The importance of operator actions for preventing core damage for at-power initiating events identified in the PRA database have been analyzed in detail to determine the importance of instrumentation required for those operator actions. A detailed discussion of this analysis is presented in Appendix A of this report. A detailed assessment of the importance of operator actions to prevent failure of containment fission product boundaries (e.g., LERF and late containment failures) has also been completed and is included in Appendix A of this report.

From a risk perspective, the following indications have been determined to typically have a high degree of importance for preventing core damage for at-power initiating events, according to a composite PRA model of Westinghouse NSSS plants:

- RWST Level (median Risk Achievement Worth (RAW) = 10.35),
- SG Wide or Narrow Range Level* (median RAW = 4.05),
- RCS Subcooling (median RAW = 4.05),
- RCS Temperature (median RAW = 4.05),
- RCS Pressure (median RAW = 4.05),
- Pressurizer Level (median RAW = 4.05),
- SG Pressure (median RAW = 4.05),
- High Head SI Flow (median RAW = 3.05),
- Power Range Neutron Flux Monitor (RAW = 2.49),
- SG Wide Range Level** (median RAW = 2.46), and
- AFW Flow (Median RAW = 2.46).

*Based on maintaining SG level during RCS cooldown and depressurization

**Based on initiation of bleed and feed mode of core cooling

The following instrumentation has been determined to typically have a high degree of importance for preventing large, early radioactive releases, based on the LERF assessment discussed in Appendix A.6. No importance measures have been quantified for the following instrumentation, as the LERF assessment was qualitative in nature:

- Containment Pressure (Wide Range),
- Penetration Flow Path Containment Isolation Valve Position, and
- RCS Pressure.

From a risk perspective, all other instrumentation has a low or negligible importance for preventing core damage, according to a composite PRA model of Westinghouse NSSS plants and therefore does not satisfy Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

4.3 INSTRUMENTATION UTILIZED IN EMERGENCY OPERATING

The EOPs for Westinghouse NSSS plants are based on the generic WOG Emergency Response Guidelines (ERGs) (Reference 11). The ERGs were developed to provide procedures for bringing the plant to a safe stable state for any accident sequence initiated at power operation that results in a reactor trip or SI signal. The ERGs are designed to be consistent with the DBA analyses, but also consider accidents beyond the design basis. In addition, the ERGs provide guidance for the use of all means to bring a plant to a safe stable state. That is, in the event of a failure of a plant system or component assumed to be operable in the design basis analyses, the ERGs provide alternate methods for achieving the same desired endstate.

Considering all of the contingency procedures for dealing with events beyond the design basis, the ERGs make use of a large amount of the plant instrumentation in providing guidance to the operators for bringing the plant to a safe stable state. The plant PRA models the significant paths through the EOPs for a wide range of possible initiating events. The plant PRA model will typically model all of the EOP actions assumed in the DBA analyses. If the EOP action is not modeled in the PRA, then it is considered to have a negligible impact on plant risk. EOP actions might not be included in the PRA model for several key reasons:

- Failure to perform the EOP action does not impact the accident sequence progression and therefore does not impact the PRA results,

- The EOP action is on a pathway that has been shown to be of very low probability and therefore does not impact PRA results, or

- The EOP action is very late in the accident sequence such that the probability of failing to take the action is considered negligible. This rationale is seldom used but can apply when the time at which the action is required approaches 24 hours after the initiating event. Examples include switchover to hot leg recirculation and refilling the Condensate Storage Tank (CST), which are typically not required in the first 12 to 24 hours of the accident.

The EOPs include Functional Restoration Guidelines (FRGs). These are symptom-based indications that an accident is not proceeding according to the design basis and provide a second level of defense against a transient or an accident progressing to core damage. The FRGs are entered from the Critical Safety Function (CSF) Status Trees; the CSF Status Trees are monitored whenever the EOPs are in use.

All EOP operator actions that are important for preventing damage to the reactor core are modeled in the plant PRA. Additionally, EOP instrumentation that may be important for an offsite emergency response to protect the health and safety of the public is included in the EAL declaration criteria. Therefore, the importance of EOP instrumentation is deferred to the DBA, PRA and EAL discussions (Sections 4.5.1 and 4.5.2).

As discussed above, the expected operator actions for design basis accidents and many beyond design basis events are contained in the EOPs and the CSF Status Trees would typically only be used for beyond design basis events. Therefore, by definition, the indications used in the CSF Status Trees do not meet Criterion 3 of 10 CFR 50.36 (c)(2)(ii) solely on the basis of their inclusion in the CSF Status Trees. However, the CSF Status Tree usage may be important from the PRA perspective and may satisfy Criterion 4 of 10 CFR 50.36 (c)(2)(ii). The indications relied upon for usage of the CSF Status Tree are discussed below.

There are four levels of priority in the CSF Status Trees:

- Only Red paths require immediate attention,
- Orange paths require immediate attention when no Red paths are available,
- Yellow paths are to be addressed at the operator's discretion when time is available,
- Green paths require no action on the Functional Restoration Procedures.

Only the Red and Orange CSF paths need to be considered in assessment of instrumentation needed to satisfy Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

Subcriticality CSF Status Tree

The Power Range neutron flux is used to indicate a Red Path to FR-S.1. The Power Range neutron flux is risk significant in the generic PRA (Appendix A) and therefore satisfies Criterion 4 and will be retained in PAM Technical Specification.

The Intermediate Range Startup Rate is used to indicate a Red Path to FR-S.1. The Intermediate Range Startup Rate is not risk significant in the generic PRA (Appendix A) and therefore does not satisfy Criterion 4 and will not be retained in the PAM Technical Specification. The Intermediate Range Startup Rate needs to be addressed in the plant specific implementation to determine its risk significance.

The Source Range neutron flux instrumentation is only used in the CSF Status Trees to indicate a Yellow of Green CSF path and therefore does not satisfy Criterion 4.

It is also noted that the Channel Operational Tests (COTs) and Channel Calibrations (CALs) in the surveillance requirements (SR 3.3.1.7, SR 3.3.1.8, SR 3.3.1.11) of the Reactor Trip System Instrumentation Technical Specification verify that IR and SR indications are available. Therefore, the Intermediate Range Startup Rate and Source Range neutron flux instrumentation can be removed from the PAM Technical Specification, but will remain controlled by the Reactor Trip Technical Specification.

Core Cooling CSF Status Tree

The Core Exit Temperature indication is used to indicate a Red Path to FR-C.1 and an Orange Path to FR-C.2. The Core Exit Temperature indication was found to be risk significant in the generic PRA (Appendix A) and will be retained in the PAM Technical Specification.

Reactor Vessel Water Level indication is used to indicate a Red Path to FR-C.1 and an orange path to FR-C.2 when the Core Exit Temperature indication is less than the CSF Status Tree setpoint value. The Reactor Vessel Water Level indication was found not risk significant in the generic PRA (Appendix A) and therefore does not satisfy Criterion 4 and will not be retained in the PAM Technical Specification. Reactor Vessel Water Level indication needs to be addressed in the plant specific implementation to determine its risk significance

Heat Sink CSF Status Tree

The Auxiliary Feedwater Flow and Steam Generator Water Level are used to indicate a Red Path to FR-H.1. The Auxiliary Feedwater Flow and Steam Generator Water Level were found to be risk significant in the generic PRA and therefore satisfy Criterion 4 and will be retained in the PAM Technical Specification.

The Heat Sink CSF Status Tree uses SG Narrow Range level as a primary indicator but operators are trained to also use SG Wide Range level. The FR-H.1, "Response to Loss of Heat Sink" procedure uses SG Wide Range level to initiate Bleed and Feed which is risk significant in the PRA. Therefore only SG Wide Range level indication needs to be included in the PAM Technical Specification.

Integrity CSF Status Tree

The RCS Cold Leg Temperature and RCS Pressure are used to indicate a Red Path and an Orange path to FR-P.1.. The purpose of the Integrity CSF Status Tree is to provide guidance on preventing a challenge to reactor vessel integrity due to Pressurized Thermal Shock (PTS). PTS is not risk significant in the generic PRA (Appendix A). Several plant specific risk assessments documented in NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)", conclude that reactor vessel failure leading to core damage due to PTS is not risk significant. Therefore Cold Leg Temperature indication does not satisfy Criterion 4 and will not be retained in the PAM Technical Specification (RCS Pressure indication is retained based on other considerations). Cold Leg Temperature indication needs to be addressed in the plant specific implementation to determine its risk significance

It is also noted that the Reactor Trip System Instrumentation Technical Specification contains requirements for the Overtemperature ΔT and Overpressure ΔT reactor trip functions. The associated COTs and CALs verify that Tavg and ΔT indications are available, based in part on the RCS Cold Leg Temperature indication.

Containment CSF Status Tree

The Containment Pressure indication is used to indicate a Red Path and an Orange path to FR-Z.1. Containment Pressure is risk significant in the generic PRA (Appendix A) and will be retained in the PAM Technical Specification.

Containment Sump Water Level Wide Range indication is used to indicate an Orange Path to FR-Z.2 which deals with containment water levels approaching the containment flooding level. Prevention of Containment Flooding is not risk significant in the generic PRA (Appendix A). Therefore, Containment Sump Water Level Wide Range indication does not satisfy Criterion 4 and will not be retained in the PAM Technical Specification. Containment Water Level Wide Range indication needs to be addressed in the plant specific implementation to determine its risk significance.

Inventory CSF Status Tree

All paths of the Inventory CSF Status Tree are Yellow or Green and therefore do not need to be addressed immediately. Pressurizer Water Level indication and Reactor Vessel Water Level indication are used to indicate Yellow Paths in the Inventory CSF Status Tree. RCS inventory concerns addressed using FR-I.1 through FR-I.3 are not risk significant in the generic PRA (Appendix A). Pressurizer Level is retained in PAM Technical Specification for other risk significant operator actions. Reactor Vessel Water Level indication does not satisfy Criterion 4 and therefore will not be retained in the PAM Technical Specification. Reactor Vessel Water Level indication needs to be addressed in the plant specific implementation to determine its risk significance.

4.4 INSTRUMENTATION UTILIZED IN SEVERE ACCIDENT MANAGEMENT GUIDANCE

The SAMG for Westinghouse NSSS plants is based on the generic WOG SAMG (Reference 12). The SAMG was developed to provide guidance for bringing the plant to a safe stable state and mitigating fission product releases to the environment for accident sequences that result in core damage. The SAMG makes use of 6 plant parameters for establishing and/or maintaining a safe stable state following a core damage accident. The parameters and instrumentation used to determine the value of each parameter are:

- SG Water Level – SG Water Level wide range or narrow range instrumentation,
- RCS Pressure – RCS Pressure wide range instrumentation,
- Core Temperature – CETs and/or Hot Leg RTDs instrumentation,
- Containment Sump Water Level – Containment Sump Water Level wide range and/or narrow range instrumentation,

- Containment Pressure – Containment Pressure wide range instrumentation,
- Containment Hydrogen – Containment Hydrogen Monitor instrumentation.

The purpose and safety significance of each of these parameters is discussed below. The SAMG is designed to be used only in the event of core damage, which means that, by definition, the accident is beyond the design basis.

SG Water Level

The primary purpose of restoring/maintaining SG water level for Westinghouse NSSS plants is to provide a heat sink for decay heat removal when core cooling is restored. The SG water level can also provide fission product scrubbing to reduce fission product releases for accident sequences in which the SG tubes act as a pathway for releases to the environment, such as for a SGTR initiating event that proceeds to core damage. Further, the SG water level is important to protect the SG tubes from thermally induced creep failure after core damage for certain accident sequences. For Westinghouse NSSS plants, thermally induced SG tube failure after core damage is considered to be a very low probability challenge to the integrity of the plant fission product barrier, and therefore maintaining SG water level only provides additional defense-in-depth assurance against this low probability challenge. Lastly, the SG water level is important for preventing SG overfill which would result in a two-phase flow being released from the SG relief valves, which may cause the valves to stick open, since the valves are only designed to relieve steam. For SGTR events, SG overfill is the primary cause of core damage leading to a large early release of fission products.

From a PRA perspective, operator actions to restore or maintain SG water level after core damage are generally not modeled. The only potential PRA accident sequences in which SG water level could have a risk impact is for the SGTR sequences. The current modeling of the consequences from a SGTR initiated core damage accident assumes no water addition to the SG for fission product scrubbing. SAMG operator actions to maintain a water cover over the top of the highest tube would provide a water pool for fission product scrubbing, which would significantly reduce the release quantities. Therefore, operator actions to maintain an adequate SG level in the ruptured SG in accordance with the SAMG are considered to be risk important.

For the SAMG, SG water level is measured for this indication using the SG wide range or narrow range level instrumentation.

RCS Pressure

The primary purpose of reducing the RCS pressure for Westinghouse NSSS plants is to permit the use of low pressure sources of water injection to the RCS. RCS depressurization is directed in the EOPs prior to core damage for all accident sequences, except the loss of all AC power with coincident failure of TDAFW pump (or diesel driven pump) to supply water to the SGs. For this core damage accident sequence, RCS depressurization using the pressurizer PORVs would only be a temporary measure to arrest core damage using the accumulator water; if AC power is not restored, the core damage scenario will continue. Typically, for Westinghouse NSSS plants, recovery of AC power in the short time frame that might be effective to arrest the core damage accident in-vessel is not a risk significant accident sequence. A secondary purpose for reducing RCS pressure is to avoid reactor vessel failure with the RCS at a high pressure (also known as a high pressure melt ejection), if the core damage accident cannot be terminated before the core melts and relocates to the reactor vessel bottom head. For Westinghouse NSSS plants, high pressure melt ejection is typically considered to have a very low probability of challenging the containment integrity.

From a PRA perspective, operator actions to depressurize the RCS after core damage are typically not modeled. The PRA accident sequences in which RCS pressure could have the largest risk impact are SGTR sequences that result in core damage. The current modeling of the consequences from a SGTR initiated core damage accident assumes that no RCS depressurization occurs to limit the primary to secondary fission product transport. Therefore operator actions to depressurize the RCS to mitigate fission product releases for core damage sequences involving a SGTR are considered to be risk important in the SAMG, and RCS pressure indication in the SAMG is considered to be risk important.

For the SAMG, RCS pressure is measured using the RCS wide range pressure instrumentation.

Core Temperature

The primary purpose of monitoring core temperature is to determine whether attempts to restore core cooling have been successful. No other actions are directed from monitoring core temperature. Recovery of core cooling would terminate the core damage accident and potentially result in a release of only a fraction of the fission products in the core. A secondary purpose of recovery of core cooling prior to reactor vessel failure would be to prevent the core from going ex-vessel. However, since ex-vessel phenomena are generally not significant contributors to plant risk, this action would have little impact on the overall plant risk.

From a PRA perspective, the recovery of core cooling after core damage has occurred is modeled in some plant PRAs. However, no credit is taken for partial core damage in assessing the change in consequences associated with recovery of core cooling while the core is still in-vessel. Operator actions to restore core cooling prior to reactor vessel failure could have the largest risk impact for the SGTR sequences that go to core damage. The current modeling of the consequences from a SGTR initiated core damage accident assumes no change in the quantity of fission products released to the atmosphere. As stated previously, there are no other significant threats to fission product boundaries associated with core temperature. Since a high core temperature is a prerequisite for entering the SAMG, the operators would already be aware of a high core temperature and already attempting to re-establish core cooling following entry into the SAMG. Since there are no additional SAMG actions cued from core temperature, it is concluded that core recovery after core damage is not a risk significant SAMG instrumentation.

For the SAMG, it is expected that indications to diagnose recovery of core cooling may be unreliable due to the high temperature environment in the RCS during core uncovery. Therefore, the SAMG recommends that no one indication be relied upon to diagnose recovery. Thus, there is no unique instrumentation identified for this purpose.

Containment Water Level

The primary purpose of monitoring containment water level in the SAMG is to enable operator actions to assure that water is available to cool any core debris that relocates to the containment if the reactor vessel fails. A secondary purpose of providing adequate containment water level is to assure that adequate water is available in the containment sump for ECC recirculation should a means of injection to the RCS become available.

In the case of water availability for core debris cooling, an adequate water supply in the containment can prevent core-concrete interactions from occurring which, in turn, prevents containment pressurization from noncondensible gases that result from core-concrete interactions. These noncondensible gases can challenge containment integrity in the long term (e.g., 3 to 4 days after core damage). In the case of water availability for ECC recirculation, the PRA results show that there is not a significant probability of an accident scenario in which injection to the RCS via ECC recirculation is available, and the containment recirculation sump is dry.

From a PRA perspective, water addition to the containment after core damage has occurred is not typically modeled in plant PRAs. Operator actions to add water to the containment for core damage accidents could have the largest risk impact for accident sequences in which containment heat removal is available, but no water is available in the containment for core debris cooling. In these cases, the challenge to containment integrity from the noncondensible gases generated from core-concrete interactions would be prevented. However, these accident sequences typically represent a small fraction of the accident sequences that result in a containment challenge following core damage. Therefore, it is concluded that the containment water level indication following core damage does not enable a risk significant action.

For the SAMG, containment water level to prevent core-concrete interactions is measured from wide range containment level instrumentation. Containment water level to assure that ECC recirculation is available can be measured from either the wide range RWST level, or the narrow range or the wide range containment level instrumentation.

Containment Pressure

The purpose of the containment pressure instrumentation, as utilized in the SAMG, is to either: a) indicate the containment pressure to allow the operators to take action to vent the containment to prevent a catastrophic containment failure due to overpressurization, or b) determine if a hydrogen burn challenge to the containment integrity exists requiring consideration of hydrogen control strategies. Since long term containment overpressurization and slow burn overpressurization are typically the dominant late containment failure modes, containment pressure is important for SAMG mitigation. The PRA analyses show that SAMG initiated operator actions for hydrogen control strategies would only be required for station blackout events after all instrumentation is lost due to battery depletion. It is not important for any accident in which instrumentation is available and therefore only long term overpressurization is of interest. SAMG initiated operator actions for containment venting due to long term overpressurization would only be considered as the containment pressure approaches the lower bound of the failure pressure. Only the containment high range pressure indication is adequate for this consideration and should be included with respect to the SAMG. Therefore, it is concluded that the containment wide range pressure instrumentation is an important SAMG indication.

For the SAMG, containment pressure is measured from wide range containment pressure instrumentation.

Containment Hydrogen

The purpose of the containment hydrogen instrumentation, as utilized in the SAMG, is to indicate the containment hydrogen concentration to allow the operators to take action to prevent a containment failure due to a hydrogen burn. The potential for early containment failures due to hydrogen burns was analyzed in the Level 2 PRA and found not to be risk significant. The potential for slow burn overpressurization containment failures is a contributor to the probability of late containment failures, but only for station blackout events with no power recovery in the first day. In this case, there would be no power available for instrumentation so the availability of hydrogen monitors is not relevant to this case.

For Westinghouse NSSS plants with ice condenser containments, the dedicated hydrogen igniters, which burn hydrogen as it is released to the containment, would prevent the containment hydrogen concentration from accumulating and reaching a level that could challenge containment if ignited.

Further, in the rulemaking to revise 10 CFR 50.44, the NRC determined that the importance of containment hydrogen monitoring could be downgraded since it was not important to risk. Therefore, it is concluded that operator actions for hydrogen control based on containment hydrogen indication is not a risk significant operator action.

Therefore, it is concluded that monitoring containment hydrogen for the purpose of venting containment to prevent a challenge to containment integrity following core damage is not a risk significant action.

For the SAMG, containment hydrogen is measured from containment hydrogen instrumentation.

4.5 INSTRUMENTATION UTILIZED IN EMERGENCY PLANNING

The E-Plan and the EPIP for Westinghouse NSSS plants rely on plant instrumentation for three distinctly separate activities: 1) Assessment of the appropriate EAL, 2) Offsite Dose Projections using the ODCM, and 3) CDA. The role of instrumentation for these activities is discussed below.

4.5.1 Emergency Action Levels

The EALs provide a means of communicating between the plant staff and the offsite authorities regarding the potential for fission product releases from the plant that could endanger the health and safety of the public in the vicinity of the plant. Based on the potential for fission product releases from the plant, the offsite authorities would invoke various levels of offsite emergency protective actions for members of the general public, including sheltering and evacuation.

The majority of the Westinghouse NSSS plants use the EAL assessment methods in either NUMARC/NESP-007 (Reference 13) or in NUREG-0654 (Reference 14). The greatest severe potential for fission product releases is associated with the General Emergency level. This is classified as a loss of any two barriers and a potential loss of a third barrier. At lesser levels (e.g., Site Emergency), the potential for fission product releases is not imminent, although the emergency level could escalate to the General Emergency Level at some future time. However, the same indications are used to classify these lower levels (e.g., Site Emergency declaration is based on a potential loss of two barriers). The plant indications, obtained from plant instrumentation, that are used in declaring a General Emergency condition are shown in the Table 6.

An evaluation of the indications and related instrumentation, based on the plant EOP structure and the results from PRA analyses, results in the following conclusions related to the EAL instrumentation shown in Table 6:

- The potential loss of the fuel rod clad barrier can be determined from elevated and increasing CET indications, reactor vessel level and loss of heat sink indications. However, the CET indication provides the most direct and unambiguous indication of the potential loss of fuel rod clad barrier. The reactor vessel level and heat sink indications only provide indication that conditions exist that may lead to a loss of the fuel rod clad barrier.

- The loss of fuel rod clad barrier will always be indicated first by high CET indications. Containment and RCS letdown radiation levels will always lag the CET temperatures and may be useful only to confirm the loss of the fuel rod clad barrier. The issue with the radiation monitors is that a pathway must exist for the fission products to reach the volume being monitored for high radiation levels. In the case of the containment, a breach in the RCS must also exist for high radiation to be present in the containment. For the letdown monitor, the RCS must not have been isolated based on a safety injection signal.

- The potential loss of the RCS barrier can be determined from the potential for reactor vessel pressurized thermal shock, and the loss of heat sink. The potential for entering a condition where pressurized thermal shock can result in a failure of the reactor vessel is a very low probability event. Current analyses and evaluation of the potential for pressurized thermal shock of the reactor vessel to result in core damage are still ongoing. However indications are that the criteria for reactor vessel aging can be relaxed as a result of these analyses because of the low potential for core damage. The potential loss of the RCS barrier due to a loss of heat sink is based on the opening of the pressurizer PORV or safety valve if a heat sink cannot be recovered. It is not a reliable indication of a loss of the RCS barrier.

- The loss of RCS barrier can be determined from an RCS leakage indication, containment radiation and diagnosis of a SGTR. The loss of the RCS barrier based on the RCS leak rate as measured by RCS subcooling is not a reliable indicator. For most LOCAs, RCS subcooling can be maintained or recovered due to the cold water addition from ECC. The loss of RCS subcooling may be a better indicator of the potential loss of the fuel rod clad barrier, since it is a precursor to core uncovery and heatup.

- The potential loss of the containment barrier can be determined from high containment pressure, and indication of a faulted SG. Based on an evaluation of extensive PRA studies, containment explosive mixtures of hydrogen are not expected in large dry, subatmospheric or ice condenser containment PWRs. Flammable mixtures of hydrogen are not expected to challenge the containment barrier in these containments. Therefore, containment hydrogen is not a reliable indicator of a potential loss of the containment barrier.

Table 6 Instrumentation Utilized for the Determination of Emergency Action Levels		
Fission Product Barrier	**Indication**	**Instrumentation**
Loss or Potential Loss of Fuel Rod Clad Barrier –	Critical Safety Function Status: Core Cooling Orange or Red	• Core Exit Thermocouples • Reactor Vessel Water Level
	Critical Safety Function Status: Heat Sink Red	• SG Wide Range Level • Auxiliary Feedwater Flow
	Reactor Coolant Activity Level	• RCS Letdown Radiation Monitor
	Core Exit Thermocouple Readings	• Core Exit Thermocouples
	Reactor Vessel Water Level	• Reactor Vessel Water Level
	Containment Radiation	• Containment High Range Radiation
Loss or Potential Loss of Reactor Coolant System Barrier –	Critical Safety Function Status: RCS Integrity Red	• RCS Wide Range Pressure • RCS Cold Leg RTD
	Critical Safety Function Status: Heat Sink Red	• SG Wide Range Level • Auxiliary Feedwater Flow
	RCS Leak Rate	• RCS Subcooling • Pressurizer Level
	SG Tube Rupture	• SG Narrow Range Level
	Containment Radiation	• Containment High Range Radiation
Loss or Potential Loss of Containment Barrier –	Critical Safety Function Status: Containment Red	• Containment Wide Range Pressure
	Containment Pressure	• Containment Wide Range Pressure • Containment Narrow Range Pressure
	Containment Explosive Mixture	• Containment Hydrogen Monitor
	Core Exit Thermocouple Reading	• Core Exit Thermocouples
	Faulted SG	• SG Pressure
	Containment Isolation Valve Status	• Containment Isolation Valve Position
	Containment Radiation	• Containment High Range Radiation

- The loss of the containment barrier is only indicated by containment pressure (a sudden and unexpected decrease in containment pressure) or containment isolation valve position indication. Containment radiation levels and CET indications are not reliable indicators of either a potential loss or a loss of the containment barrier for PWRs. PRA results show that there is no correlation between these parameters and containment failure.

For the purposes of the protection of the health and safety of the offsite general public, the key indicators of the need to implement offsite emergency protective actions are high CET indications, high containment radiation levels, failure of complete containment isolation, and/or high containment pressures. The other indications are most useful to validate the loss of barriers, not as primary indications of the potential for or the loss of the barrier.

4.5.2 Core Damage Assessment

The ability to assess the occurrence of and degree of core damage is a NUREG-0737 (Reference 15) requirement. In 1999, a core damage assessment methodology was developed by the Westinghouse Owners Group (WOG) in conjunction with the elimination of the requirements for a post accident sampling system (WCAP-14696-A, Reference 16). The findings of the core damage assessment would provide input to the offsite emergency planning activities.

The new core damage assessment methodology relies solely on instrumentation to determine the occurrence of and degree of core damage. The methodology uses two primary indicators, based on the analytical modeling of a wide range of core damage accidents:

- CETs, and
- Containment radiation.

Due to the variability in these indications across a wide range of potential core damage sequences, a series of secondary indicators was specified. The variability in the indications from these secondary indicators across the same range of accident sequences is much larger than the variability of the primary indicators. However, it is believed that these secondary indicators could be used to confirm the primary indications. Where differences in the expected behavior between the primary and secondary indicators are found, a number of considerations are called upon to arrive at a best estimate of the occurrence of core damage and the degree of core damage. The secondary indicators used in WCAP-14696 are:

- Containment hydrogen,
- Reactor vessel level indication,
- RCS hot leg RTDs, and
- Source range neutron flux.

It should be noted that the instrumentation for core damage assessment is also used in other key functions discussed in this report. None of the instrumentation recommendations in this report are solely based on the core damage assessment.

4.5.3 Offsite Dose Calculation Manual

The ODCM is an offsite emergency planning tool used to project offsite doses in the event of an accident. Typically, the ODCM initial input to the dose projections is the UFSAR dose analysis or the PRA Level 2 source term analyses. However, once information from the actual event becomes available, that current information can be used to refine the offsite dose projections.

Typically, the plant information that is most useful in refining the offsite dose projections is the containment radiation levels as indicated by the containment radiation monitor. This information is used to make projections of offsite dose levels in the event of a failure of the containment integrity. The containment radiation levels, in conjunction with the containment pressure, can also be used to project offsite doses from containment leakage. However, most often the offsite dose measurements are used in place of containment leakage assumptions, since the containment design leakage rate represents a conservative offsite dose projection. Therefore, containment pressure is not important to the ODCM.

The offsite dose projection tools used at most plants also include the capability to use effluent radiation monitor information as input to the dose projections. However, this is typically only used to validate the offsite field survey information, since any radiation releases indicated by effluent monitors would be classified as an ongoing release and the primary input would be from offsite field radiation surveys. Additionally, it is likely that effluent monitors would quickly become saturated in the event of an accident involving any significant fuel damage. Thus, the effluent monitors may not be available to provide information for offsite radiological protection recommendations in the EPIPs.

Therefore, only the containment radiation monitor is useful in refining the offsite dose projections using the ODCM.

Some plants do not rely on plant instrumentation for offsite dose projections and utilize default values contained in the UFSAR for offsite dose projections. For these plants, the containment radiation monitor would not be used for refining offsite dose projections using the ODCM.

4.6 SUMMARY OF INSTRUMENTATION IMPORTANCE

A composite list of PAM instrumentation relied upon in the DBA analysis, the PRA, accident management (EOPs and SAMG), and offsite emergency protective actions was determined based on the assessments discussed above. Table 7 provides a summary of the instrumentation that is relied upon in each of these applications, without making any assessment of the importance of the instrumentation for each application. Each of the instruments identified in Tables 7 and 8 is further assessed as to its importance in accident management and mitigation to determine whether it satisfies Criterion 3 or 4 of 10 CFR 50.36. The importance of the instrumentation will be discussed in Section 5.0 of this report.

Table 7 Significance of PAM Instrumentation Contained in Current Technical Specifications

Instrument	Design Basis Accident	Risk (PRA)	Accident Management		Emergency Plan		
			EOPs	SAMG	EAL	CDA	ODCM
PAM Instrumentation contained in NUREG-1431							
Power Range Neutron Flux		✔	✔		✔		
Source Range Neutron Flux						✔	
RCS Hot Leg Temperature	✔		✔	✔	✔	✔	
RCS Cold Leg Temperature			✔		✔		
RCS Pressure (Wide Range)	✔	✔	✔	✔	✔		
Reactor Vessel Water Level			✔		✔	✔	
Containment Sump Water Level (Wide Range)				✔			
Containment Pressure (Wide Range)		✔	✔	✔	✔		
Containment Isolation Valve Position		✔	✔		✔		
Containment Area Radiation (High Range)			✔		✔	✔	✔
Pressurizer Level	✔	✔	✔		✔		
Steam Generator Water Level (Wide Range)	✔	✔	✔	✔	✔		
Condensate Storage Tank Level							
Core Exit Temperature (Quadrants 1-4)		✔	✔	✔	✔	✔	
Auxiliary Feedwater Flow		✔	✔		✔		

Table 7 (cont.) Significance of PAM Instrumentation Contained in Current Technical Specifications

Instrument	Design Basis Accident	Risk (PRA)	Accident Management		Emergency Plan		
			EOPs	SAMG	EAL	CDA	ODCM
PAM Instrumentation NOT contained in NUREG-1431							
AFW Valve Position							
BAT Level							
Condenser Air Ejector (High Range)							✔
Containment Enclosure Negative Pressure							
Containment Sump Level (Narrow Range)							
Containment Pressure (Narrow Range)							
Intermediate Range Neutron Flux							
Plant Vent Stack (High Range)							✔
PORV Block Valve Position							
PORV Position							
Pressurizer Pressure							
RCS Subcooling Margin	✔	✔	✔				
RWST Level	✔	✔	✔				
Pressurizer Safety Valve Position							
SAT Level							
Spent Fuel Pool Exhaust Radiation							✔
Steam Generator Blowdown Radiation							
Steam Generator Pressure	✔	✔	✔		✔		
Steam Generator Water Level (Narrow Range)	✔	✔	✔	✔	✔		
Steam Line Radiation							✔
TDAFW Pump Exhaust Radiation							✔

Table 8 provides an alternate summary of the potential PAM indications. In this summary, the manner in which the instrumentation is used in the various accident management tools is identified.

Table 8 Summary of Important Indications for Accident Management					
Indication/Purpose	**DBA**	**EOP**	**SAMG**	**PRA**	**E-Plan**
SG Level					
• Diagnose SGTR	✔	✔		✔	
• Maintain SG heat sink	✔	✔		✔	✔
• Prevent SG overfill	✔	✔		✔	
• Initiate Bleed and Feed		✔		✔	
• Scrub Fission Products for SGTR			✔		
SG Pressure					
• Diagnose secondary side break or stuck open relief valve	✔	✔		✔	
• Cooldown target for RCS depressurization SGTR	✔	✔		✔	
RCS Pressure					
• Cooldown target for RCS depressurization	✔	✔		✔	
• High Pressure Melt Ejection prevention			✔	✔	
• Maintain cooldown rate	✔	✔		✔	
• RCS Integrity					✔
RCS Subcooling					
• Maintain subcooling during RCS cooldown and depressurization	✔	✔		✔	
• SI Termination	✔	✔		✔	
Pressurizer Level					
• SI termination to prevent pressurizer overfill	✔	✔		✔	
Core Temperature					
• Diagnose inadequate core cooling		✔	✔	✔	✔
Neutron Flux					
• Diagnose subcriticality		✔		✔	✔
Containment Pressure					
• Diagnose inadequate containment cooling		✔	✔	✔	✔
Containment Radiation					
• Diagnose core damage					✔
Containment Isolation Valve Position					
• Diagnose unisolated containment				✔	✔
RWST Level					
• Diagnose RWST refill				✔	
High Head SI Flow					
• Diagnose manual SI				✔	
Auxiliary Feedwater Flow					
• Diagnose loss of heat sink				✔	
Service Water Flow Rate System Availability					
• Diagnose loss of Service Water				✔	
Component Cooling System Availability					
• Diagnose loss of component cooling				✔	

5 INSTRUMENTATION IMPORTANCE

The importance of the PAM instrumentation to plant safety should bear a direct relationship to the criteria in 10 CFR 50.36 (c)(2)(ii) and the Regulatory Guide 1.97 classification of the instrumentation. The importance of the instrumentation that is used in plant safety assessments and tools (i.e., identified in Table 7 and Table 8) was further evaluated to determine whether it satisfies the 10 CFR 50.36 criteria and to determine the applicable Regulatory Guide 1.97 classification with respect to its inclusion in the Technical Specifications. As noted previously, the original classification in Regulatory Guide 1.97 was done based on information and knowledge available in the early 1980's. This assessment is based on the information and knowledge currently available and can therefore be used to revise the original bases for the Regulatory Guide 1.97 classifications.

The assessment described in this section of the report focuses on the instrumentation that is relied upon in plant safety analyses, accident management and offsite protective actions. That is, each of the instruments identified in Tables 7 and 8 is further assessed as to its importance in accident management and mitigation to determine whether it satisfies Criterion 3 or 4 of 10 CFR 50.36. No further assessment is required for any instrumentation that is not relied upon in the safety assessments in this report. However, a brief discussion is merited on several of the current PAM instrumentation that are not considered to be significant for plant safety and is included at the end of the discussion of the primary instrumentation included in the DBA analysis and accident management.

5.1 INSTRUMENTATION RELIED UPON TO MITIGATE ACCIDENTS

This section provides a discussion of the results of an evaluation of the importance of instrumentation relied upon to mitigate accidents. The evaluation uses the screening criteria defined in Section 3.2 of this report. The results of the evaluation are expressed in terms of whether any of the 10 CFR 50.36 (c)(2)(ii) criteria (Criterion 3 and/or 4) are met. The recommended Regulatory Guide 1.97 classification of the instrumentation for the purpose of determining whether it should be included in the Technical Specifications is also presented.

Power Range Neutron Flux

The power range neutron flux indication provides the most direct indication of reactor criticality. The power range neutron flux instrumentation provides this indication for events in which subcriticality is not initially achieved. The intermediate range and source range neutron flux instrumentation provide an indication of sustained subcriticality, such as during and following RCS depressurization.

The Westinghouse NSSS plant PRA survey contained in Appendix A shows that power range neutron flux is a key indication for accident management operator actions to initiate manual reactor trip to bring the reactor to a subcritical condition. Subsequent operator actions to assure that the reactor remains in subcritical state, such as during and following RCS depressurization, were not determined to be important for long term core cooling. Therefore, the intermediate range and source range indications are not identified as key instruments in this assessment. Additionally, EALs in the E-Plan typically utilize the power range neutron flux as an indication of a potential loss of a fission product barrier in the assessment of the declaration of a General Emergency level and the potential need for offsite radiological protection actions. Therefore, the power range neutron flux indication meets Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The power range neutron flux indication is a Type B variable since it provides information to verify the automatic actuation of Reactor Protection System (RPS). The power range neutron flux indication is a Category 1 variable since it provides direct information to verify accomplishment of the subcriticality safety function.

Source Range Neutron Flux

After subcriticality is achieved, the source range neutron flux monitor can be used to confirm continued subcriticality by monitoring the startup rate. A positive startup rate indicates that criticality is being approached. The source range neutron flux indication can be used as a backup to the power range neutron flux indication during shutdown to determine whether sufficient negative reactivity (e.g., boron, RCS temperature during RCS cooldown) is available for long term subcriticality. Since this source range neutron flux does not provide an indication for operator actions for which no automatic control is provided and is not important from a risk perspective, it does not meet either Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii) and should not be included in the Technical Specifications.

The source range neutron flux indication provides a verification of the automatic actuation of the RPS and is therefore a Type B variable. The source range neutron flux indication provides diagnostics for maintaining subcriticality during an RCS cooldown and depressurization and is therefore a Category 3 variable.

RCS Hot Leg Temperature (Wide Range)

The RCS hot leg Temperature indication provides information to indicate the temperature of the RCS hot leg fluid. It can be used by the plant operators to verify adequate core cooling, RCS subcooling, and in conjunction with the RCS cold leg temperature indication, the effectiveness of RCS heat removal by the secondary system. However, it is not the primary indication used by the plant operators for any of those determinations. Since the RCS hot leg temperature wide range indication does not provide an indication for operator actions for which no automatic control is provided and 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 Technical Specifications.

The RCS hot leg temperature indication provides information to indicate whether the core cooling safety function is being accomplished and is therefore a Type B variable. The RCS hot leg temperature wide range indication is a backup to the CETs for indicating that the core cooling safety function is being accomplished, and is therefore a Category 3 variable.

RCS Cold Leg Temperature (Wide Range)

The RCS cold leg temperature indication provides information to indicate the temperature of the RCS cold leg fluid. It can be used by the plant operators, in conjunction with the RCS hot leg temperature indication, to verify the effectiveness of RCS heat removal by the secondary system. However, it is not the primary indication used by the plant operators for that determination. Since the RCS cold leg temperature wide range indication does not provide an indication for operator actions for which no automatic control is provided and 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 Technical Specifications.

The RCS cold leg temperature indication provides information to indicate whether the core heat removal safety function is being accomplished and is therefore a 1.97 Type B variable. The RCS cold leg temperature wide range indication is a diagnostic indication and is therefore a Category 3 variable.

RCS Pressure (Wide Range)

The RCS pressure indication is used for all accident sequences. There is no other indication that can be used to directly indicate RCS pressure over the range of pressure required for accident management. Operator actions for a cooldown target for RCS depressurization and for maintaining subcooling (a combination of RCS pressure and temperature) during RCS cooldown and depressurization and for SI termination are performed using the RCS pressure indication. RCS subcooling, which utilizes RCS wide range pressure is also used as a backup for diagnosis of an inadequate core cooling condition. Also, operator actions in the EOPs and SAMG utilize the RCS pressure indication to diagnose the need to depressurize the RCS to minimize the potential for containment integrity challenges from a high pressure melt ejection and to mitigate SGTR fission product releases that bypass the containment.

The DBA analyses indicate that RCS cooldown and depressurization for a SGTR accident, to below the SG pressure in the ruptured SG to terminate break flow, and for SI termination to prevent pressurizer overfill, are operator actions for which no automatic control is provided. Therefore, RCS pressure wide range satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii). Additionally, the PRA shows that RCS depressurization to terminate break flow for an SGTR event and depressurization of the RCS after core damage to prevent a high pressure melt ejection (see Appendix A) that could challenge containment integrity are risk significant operator actions. Therefore, RCS pressure wide range also satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The RCS pressure wide range indication is a Type A variable since it provides information for operator action for SGTR break flow termination for which no automatic control is provided. RCS pressure wide range is a Category 1 variable, since, together with SG pressure, provides information to verify that break flow through a ruptured SG tube is terminated, thereby satisfying the inventory safety function.

Reactor Vessel Water Level

The reactor vessel water level indication is used in the plant EOPS as an indication of inadequate core cooling and as an indication of the potential for void formation that can interfere with natural circulation cooling. Some reactor vessel water level instrumentation only measures upper head voiding, versus others that measure well into the core region and therefore it only serves as a backup indication of upper head voiding and an indication of potential core uncovery. Since neither of these indication functions provide an indication for operator actions for which no automatic control is provided and the indication 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 Technical Specifications.

The reactor vessel water level indication provides information to indicate whether the core cooling safety function is being accomplished and is therefore a Type B variable. The reactor vessel water level indication is a backup to the CETs for identifying an inadequate core cooling condition and is therefore a Category 3 variable.

Containment Sump Water Level (Narrow Range)

The containment sump water level indication provides information to indicate whether sufficient water is available in the containment sump at the time ECC is transferred from the injection mode to the recirculation mode, and when the recirculation spray system is automatically started for subatmospheric containments. It also provides an indication of excessive containment sump water levels that could result in flooding of key equipment and instrumentation. ECC injection (from the RWST) is switched over to recirculation (from the sump) to provide long term ECC when the RWST is emptied. The required operator actions associated with switchover to recirculation are plant specific, with some plants having fully automatic switchover, some having semi-automatic switchover, and some having totally manual switchover. The switchover to recirculation is initiated based on RWST level. For all DBA and for all accidents analyzed in the PRA where the RCS inventory loss is inside containment, the design of the plant ensures that there will be adequate water in the containment sump to support switchover to recirculation. Therefore, no operator action is required in the design basis analyses based on containment sump level. Since the containment sump water level narrow range indication does not provide an indication for operator actions for which no automatic control is provided, it does not satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

Containment water level instrumentation is used in the EOPs to define the loss of ECC recirculation capability. For diagnosis of a LOCA inside containment, the containment sump water level is used in the EOPs as the third indication (after the containment pressure and containment radiation indications). Additionally, for diagnosis of a LOCA outside containment, the containment sump water level indication is the third indication utilized in the EOPs (after a decreasing RCS pressure indication and increasing auxiliary building radiation indication). The most probable LOCA outside containment, according to PRA studies, is the interfacing system LOCA outside containment in the low pressure RHR piping connected to the RCS. This would likely result in a containment sump water level indication, since the flow from the relief valves on the RHR piping is routed back to the pressurizer quench tank whose rupture disk would quickly open and release fluid to the containment. Thus "no containment sump level" is not a reliable indicator of a LOCA outside containment. In summary, diagnosis of RCS pressure boundary integrity as a critical safety function in the EOP functional restoration guidelines does not rely on containment water level indication. It is used in the SAMG to assure that adequate water is available in the containment sump(s) for ECC recirculation, should the capability to inject into the RCS from the containment sump become available.

In the PRA models, operator actions to refill the RWST based on inadequate containment sump level for continued core cooling are typically modeled for SGTR and LOCAs outside of containment. The risk importance of RWST refill identified in Appendix A shows that it can have a high risk importance for some plants, although the median value might not indicate a high risk importance. As PRAs are updated to more closely model the expected accident management strategies (as opposed to more conservative models), the RWST refill for these events may become more risk important. However, containment sump level would not be used as the primary indication for the need to begin RWST refill. There are a number of other indications available to provide information that RWST refill would be required for long term core cooling for these accidents. The accident type alone (e.g., SGTR or LOCA outside containment) and the current RWST level would be sufficient to provide an indication that long term core cooling using recirculation is not an available accident management strategy.

Although RWST refill may be risk important, containment sump water level is not the primary indicator of the need for operator action to begin RWST refill. Therefore, it is concluded that the containment sump water level narrow range indication does not provide an indication for operator actions which are important to mitigating core damage or containment releases and therefore it does not satisfy Criterion 4 of 10 CFR 50.36 (c)(2)(ii). Since containment sump water level narrow range does not satisfy either Criterion 3 or 4 of 10 CFR 50. 36(c)(2)(ii), it should not be included in the Technical Specifications.

The containment sump water level indication provides information to indicate whether the core cooling safety function can be accomplished when the Emergency Core Cooling System switchover to the recirculation mode of operation occurs, and is therefore a Type B variable. The containment sump water level narrow range indication provides information on the status of SI from the RWST and is therefore a Category 2 variable.

Containment Pressure (Wide Range)

The containment pressure indication provides information for assessing an inadequate containment cooling condition and for determining the potential challenge to the containment pressure retaining integrity. The wide range containment pressure instrumentation provides an adequate range and sensitivity for this purpose. Other containment pressure instrumentation does not extend beyond the design basis pressure and therefore does not have sufficient range to provide the indication of an imminent containment integrity challenge due to overpressurization. This instrumentation is only used in the EOPs to define the potential for a challenge to containment integrity due to overpressurization. If containment heat removal systems are functioning properly, no challenge to containment integrity should occur due to containment pressure. It is used in the SAMG to indicate a possible containment integrity challenge and to initiate the assessment of containment venting strategies. It is used in the EALs to indicate the potential for the loss of the containment fission product barrier. It can be used in the OCDM as input to offsite dose projections from containment leakage after a core damage accident. In the PRA models, operator actions to control containment pressure are not typically modeled. Therefore, the importance in the PRA cannot be established. If it were modeled in the PRA, it would not impact the core damage frequency or the large early release frequency. It is only the late containment failure frequency that would be impacted.

The containment pressure indication is used as an indicator of the potential loss of a fission product barrier in the EALs in the E-Plan. Containment pressure is a key indicator in the declaration of a General Emergency level and the potential need for offsite radiological protection actions. Therefore, the containment pressure wide range instrumentation satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The containment pressure wide range indication provides information to identify a fission product barrier challenge and is therefore a Type C variable. It provides direct verification of containment cooling to maintain the containment fission product barrier safety function and is therefore a Category 1 variable.

Penetration Flow Path Containment Isolation Valve Position

The containment isolation valve position indication provides a direct indication of a failure to completely isolate containment following the receipt of a containment isolation signal. Each motor operated isolation valve has an indication lamp on the main control board that is lit based on the isolation valve limit switch position. This instrumentation is used in the EOPs to assure that automatic containment isolation has occurred. It is used in the EALs to indicate the loss of the containment fission product barrier. It may also be used in some OCDM models to project offsite doses from a loss of containment isolation during a core damage accident. In the PRA models, operator actions to manually isolate the containment are typically modeled in the Level 2 PRA. The importance in the PRA cannot be determined because it does not impact core damage frequency. However, failure of containment isolation can be a major contributor to LERF. If a Risk Achievement Worth (RAW) indicator for LERF for Westinghouse NSSS plants were available, it is likely that operator actions to isolate the containment upon failure of automatic isolation would be risk significant for some Westinghouse NSSS plants. Since the containment isolation valve position indication is important in the PRA and is utilized in the E-Plan, the containment isolation valve position indication satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The containment isolation valve position indication provides verification of automatic actuation of Phase A and Phase B containment isolation and is therefore a Type B variable. Since it provides a direct verification of containment isolation to maintain the containment fission product barrier safety function, it is a Category 1 variable.

Containment Area Radiation (High Range)

The containment area high range radiation monitors provide an indication of a loss of one or more fission product barriers (fuel rod cladding or RCS barrier). Other containment radiation instrumentation is available to indicate radiation levels during normal plant operation or to provide indication of fission product particulates.

EALs in the E-Plan utilize the containment area high range radiation monitor as an indication of a potential loss of one or more fission product barriers in the assessment of the declaration of a General Emergency level and the potential need for offsite radiological protection actions. CDA also uses the containment area high range radiation monitor as an input to the determination of core damage. The containment area high range radiation monitor provides an adequate range and sensitivity for determination of core damage. Other containment radiation instrumentation does not have the desired range or sensitivity. Since it is used in determining the need for offsite radiological protection activities, the containment area high range radiation monitor satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The containment area high range radiation monitor provides information to identify a fission product barrier challenge and is therefore a Type C variable. It also provides direct verification of the core cooling safety function and is therefore a Category 1 variable.

Containment Hydrogen Monitors

This instrument is not addressed in this assessment, since it was included in the rulemaking for 10 CFR 50.44. The statement of considerations for the 50.44 rulemaking (Reference 17) states that this instrument can be relocated from the Technical Specifications and can be re-classified as Type C, Category 3 per the Regulatory Guide 1.97 definitions.

Pressurizer Level

The primary purpose of the pressurizer level indication following an accident is for the SI termination criteria to prevent pressurizer overfill. If SI termination is not accomplished before pressurizer overfill, a small LOCA condition results due to the opening of the pressurizer power operated relief valves (PORVs). This is the only instrument that provides this indication. The design basis analysis assessment indicates that SI termination in the event of a SGTR ~~or~~ SLB is required for long term core cooling and is an operator action for which no automatic control is provided. It is also identified as an implicit assumption in other DBA analyses. Therefore, this indication satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii). The PRA assessment also indicates that SI termination in the event of a SGTR is required for long term core cooling. Therefore, this indication satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The pressurizer level indication provides primary information needed to permit the operators to take specified manual actions to terminate SI and is therefore a Type A variable. It also provides information related to satisfying the RCS inventory safety function to permit SI termination and is therefore a Category 1 variable.

Steam Generator Level (Wide Range)

SG level indication can be provided by either the narrow range or the wide range SG level instrumentation. Operator actions for diagnosis of a SGTR, maintenance of adequate SG level to provide a heat sink, controlling SG level to prevent SG overfill, and covering the tubes to scrub fission products for an SGTR are performed utilizing the SG narrow range level indication. However, the wide range SG indication encompasses the narrow range span and can be used in level ranges where the narrow range SG level indication is not available. The operators are trained in the use of wide range SG level indication, as well as the narrow range SG level indication. The initiation of bleed and feed can only be performed based on wide range SG level indication, since the narrow range SG level indication does not have sufficient range to enable the diagnosis of the need to initiate bleed and feed cooling, which is at a very low SG water level. Therefore, the SG level indication that can provide indication for all of the important operator actions is the wide range SG level instrumentation.

The design basis analyses assume that controlling SG level for long term core cooling and using SG level for the diagnosis of a SGTR are operator actions for which no automatic control is provided. Therefore, the SG wide range level indication satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii). The PRA shows that the initiation of bleed and feed core cooling, as well as the design basis functions, are risk significant operator actions. Also, the SAMG assessment shows that maintaining the water level over a ruptured SG tube is a risk significant operator action. Therefore, SG wide range level indication also satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The SG wide range level indication provides information for operator action to maintain a heat sink following a DBA for which no automatic control is provided and is therefore a Type A variable. It also provides the direct verification of satisfying the heat sink safety function and is therefore a Category 1 variable.

Condensate Storage Tank Level

This instrumentation is used in the EOPs to define the potential loss of the SG heat sink due to low tank inventory as a continued water supply for the AFW system. Therefore, the CST level instrumentation is the primary indication of the ability to continue AFW flow to the steam generators to provide a secondary side heat sink for decay heat removal from the reactor core. CST refill is a long-term action that is not credited in the UFSAR analyses; CST refill is typically not required in the first 16 to 20 hours after an accident. For the design basis events, the plant would either be on normal RHR cooling (for non-LOCA events) or ECC recirculation in a time frame well before the CST inventory is exhausted. In the PRA models, operator actions to refill the CST based on low CST level indication are modeled in some PRAs. The results of the PRA assessment in Appendix A show that the CST level indication has low risk significance. The low risk significance is based on: a) the low probability that natural circulation would be the only means of maintaining the core in a safe stable state following an accident, and b) the long time to deplete the CST during which CST refill would have begun and been monitored based on available CST level indications (including local indications), if long term natural circulation decay heat removal was selected as the long term safe stable state. Therefore, the CST level indication does not satisfy either Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii) and should not be included in Technical Specifications.

The CST level control room indication provides information to indicate whether a continued SG heat sink can be maintained and is therefore a Type B variable. The CST level indication provides information for the long term AFW system operating status and is therefore a Category 2 variable.

Core Exit Temperature

The core exit thermocouples provide the most direct measurement of the core temperature and the highest RCS fluid temperature since they are located in the core outlet region of the reactor vessel. The primary purpose of core temperature indication is to provide input to the RCS subcooling calculation and for the diagnosis of an inadequate core cooling condition.

The design basis analyses indicate that the determination of RCS subcooling is required to support the RCS cooldown and depressurization for an SGTR to terminate break flow, which is an operator action for which no automatic control is provided. The RCS subcooling indication is required for the successful completion of this operator action. The determination of RCS subcooling is also identified as an implicit assumption in other DBA analyses. Therefore, the CET indication, which is an input to RCS subcooling, satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

The composite PRA database in Appendix A shows that RCS subcooling indication, which is required to support RCS cooldown and depressurization to terminate break flow for an SGTR event is a risk significant operator action. In addition, the SAMG uses CET temperature as a transition from the EOPs to the SAMG where several risk significant operator actions, including RCS depressurization to prevent a containment challenge from High Pressure Melt Injection (HPME), have been identified. Also, EALs in the E-Plan utilize the CET temperature as an indication for the potential loss of a fission product barrier which is important in the assessment of the declaration of a General Emergency level and the potential need for offsite radiological protection actions. Therefore, the CET indication satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The CET indication provides RCS subcooling information to the operators for the initiation of RCS cooldown and depressurization and is therefore a Type A variable. It also provides a direct verification of satisfying the core cooling safety function and is therefore a Category 1 variable.

Auxiliary Feedwater Flow

The loss of main feedwater is a design basis event whose consequences are analyzed and documented in the UFSAR. It is assumed that the auxiliary feedwater will automatically start and provide inventory to maintain a heat sink. The loss of all feedwater (main feedwater and auxiliary feedwater) not a design basis event and operator actions to re-establish feedwater sources are not modeled in the design basis analyses. Thus, the auxiliary feedwater flow rate indication does not meet Criterion 3 of 50.36 (c)(2)(ii).

This instrumentation is used in the EOPs to define the potential loss of heat sink and the need to either establish another SG feed source or to initiate core cooling via RCS bleed and feed. It is used in the EALs to indicate the potential for the loss of the fuel rod cladding fission product barrier. In the PRA models, operator actions to establish bleed and feed core cooling are modeled based on the SG wide range level indication as opposed to a loss of AFW flow. Operator actions to establish an alternate feedwater source or to re-align AFW are modeled based on an inadequate AFW flow and a decreasing SG water level as indicated by the AFW flow rate and SG water level. The PRA assessment in Appendix A indicates that the AFW flow rate may be risk significant. These operator actions to re-align feedwater sources would be taken only if a decreasing SG water level were observed. Thus, AFW flow rate is not essential to successful core cooling because a decreasing SG water level would trigger the same actions. However, in light of the PRA success criteria for operator actions, the AFW flow rate provides a rapid indication of the need for further actions and the reliance on a decreasing SG water level may impact the probability of success of these operator actions. The AFW flow rate indication is the basis for a risk important operator action in the PRA and therefore satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The AFW flow rate indication provides information used for the verification of the automatic actuation of AFW and is therefore a Type B variable. It provides the direct verification of satisfying the heat sink safety function and is therefore a Category 1 variable.

Containment Sump Water Level (Wide Range)

Containment Sump Water Level (Wide Range) and Containment Water Level are used interchangeably in this report. This instrumentation is used in the EOPs to define containment flooding. Although it is not used in the EOPs, the containment wide range water level could be used in the EOPs to verify that the contents of the RWST were emptied into the containment if the RWST level instrumentation were unavailable. The design basis containment water level is established to assure that no important components are submerged during a DBA. It is used in the SAMG to indicate the desired containment water level after core damage has occurred. In the SAMG, a backup method of containment water level is provided (Computational Aid, CA-4) based on the potential for the post core damage environment to render the containment water level instrumentation unavailable. The PRA does not model any operator actions related to containment water level using the wide range indications. Thus, it is concluded that the wide range containment water level instrumentation is not risk significant. Since there are no operator actions in the design basis analyses based on containment wide range water level, this indication does not satisfy Criterion 3 of 10 FR 50.36 (c)(2)(ii). The containment wide range water level indication does not support any risk important operator actions in the PRA and therefore this indication does not satisfy Criterion 4 of 10 FR 50.36 (c)(2)(ii).

The containment wide range water level indication provides information to indicate whether the core cooling safety function can be accomplished when the Emergency Core Cooling System (ECCS) switchover to the recirculation mode of operation occurs, and is therefore a Type B variable. The containment wide range water level indication provides information on the status of SI from the RWST and is therefore a Category 2 variable.

Pressurizer Pressure

The pressurizer pressure is only specifically used in the EALs to determine the potential for loss of the RCS fission product barrier. It can be used elsewhere as a backup to the RCS pressure indication, but the range of the pressurizer pressure indication is very limited. Therefore, it is not typically used in the EOPs. This instrumentation is not considered to be risk significant. The pressurizer pressure indication does not support any risk important operator actions in the PRA and does not support any operator actions in the design basis analyses. Therefore this indication does not meet Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii).

Because of its limited application, it is recommended that this instrumentation not be classified by the Regulatory Guide 1.97 definitions.

RCS Radiation Level

The RCS radiation level indication is typically provided by RCS letdown radiation monitors. These monitors are located in the letdown line, which is isolated upon the receipt of a SI signal. As discussed in WCAP-14696-A (Reference 16), the reactor coolant radiation level is only important for DBAs where there is fuel rod cladding damage without coincident core overheating, such as local reactivity events caused by the withdrawal of a single Rod Control Cluster Assembly (RCCA). For these events, the reactor is tripped and shutdown by the RPS. The letdown radiation monitor indication would be used by the plant operators to decide whether the declaration of an Unusual Event condition was appropriate. However, this determination is not shown to be important to risk in PRAs. Since the letdown radiation monitor 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 Technical Specification.

The RCS radiation level indication provides information to indicate the potential for a breach of the fuel cladding fission product barrier and is therefore a Type C variable. The RCS radiation level indication provides diagnostic indications of core damage not associated with core overheating and is therefore a Category 3 variable.

RCS Subcooling Monitor

The RCS subcooling margin indication provides information to the operators related to satisfying one of the SI termination criteria following a steam line break or SGTR accident. The inputs to the RCS subcooling monitor are the CETs for RCS temperature and the wide range RCS pressure indication for RCS pressure. Since both of these indications are independently displayed in the control room and are also included in the Technical Specifications based on satisfying Criterion 3 and 4 of 10 CFR 50.36 (c)(2)(ii), the subcooling monitor provides a verification of the other indications. Therefore, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii) and should not be included in the Technical Specifications.

The RCS subcooling margin indication provides information to indicate whether the core cooling safety function is being accomplished, and is therefore considered a Type B variable. The RCS subcooling indication is a backup to the CETs and RCS pressure, and is therefore considered a Category 3 variable.

RWST Level (Wide Range)

This instrumentation is used in the design basis analyses to indicate the point at which transfer to ECC recirculation and containment spray recirculation should be initiated. The operator actions required for transfer to ECC and containment spray recirculation are typically cued based on the RWST low and low-low level alarms, as opposed to the RWST level instrumentation itself. RWST level instrumentation only validates the alarm and does not provide the primary indication for operator actions. The required operator actions associated with switchover to recirculation are plant specific, with some plants having fully automatic switchover, some having semi-automatic switchover, and some having totally manual switchover. Transfer to ECC recirculation was found to be a risk significant operator action in the PRA as discussed in Appendix A. While the PRA typically also models transfer to containment spray recirculation as an action to continue containment heat removal, risk importance measures are not available since they only potentially impact late containment failure probability and not core damage frequency. While the risk importance of transfer to containment spray recirculation would be greater for plants without safety related fan cooler units for containment heat removal, it does not impact the conclusions, since containment heat removal via containment spray also requires the ECC recirculation heat exchanger to be in operation. Since the operator action is taken based on the RWST low level and low-low level alarms, the RWST level instrumentation is not risk significant and is only used to validate the alarm function.

In addition, the RWST level instrumentation provides an indication of the need to initiate make-up to the RWST to maintain long term cooling. The PRA assessment in Appendix A shows that make-up to the RWST to provide long term core cooling for the SGTR and interfacing system LOCA accidents are risk significant operator actions that are keyed from the RWST level instrumentation. For all other accident sequences modeled in the PRA, refilling the RWST is typically not modeled in the PRA since it is only a remedial action; the RWST can only be refilled for a finite number of times and other actions also not modeled in the PRA would be required to achieve a safe stable state. Therefore, the RWST Level instrumentation satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The RWST Level indication as used here refers to the wide range indication as opposed to the narrow range indication that is only used as a level indication associated with the Technical Specifications minimum required RWST level.

The RWST wide range level indication provides information to indicate the continued operation of SI for continued inventory control and is therefore a Type D variable. It also provides information to indicate the need to refill the RWST to continue inventory control for SGTR and Interfacing System Loss of Coolant Accident (ISLOCA) events and is therefore a Category 1 variable.

Steam Generator Pressure

This instrument is used in design basis analyses and EOPs to indicate a loss of secondary side coolant accident (a main steamline or feedline break). It is also used in the design basis and EOP analyses for the SGTR accident to indicate the termination of the reactor coolant loss through the ruptured SG tube. Therefore, the SG pressure indication satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii). SG pressure is modeled in the PRA operator actions to terminate the break flow through a ruptured SG tube by depressurizing the RCS to a point just below the SG pressure, per the plant EOPs. Therefore, the SG pressure indication satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).,

The SG pressure indication provides information for operator action for SGTR break flow termination for which no automatic control is provided and is therefore a Type A variable. Together with RCS pressure, the SG pressure indication provides information to verify that break flow through a ruptured SG tube is terminated, thereby satisfying the inventory safety function and is therefore a Category 1 variable.

Steam Generator Narrow Range Level

This instrumentation is used in the design basis analyses and the EOPs to determine the SG level required for an effective heat sink, and as a primary means of diagnosing an event. It is used in the SAMG to indicate an effective heat sink for core cooling recovery. It is used in the EALs as an indicator of the potential loss of the RCS fission product barrier. In the PRA models, operator actions to diagnose the SGTR event are typically modeled in the same operator action to isolate the ruptured SG secondary side and terminate AFW to the ruptured SG. The SG narrow range level is also implicitly used in the PRA for two additional actions: maintaining SG water level to provide an effective heat sink and terminating feedwater flow to prevent SG overfill for tube rupture events. The PRA assessment in Appendix A indicates that SG level is an indication required for risk significant operator actions. Since the SG narrow range level indication provides an indication for operator actions for design basis accidents for which no automatic actuation is provided and it is important from a risk perspective, it satisfies both Criterion 3 and 4 of 10 CFR 50.36 (c)(2)(ii). However, the SG wide range level indication can also be used for this purpose and the plant operators are trained in this application. Since the SG wide range indication has been determined to also meet both Criterion 3 and 4 of 10 CFR 50.36 (c)(2)(ii) for the indication of a loss of SG heat sink and it can also serve to indicate SG level for effective heat removal (for which the SG narrow range indication cannot), it is the preferred SG level indication to be included in the Technical Specifications. It is also proposed that the SG narrow range level indication not be included in the Technical Specifications.

The SG level narrow range indication provides information to indicate whether the SG heat sink safety function is being accomplished and is therefore a Type B variable. The SG level narrow range indication provides information on the status of SG feedwater delivery and is therefore a Category 2 variable.

High Head SI Flow

This instrumentation is used in the EOPs to diagnose the need for manual actuation for either the low head or high head SI functions. In the PRA models, operator actions to manually start SI are typically modeled as an operator action in the event of failure of the automatic actuation system. The PRA assessment in Appendix A indicates that this action is risk significant. A review of PRA success criteria shows that only high head SI pumps are required for successful core cooling for all events except a double-ended guillotine LOCA (operator action to manually initiate SI is not modeled in the large LOCA event because of the short time available for success). Therefore the requirement for ECC flow indication should only apply to the high head SI pumps. High Head SI flow rate provides the basis for a risk significant operator action in the PRA and therefore satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The High Head SI flow rate indication provides information for the verification of automatic actuation of SI and is therefore a Type B variable. The High Head SI flow rate indication provides direct information to verify the operation of SI to maintain the inventory safety function for core cooling and is therefore a Category 1 variable.

Pressurizer PORV Position Indication

The PORV Limit Switch Position Indicators provide information to the control room operators related to the position of the pressurizer PORVs. It could be used to diagnose a high RCS pressure or a stuck open PORV (LOCA) at lower RCS pressures. The DBA analysis of an inadvertent opening of the PORV does not rely on operator diagnosis and closure of the PORV or block valve; the DBA analysis assumes that automatic safety injection actuation will provide adequate protection. Since the PORV Limit Switch 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.

The PORV Limit Switch Position Indication provides information to indicate the status (position) of the pressurizer PORVs, which are one means for identifying RCS depressurization, and is therefore a Type D variable. The PORV Limit Switch Position Indication provides information on the status of the pressurizer PORVs for RCS integrity and is therefore a Category 2 variable.

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 be 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 Anticipated Transient Without Scram (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.

5.2 OTHER IMPORTANT INSTRUMENTATION

The PRA results from the Westinghouse NSSS plant survey (Appendix A) indicate that several other instruments may be important to risk. These instruments are discussed below to determine whether they should be included in the Technical Specifications.

Component Cooling Water Temperature and Flow

The loss of all Component Cooling Water capability is a beyond design basis event since no single failure can disable the entire system function to support key design basis functions, such as long term decay heat removal after a loss of coolant accident. In the event of a loss of all Component Cooling Water (CCW), the pumps required to perform a safety function (e.g., charging pumps) may be inoperable. In addition, the loss of cooling to the reactor coolant pump seals will result in pump seal leakage that depletes reactor coolant inventory. The PRA assessment in Appendix A indicates that operator actions to restore CCW action may be risk significant. These actions are in response to a loss of CCW system availability as an initiating event. Since the CCW system is a normally operating system, the sudden unavailability of the system would be indicated by a wide variety of instrumentation. Even though the implementation of reactor coolant pump trip is an important operator action, a single set of instrumentation is not key to the diagnosis of a loss of all CCW. Since there are numerous indications of a loss of CCW capability (e.g., CCW pressure, CCW temperature, CCW surge tank level, and CCW flow), no single instrument is critical to the diagnosis of the loss of CCW. Since the complete loss of CCW is not a design basis event, the instrumentation to support diagnosis and recovery of CCW does not meet Criterion 3 of 10 CFR 50.36 (c)(2)(ii). Since no single instrument is critical to the diagnosis of the loss of CCW, the CCW temperature and flow indications do not satisfy Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

Service Water Temperature and Flow

The loss of all Component Cooling Water capability is a beyond design basis event since no single failure can disable the entire system function to support key design basis functions, such as long term decay heat removal after a loss of coolant accident. In the event of a loss of all Service Water (SW), CCW may also be lost (not all plants have the normal SW/CCW dependency), which impacts the safety systems discussed above. The PRA assessment in Appendix A indicates that operator actions to restore SW may be risk significant. These actions are in response to a loss of SW system availability as an initiating event. Since the SW system is a normally operating system, the sudden unavailability of the system would be indicated by a wide variety of instrumentation. Since there are numerous indications of a loss of SW capability (e.g., SW temperature, SW flow, CCW pressure, CCW temperature, CCW surge tank level, and CCW flow), no single instrument is critical to the diagnosis of the loss of SW. Since the complete loss of SW is not a design basis event, the instrumentation to support diagnosis and recovery of SW does not meet Criterion 3 of 10 CFR 50.36 (c)(2)(ii). Since no single instrument is critical to the diagnosis of the loss of SW, the SW temperature and flow indications do not satisfy Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

5.3 PAM INSTRUMENTATION TO BE INCLUDED IN TECHNICAL SPECIFICATIONS

The evaluations performed in this report to determine the appropriate Regulatory Guide 1.97 classifications are for the purpose of determining the most appropriate PAM instrumentation to be included in the Technical Specifications, as well as the appropriate classifications for the PAM instrumentation proposed to be relocated from the Technical Specifications to LCDs.

Table 9 provides a summary of the PAM instrumentation that is currently contained in the Technical Specification 3.3.3 of NUREG-1431, Rev. 3, plant specific Technical Specifications for Westinghouse NSSS plants that have not converted to NUREG-1431, as well as the PAM instrumentation recommended to be included in the Technical Specifications. This recommended list of PAM instrumentation is based on the technical assessment as discussed in the previous sections.

The PAM instrumentation that was shown to have safety significance as determined by satisfying 10 CFR 50.36 (c)(2)(ii) Criterion 3 and/or 4 is shown in Table 10. All other PAM instrumentation does not satisfy Criterion 3 or Criterion 4 of 10 CFR 50.36 (c)(2)(ii) and can therefore be relocated from the Technical Specifications.

The technical assessment provided in this report evaluates the PAM instrumentation contained in NUREG-1431 and also contained in plant specific Technical Specifications for plants that have not converted to NUREG-1431 against each of the criteria in 10 CFR 50.36 (c)(2)(ii). More specifically, this report provides a basis for showing compliance with Criterion 4 of 10 CFR 50.36 in that the importance of instrumentation results from PRA studies at each of the Westinghouse NSSS plants was considered in detail. The PRA studies provide a basis for deciding which instruments should be included the Technical Specifications. The total plant accident management response, including the PRA insights, provides a basis for the relocation of a number of PAM instrumentation from the NUREG-1431. On the other hand, the PRA and other accident management tools used at the plant (e.g., the EALs in the E-Plan) also provides a basis for the inclusion of two PAM instruments not previously included in Technical Specification 3.3.3 of NUREG-1431; RWST Level indication, and the High Head SI Flow Rate indication.

The NRC comments on the use of risk assessment (Reference 4) i.e., to the non-Type A, Category 1 variables were also considered in this assessment. Finally, the use of instrumentation in all plant accident management activities (e.g., SAMG and the E-Plan) was also included in the development of the recommended PAM instrumentation to be included in the Technical Specifications.

The results of the evaluation performed in this report justify the relocation of the Category 1, non-Type A Instruments from the Technical Specifications to licensee controlled documents, consistent with the NRC statement in their letter (Reference 4) regarding the Category 1, non-Type A instruments. Additionally, this evaluation also provides a technical basis for the relocation from the Technical Specifications of the some Type A PAM instruments based on a current knowledge and the application of Criterion 3 and 4 of 10 CFR 50.36.

Table 9 Comparison of PAM Instrumentation			
Instrument	**Plant Specific Technical Specifications**	**NUREG-1431**	**WCAP-15981**
Power Range Neutron Flux	✔	✔	✔
RCS Pressure (Wide Range)	✔	✔	✔
Containment Pressure (Wide Range)	✔	✔	✔
Containment Isolation Valve Position	✔	✔	✔
Steam Generator Water Level (Wide Range)	✔	✔	✔
Core Exit Temperature	✔	✔	✔
Steam Generator Pressure	✔		✔
High Head SI Flow	✔		✔
Auxiliary Feedwater Flow	✔	✔	✔
RCS Hot Leg Temperature	✔	✔	
RCS Cold Leg Temperature	✔	✔	
Source Range Neutron Flux	✔	✔	
Containment Area Radiation (High Range)	✔	✔	✔
Condensate Storage Tank Level	✔	✔	
Pressurizer Level	✔	✔	✔
Containment Sump Water Level (Wide Range)	✔	✔	
RCS Subcooling Margin	✔		
Reactor Vessel Water Level	✔	✔	
Hydrogen Monitors	✔	✔	Note 1
Containment Sump Water Level (Narrow Range)	✔		
RWST Level (Wide Range)	✔		✔
Steam Generator Water Level (Narrow Range)	✔		
Spent Fuel Pool Exhaust Radiation (High Range)	✔		
Condenser Air Ejector (High Range)	✔		
Plant Vent Stack (High Range)	✔		
Steam Generator Blowdown Radiation	✔		
Steam Line Radiation	✔		
TDAFW Pump Exhaust Radiation	✔		
Pressurizer Pressure	✔		
PORV Position	✔		
PORV Block Valve Position	✔		
Pressurizer Safety Valve Position	✔		
Auxiliary Feedwater (AFW) Valve Position	✔		
Boric Acid Tank (BAT) Level	✔		
Containment Enclosure Negative Pressure	✔		
Containment Pressure (Narrow Range)	✔		
Intermediate Range Neutron Flux	✔		
Residual Heat Removal (RHR) Flow	✔		
Spray Additive Tank (SAT) Level	✔		

Note:

1. Hydrogen Monitors are not addressed in this report, since they are already addressed in the 50.44 Rulemaking Package (Reference 17)

The recommended Type and Class based on the current accident management usage discussed in this report, and the basis for that recommendation, was included in the discussion of each indication. The summary of the recommended classifications are provided in the following Table 10 for those PAM indications that are recommended for inclusion in the Technical Specifications based on the current accident management usage discussed in this report. Table 11 provides a similar summary for key instrumentation that is currently in the PAM Technical Specification in NUREG-1431 and that is not recommended for inclusion in the revised PAM Technical Specifications.

Table 10 Regulatory Guide 1.97 Classification for Recommended PAM Technical Specification Instrumentation

Function	Typical Reg. Guide 1.97 Variable Type/Category(1)	WCAP-15981 Type/ Category	Basis
1. Power Range Neutron Flux	B1	B1	Provides verification of automatic actuation of RPS – Type B. Provides direct information to verify accomplishment of the subcriticality safety function – Category 1.
2. Steam Generator Pressure	A1	A1	Provides information for operator action for SGTR break flow termination for which no automatic control is provided – Type A. Together with RCS pressure, provides information to verify that break flow through a ruptured SG tube is terminated thereby satisfying the inventory safety function – Category 1.
3. RWST Level	A1	D1	Provides information to indicate the continued operation of SI for continued inventory control – Type D. Provides information to indicate the need to refill the RWST to continue inventory control for SGTR and ISLOCA events – Category 1. (2)
4. High Head SI Flow	D2	B1	Provides verification of automatic actuation of SI – Type B. Provides direct information to verify the operation of SI to maintain the inventory safety function for core cooling – Category 1.
5. RCS Pressure (Wide Range)	A1	A1	Provides information for operator action for SGTR break flow termination for which no automatic control is provided – Type A. Together with SG pressure, provides information to verify that break flow through a ruptured SG tube is terminated thereby satisfying the inventory safety function – Category 1.
6. Containment Pressure (Wide Range)	C1	C1	Provides information to identify a fission product barrier challenge – Type C. Provides direct verification of containment cooling to maintain the containment fission product barrier safety function – Category 1.

Table 10 Regulatory Guide 1.97 Classification for Recommended PAM Technical Specification Instrumentation (cont.)

Function	Typical Reg. Guide 1.97 Variable Type/Category	WCAP-15981 Type/ Category	Basis
7. Penetration Flow Path Containment Isolation Valve Position	B1	B1	Provides verification of automatic actuation of Phase A and Phase B containment isolation – Type B. Provides direct verification of containment isolation to maintain the containment fission product barrier safety function – Category 1.
8. Containment Area Radiation (High Range)	A1	C1	Provides information to identify a fission product barrier challenge – Type C. Provides direct verification of satisfying the core cooling safety function – Category 1.
9. Pressurizer Level	A1	A1	Provides primary information needed to permit operators to take specified manual actions to terminate SI – Type A. Provides information related to satisfying the RCS inventory safety function to permit SI termination – Category 1.
10. SG Water Level (Wide Range)	D1	A1	Provides information for operator action maintaining a heat sink for which no automatic control is provided – Type A. Provides direct verification of satisfying the heat sink safety function – Category 1.
11. Core Exit Temperature	A1	A1	Provides information needed to permit the operators to take specified manual actions to initiate RCS depressurization – Type A. Provides direct verification of satisfying the core cooling safety function – Category 1.
12. AFW Flow	A1	B1	Provides verification of automatic actuation of AFW – Type B. Provides direct verification of satisfying the heat sink safety function – Category 1.

Note:

(1) Only the highest Reg. Guide 1.97 classification is shown in this table.

(2) If switchover to ECC recirculation is based on the RWST level indication rather than the RWST level alarm, it should be classified as a Type A variable rather than a Type D variable.

Table 11 Regulatory Guide 1.97 Classification for PAM Instrumentation Relocated to LCDs			
Function/No.	**Typical Reg. Guide 1.97 Variable Type/Category**	**WCAP-15981 Type/ Category**	**Basis**
1. Source Range Neutron Flux	B1	B3	Provides verification of automatic actuation of RPS – Type B. Provides diagnostics of continued subcriticality during RCS cooldown and depressurization – Category 3.
2. RCS Hot Leg Temperature	A1	B3	Provides information to indicate whether the core cooling safety function is being accomplished – Type B. Provides backup to the CETs – Category 3.
3. RCS Cold Leg Temperature	A1	B3	Provides information to indicate whether the core cooling safety function is being accomplished – Type B. Provides backup to the CETs – Category 3.
4. Reactor Vessel Water Level	B1	B3	Provides information to indicate whether the core cooling safety function is being accomplished – Type B. Provides backup to the CETs – Category 3.
5. Containment Sump Water Level (Wide Range)	A1	B2	Provides information to indicate whether the core cooling safety function can be accomplished when RWST switchover occurs – Type B. Provides information on the status of ECC recirculation delivery – Category 2.
6. Condensate Storage Tank Level	A1	B2	Provides information to indicate whether continued SG heat sink can be maintained – Type B. Provides in formation indicating long term AFW system operating status – Category 2.

Note:
Only the highest Reg. Guide 1.97 classification is shown in this table.

6 REQUIRED NUMBER OF PAM INSTRUMENTATION CHANNELS AND ALTERNATE INDICATIONS

The recommended required number of PAM instrumentation channels, and those functions that have alternate indications, are identified in Table 12. The basis for the required number of channels is provided in the following section of this report.

If one or two required number of channels for a PAM Function is inoperable, alternate indications may be available for the operator to use in diagnosing and/or performing the key operator actions summarized in Table 8. While these alternate indications do not provide direct indication from which the operator cue is taken in the applicable procedures and guidelines, the operators are trained to utilize these alternate indications when the primary indication is not available. As a note, many of these alternate indications are routinely used by the operators to assess the accuracy of the primary indications from which actions are prescribed in the procedures and guidelines. The basis for the applicability of alternate indications is described in this section and is also summarized in Table 12. In the discussion below, only those generically applicable alternate indications were identified; plant specific alternates may be available and justified on a plant specific basis.

Table 12 Recommended Number of PAM Instrumentation Channels

Instrument	Required Channels (see Note 1)	Alternate Indication (see Note 2)
Power Range Neutron Flux	2	Yes
Steam Generator Pressure	2 per SG	No
RWST Level (Wide Range)	2	No
High Head SI Flow	1 per train	Yes
Reactor Coolant System Pressure (Wide Range)	2	No
Containment Pressure (Wide Range)	2	No
Penetration Flow Path Containment Isolation Valve Position	2 per penetration flow path (Note 3)	No
Containment Area Radiation (High Range)	2	Yes
Pressurizer Level	2	No
Steam Generator Water Level (Wide Range)	2 per SG	Yes
Core Exit Temperature	2 (Note 4)	No
Auxiliary Feedwater Flow	2	Yes

Notes:

1. It should be noted that some plant designs may only contain a single channel for certain PAM functions, and do not have to meet the required number of channels identified in this table.
2. Details of the alternate instrumentation are shown in Table 13.
3. Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.
4. The basis for 2 Core Exit Temperature channels is discussed below.

Table 13 Summary of Alternate PAM Instrumentation	
Primary Instrumentation	**Alternate Instrumentation**
SG Water Level (Wide Range)	SG Narrow Range Level AND Auxiliary Feedwater Flow Rate
Power Range Neutron Flux	Intermediate or Source Range Indications AND either the Rod Position Indicators OR Rod Bottom Lights
Containment Area Radiation (High Range)	Portable Radiation Monitors
High Head Safety Injection Flow	High Head Safety Injection Pump Amperage AND SI Pump Discharge or Header Pressure AND Automatic SI valve position
Auxiliary Feedwater Flow	Motor Driven Pumps: Pump Amperage AND Pump Discharge Pressure OR flow control valve (SG supply) position
	Turbine Driven Pump: Pump Discharge Pressure OR steam supply valve position AND flow control valve (SG supply) position

Power Range Neutron Flux

The power range neutron flux indication is used immediately following an accident or receipt of a reactor trip signal. For the purposes of providing an indication of the failure to achieve subcriticality, which would result in operator actions to manually trip the reactor, the power range neutron flux indication is the only direct means of providing this information.

If the power range neutron flux indication is not available, an alternate method of monitoring subcriticality is a combination of either the intermediate range or source range neutron flux indications, AND either the rod bottom lights or rod position indicators. The rod bottom lights and rod position indicators are the primary alternate indications used in the EOPs to verify the accuracy of the power range neutron flux indication. If the control rods are fully inserted, initial subcriticality is ensured by the plant design basis. Therefore, these alternate indications can also provide the information necessary for the operators to determine the need to initiate a manual reactor trip. The intermediate and source range indications are the primary backup indications used in the Functional Restoration Guidelines for diagnosis of a potential loss of core shutdown (once initial subcriticality is reached). Thus, the intermediate and source range indicators can provide an alternate to the power range monitor to diagnose continued subcriticality. The combination of either the source or intermediate range neutron flux monitors and either the rod bottom lights or rod position indicators provide an alternate indication of subcriticality to the power range neutron flux monitor.

The power range neutron flux reactor trip function is required to be Operable in Modes 1 and 2. The PRA typically shows that power range neutron flux is a key indication for accident management operator actions to initiate manual reactor trip to bring the reactor to a subcritical condition, which is a keff of < 0.99. This is consistent with the keff of > 0.99 specified as the reactivity condition for Mode 2 and for power operation in Mode 1. Subsequent operator actions (in Mode 3 after a reactor trip) to assure that the reactor remains in subcritical state, where the power range neutron flux indication may no longer be operable, such as during RCS depressurization, were not determined to be important for long term core cooling. Therefore, for the required PAM indication function (i.e., confirming a reactor trip from Modes 1 and 2), the Power Range Neutron Flux indication is only required to be Operable in Modes 1 and 2. This also makes the PAM Technical Specification Mode of applicability for the Power Range Neutron Flux indication consistent with the corresponding Mode of applicability of the Reactor Trip System Instrumentation Technical Specification.

SG Pressure

SG pressure is used following an accident or receipt of a reactor trip signal to indicate secondary side integrity. It is also used as the target pressure for RCS cooldown and depressurization to terminate the break flow following a SGTR. There is no reliable alternate indication for determining the SG pressure. Therefore, no alternate indication is proposed in the event that SG pressure indication is unavailable.

RWST Level

RWST level indication is required following an accident or receipt of a reactor trip signal. The RWST level instrumentation provides an indication of the need to initiate RWST makeup for accident sequences in which most of the discharge of reactor coolant is to locations outside of the containment. The narrow range RWST level indication only has a sufficient range to indicate the RWST level associated with the Technical Specification requirement for the minimum RWST level and does not extend to the level needed to indicate the need for RWST refill following an accident. Thus, there is no alternate instrumentation to support the operator action to refill the RWST to provide continued makeup to the RCS for long term core cooling if the instrumentation is unavailable.

High Head Safety Injection Flow

There is typically only one channel of High Head SI Flow instrumentation per train to provide indication of SI flow for the diagnosis of the need for operator actions to manually initiate an SI signal or to start the high head SI pumps in the event that automatic SI initiation does not occur. An alternate method of monitoring flow from the high head SI pumps can be inferred from the high head SI pump amperage and the high head SI pump discharge or header pressure indications, and the automatic SI valve position indication. All of these indications are typically used in the EOPs to provide verification of the satisfactory operation of the High Head SI pumps. Since each indication only provides a portion of the verification of High Head SI flow, all three indications are required to provide a high degree of confidence of adequate High Head SI flow if the High Head SI flow rate indication is inoperable.

RCS Pressure (Wide Range)

RCS pressure indication is used for determining RCS pressure and RCS subcooling following an accident or receipt of a reactor trip signal. The pressurizer pressure indication does not have sufficient range to satisfy any of the indications that prompt important operator actions based on RCS pressure. Therefore, no alternate indication is proposed in the event that RCS pressure indication is unavailable.

Containment Pressure (Wide Range)

Containment pressure indication is required following an accident or receipt of a reactor trip signal. The containment pressure wide range indication provides information for the determination of an inadequate containment cooling condition and for the determination of a challenge to the containment pressure retaining integrity. The narrow range containment pressure instrumentation, which only extends to the design basis pressure, could be used to determine an inadequate containment cooling condition. However it does not have a sufficient range to be useful in determining the potential of a challenge to containment integrity due to overpressurization. Therefore, no alternate indication is proposed if the containment pressure indication is unavailable.

Penetration Flow Path Containment Isolation Valve Position

The Penetration Flow Path Containment Isolation Valve Position indication provides a direct indication of a failure to completely isolate containment following the receipt of a containment isolation signal. In penetrations that contain two motor operated isolation valves, the indication from each valve is typically provided by separate electric trains so that in the event of a failure of one train of electric power, the indication from the other train would be available. The important operator action taken from this information is for manual containment isolation in the event that automatic isolation does not occur, and also for input to the declaration of the appropriate EAL condition. This instrumentation is the only means of confirming that all containment isolation valves are in the isolation position following an automatic containment isolation signal. Therefore, no alternate indication is proposed if the penetration flow path containment isolation valve position indication is unavailable.

Containment Area Radiation (High Range)

The containment area radiation provides an indication of a loss of one or more fission product barriers. In the event that both required channels are unavailable, an alternate method of monitoring is the use of portable radiation monitors outside of containment to infer the order of magnitude of the level of radiation inside the containment. The Core Damage Assessment methodology in WCAP-14696-A shows that the details of the accident sequence can account for differences in containment radiation levels that are an order of magnitude different. Portable radiation monitors are capable of providing information for an order of magnitude estimate.

Pressurizer Level

The pressurizer level indication is used for determining pressurizer level for SI termination following an accident. There are no other means of inferring pressurizer level in the event that the pressurizer level indication is unavailable. Therefore no alternate indication is proposed.

Steam Generator Water Level (Wide Range)

The SG level indication is used to maintain a heat sink and for the diagnosis of a SGTR accident, and can be fulfilled by one channel of SG narrow range instrumentation per SG is available. The indication for the initiation of bleed and feed requires that all SGs indicate a very low level. An alternate indication for SG level Wide Range is a combination of one SG level Narrow Range channel, and the AFW flow rate to that SG. This combination can be used to infer that an inventory is available in the SG in place of the SG level wide range indication. The SG narrow range level indication provides a suitable alternate for the diagnosis functions of the SG level wide range indication except for the initiation of bleed and feed cooling when the SG level approaches dryout. Bleed and feed cooling is only required to be implemented if the level in all SGs is below the setpoint level (which is near the dryout stage). If the SG level wide range indication is inoperable, a suitable alternate indication for the initiation of bleed and feed cooling would be a SG level narrow range indication off-scale low in ALL SGs AND an AFW flow rate below that needed for decay heat removal (which is already specified in the EOPs). Therefore, the combination of SG narrow range level and AFW flow rate adequately serve as an alternate if the SG wide range indication is inoperable.

RCS Temperature

RCS temperature indication is required following an accident for operator determination of RCS subcooling for both RCS cooldown and depressurization, and for SI termination. This PAM indication is provided by the CETs. The required number of CET channels is discussed under the core temperature indication requirements below.

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.

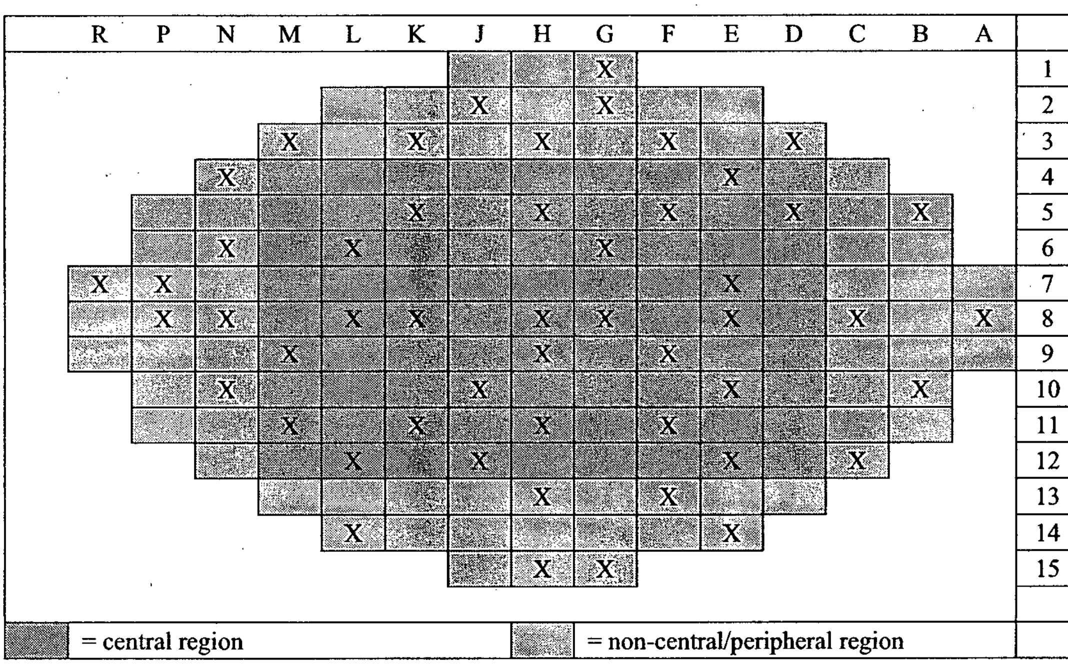

Figure 1 Typical Core Exit Thermocouple Locations for a Three Loop Plant

The only alternate indication used in the WOG ERGs for the indication of inadequate core cooling is the reactor vessel level indication. However the reactor vessel level indication is not used to indicate the need to transition from the EOPS to the SAMG; only the CET indications provide an operator cue for this transition. Since the CETs are used for important operator actions in the SAMG, it is concluded that there are no appropriate alternate indications for the CETs.

Auxiliary Feedwater Flow Indication

The AFW Flow instrumentation provides an indication of AFW flow that supports the diagnosis of the need for operator actions to manually initiate an AFW signal or start AFW pumps in the event that automatic AFW initiation does not occur. The AFW Flow instrumentation provides the most direct indication of AFW flow to allow the diagnosis of the need for operator actions to manually start the AFW pumps to initiate an alternate source of feedwater. An alternate method of inferring AFW flow rate for the motor driven pumps can be provided by the AFW pump amperage AND the AFW pump discharge pressure OR the flow control valve position (SG supply) indications. An alternate method of inferring AFW flow rate for the turbine driven pump, is the AFW pump discharge pressure OR the steam supply valve position AND the flow control valve position (SG supply) indications.

All of these indications are typically used in the EOPs to provide verification of the satisfactory operation of the AFW pumps. Since each indication may only provide a portion of the verification of AFW flow, some combination of the alternate indications, as shown in Table 12, provides a high degree of confidence of adequate AFW flow if the AFW flow rate indication is inoperable.

These alternate indications are appropriate since the risk significant action is to provide an alternate SG feed source if no AFW pumps are available.

7 SUMMARY AND CONCLUSIONS

A thorough assessment of the significance of the PAM instrumentation to plant safety has been made to determine the PAM instruments that should be included in the Technical Specifications to assure that the plant operators can bring the plant to a safe stable condition for events where automatic actuation of safety systems is not provided.

The assessment described in this report considered the criteria in 10 CFR 50.36 (c)(2)(ii) and the Regulatory Guide 1.97 instrumentation classifications as they relate to the inclusion of instrumentation in the Technical Specifications. The assessment concluded that the PAM instrumentation that should be included in Technical Specification 3.3.3 of NUREG-1431 to assure operability in the event of an accident are:

- Power Range Neutron Flux
- Steam Generator Pressure
- Refueling Water Storage Tank Level (Wide Range)
- High Head SI Flow
- Reactor Coolant System Pressure (Wide Range)
- Containment Pressure (Wide Range)
- Penetration Flow Path Containment Isolation Valve Position
- Containment Area Radiation (High Range)
- Pressurizer Level
- Steam Generator Water Level (Wide Range)
- Core Exit Temperature
- Auxiliary Feedwater Flow

The assessment also identified alternate indications for certain PAM instrumentation. The PAM instruments for which an alternate indication is available are:

- Power Range Neutron Flux
- High Head SI Flow
- Containment Area Radiation (High Range)
- Steam Generator Water Level (Wide Range)
- Auxiliary Feedwater Flow

The results of the assessment provide the basis for the relocation of the following PAM instrumentation contained in Technical Specification 3.3.3 of NUREG-1431 to LCDs:

- Source Range Neutron Flux
- RCS Hot Leg Temperature
- RCS Cold Leg Temperature
- Reactor Vessel Water Level
- Containment Sump Water Level (Wide Range)
- Condensate Storage Tank Level

The additional plant specific PAM instrumentation that is identified in Table 9, or other plant specific PAM instrumentation that is not identified in Table 9, that does not satisfy the requirements for inclusion in the Technical Specifications based on the methodology contained in this report can also be relocated from the Technical Specifications to LCDs.

The generic list of PAM instrumentation proposed to be included in Technical Specification 3.3.3 of NUREG-1431, and those instruments proposed to be relocated from plant specific Technical Specifications to LCDs must be confirmed on a plant specific basis by reviewing the plant specific DBA analyses, PRA, EOPs, SAMGs, and EPIP.

8 IMPLEMENTATION

The plant specific implementation of this methodology contained in this report requires a plant specific evaluation of the accident management application of PAM instrumentation contained in the: 1) Design Basis Accidents, 2) Emergency Operating Procedures, 3) Probabilistic Risk Assessment, 4) Severe Accident Management Guidelines, and 5) Emergency Plan as discussed in this report.

The generic list of PAM instrumentation proposed to be included in Technical Specification 3.3.3 of NUREG-1431, and those instruments proposed to be relocated from the plant specific Technical Specifications to LCDs must be confirmed on a plant specific basis by reviewing the plant specific DBA analyses, PRA, EOPs, SAMGs, and EPIP.

The overall process to be used by licensees to identify the PAM instrumentation that should be included in the Technical Specifications is discussed in Table 14 and shown in Figure 2. This process is identical to that described in this report, except that plant specific information would be used in place of generic information to determine the PAM instrumentation to be included in the plant specific PAM Technical Specifications. As discussed in Section 3.2, this is not a risk informed application to be evaluated using the guidance in Regulatory Guide 1.174. The methodology directly assesses the importance of instrumentation with respect to the criteria of 10 CFR 50.36 (c)(2)(ii) rather than focusing on the five elements of a risk informed approach as specified in Regulatory Guide 1.174. The methodology uses risk assessment as one element of the overall method to determine the instrumentation to be included in the PAM Technical Specification.

The first step in the process (Step 1 in Table 14) is to identify the operator actions that are assumed in DBA analyses using the criteria in Table 5. These operator actions satisfy Criterion 3 of 10 CFR 50.36 because no automatic actuation of equipment is included in the plant design for these actions.

The next steps are to identify the risk important operator actions from the plant PRA using the criteria in Table 5. This part has two distinct steps: a) verification of the PRA technical adequacy for this application (Step 2 in Table 14), and b) use of the PRA information to identify operator actions based on instrumentation that satisfies Criterion 4 of 10 CFR 50.36 (Step 3 in Table 14).

As discussed in Section 3.1, the licensee should ensure that the internal events PRA is technically adequate for this application. PRA technical adequacy of the internal events PRA for risk informed applications is typically addressed through the PRA peer reviews and self assessments using a variety of guidance, including the American Society of Mechanical Engineers (ASME) PRA Standard (Reference 7), Nuclear Energy Institute (NEI) PRA Peer Review Process Guidance (Reference 8) and/or Regulatory Guide 1.200 (Reference 9). Since this is not a risk informed application as described in Regulatory Guide 1.174, only a limited assessment of the PRA technical adequacy is required for this application. The limited assessment of the PRA technical adequacy only needs to consider the areas of the accident sequence analysis and the human reliability analysis to assure that treatment of operator actions based on plant instrumentation is appropriate. In particular, the licensee should confirm that all operator actions potentially impacted by the subject instruments have been identified, that the treatment of these operator actions in the PRA is appropriate (including the human error probability values and dependencies), and that there are no peer review comments that can affect the conclusions regarding instrument importance.

The RAW and Fussell-Vesely (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.

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 Human Reliability Analysis (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.

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.

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 PRA and HRA treatment of operator actions in the PRA associated with any variables (instrumentation) to be relocated from the PAM Technical Specification • Verify that any instrumentation proposed for relocation from the PAM Technical Specification does not cue an operator action important to risk for external initiating events • Identify appropriate changes to the Regulatory Guide 1.97 classifications to be consistent with the inclusion in, or relocation from, the PAM Technical Specification

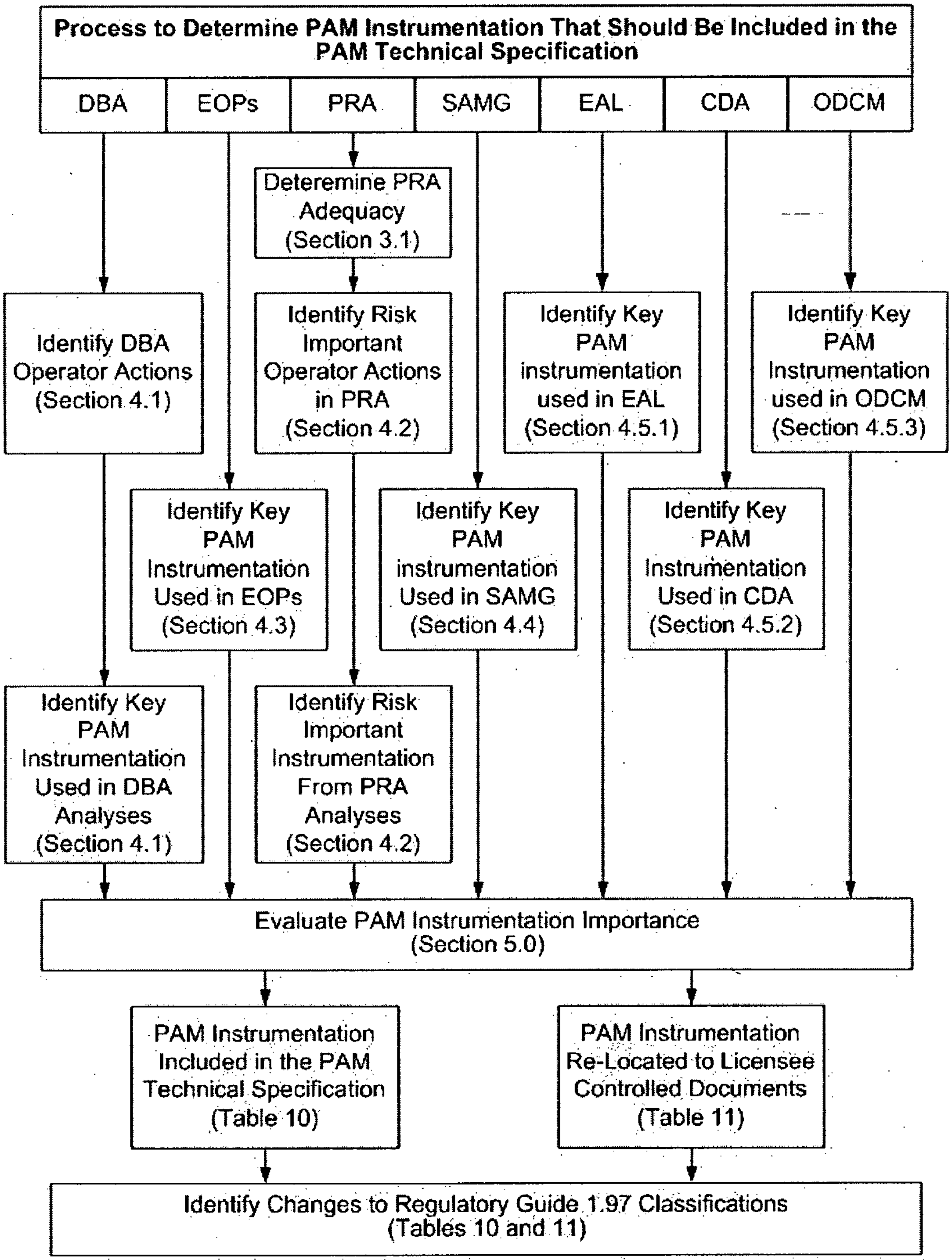

Figure 2 Process to Determine PAM Instrumentation That Should Be Included in the PAM Technical Specifications

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APPENDIX A
INSTRUMENTATION IMPORTANCE IN PRAs

A.1 BACKGROUND

In the early 1980's several plant specific Probabilistic Risk Assessments (PRAs) were performed (e.g., Zion Units 1 and 2, Indian Point Units 2 and 3) to resolve regulatory concerns related to severe accidents. Several other PRAs were completed throughout the 1980's. The comprehensive NUREG-1150 study was completed in the late 1980's using five reference plants to characterize severe accident risks. These studies identified plant specific design and operational differences as a primary reason for significant differences in the severe accident risks. In this context, severe accident risks are a measure of probability and consequences.

Several measures of severe accident risks have been identified and subsequently used as risk "metrics." The most common of these is core damage frequency (CDF) and large early release frequency (LERF). Also, "importance measures" were developed to indicate the contribution of systems, components, and operator actions to these risk metrics:

- The Risk Achievement Worth (RAW) is a measure of the increase in risk (CDF or LERF) if the system, component, or operator action is assumed to fail with a probability of unity. It is defined as the ratio of the CDF or LERF with failure of the component set to unity and the CDF or LERF using the best estimate failure value.

- The Risk Reduction Worth (RRW) is a measure of the decrease in risk if the failure probability is set to zero. It is defined as the ratio of the CDF or LERF with failure of the component set to zero and the CDF or LERF using the best estimate failure value.

- The Fussell-Vesely (F-V) measure is a derivation of the RRW and is defined as, F-V=1+1/RRW.

Because of the potential for plant specific differences to control the severe accident risks, each plant was required to perform an Individual Plant Examination (IPE) in response to NRC Generic Letter 88-20 in the late 1980's. The purpose of the IPE was to identify any plant specific vulnerabilities (weaknesses) that would dominate the risk profile of the plant. In some cases, plant modifications were made to address specific vulnerabilities that were determined to be unacceptable. While a quantitative PRA was required to quantify the risks associated from internally initiated accidents from an at-power plant operating state, GL 88-20 also required at least a qualitative risk assessment of external initiating events, such as seismic and fire. This is commonly referred to as an Individual Plant Examination for External Events (IPEEE).

Subsequently, each plant's IPE has evolved into a Probabilistic Risk Assessment (PRA) study. The primary difference between the IPE and the PRA is in the depth to which the plant is modeled; the IPE only modeled the plant features necessary to identify vulnerabilities, while the PRA models include many more systems and components that have a somewhat lower overall contribution to risk. These plant specific PRA models have been used to address regulatory and plant operational differences to ensure that the severe accident risks remain low during all phases of plant operations.

As the PRA models have become more mature and confidence has been gained in their application, the PRA has been used, along with deterministic analyses and engineering judgment, to relax unnecessarily restrictive regulatory requirements. The NRC has developed guidance on the use of PRA to change regulatory requirements in the form of Regulatory Guide 1.174. This approach has been termed risk-informing regulatory requirements. This regulatory guide uses the change in CDF and LERF due to the proposed change in regulatory requirements, along with importance measures to determine, in part, whether such a regulatory requirement change is acceptable. This process is also being used in the development of the proposed 10 CFR 50.69 rulemaking to determine the risk informed repair and replacement treatment requirements.

A.2 INSTRUMENTATION MODELING IN PRAs

Instrumentation is typically not modeled explicitly in the PRA. Rather, assumptions about the instrumentation availability and reliability are typically included in other PRA models. For example, the reliability of instrumentation to generate a reactor trip or SI signal is typically included in the overall reactor trip or SI signal model. The reactor trip or SI model combines the instrumentation failure with many other potential failure modes to determine the reliability of the reactor trip of SI function itself.

In the case of Post Accident Monitoring (PAM) instrumentation, the assumptions regarding its availability and reliability are most often included as part of the Human Reliability Analysis (HRA). In other words, the failure of the instrumentation is modeled as one of the causes of a failure of a required human interaction to achieve a safe, stable, plant state. Since the PAM instrumentation does not generate any automatic signals, the importance of PAM instrumentation can be investigated by identifying the operator actions that rely on instrumentation and determining the risk importance (e.g., RAW and F-V) of that operator action.

A.3 IMPORTANCE DATA FROM PRAs

In 1997, the WOG authorized a program for the collection of important features and results from Westinghouse NSSS plant specific PRA studies into a comprehensive database. This database was subsequently completed as a proprietary product for WOG utility use in 1999. The database was constructed by requesting that each Westinghouse NSSS licensee provide their current PRA values for certain parameters that were thought to be the more dominant contributors to core damage. In the case of HRA results, a prescribed set of operator actions were defined for the primary input based on those operator actions that were identified to be the most important to the PRA results. The database also contains other important operator actions from utility PRAs, as provided by those utilities.

A database update was conducted in 2001, and completed for WOG utility use in 2002, to reflect newer PRA results. The new results were a product of significant recent changes in utility PRAs as a result of utilities upgrading the PRA models for both risk informed applications and to respond to the PRA Peer Review findings.

The 1999 PRA survey results were collected for both RAW and F-V values for operator actions modeled in the PRA. However, the PRA information collected in the 2002 survey only included F-V values of operator actions, since the importance measures for operator actions generally focus on improvements in operator actions via training and/or procedure modifications. However, the RAW importance measure is more appropriate for the evaluation of the PAM instrumentation based on the potential decrease in equipment reliability if it is removed from the Technical Specifications. In the 2002 survey, it was not foreseen that the operator action importance measures would be used to investigate the reliability of the information upon which the operator actions are based.

To address the issue of operator action naming, common sets of operator action titles were developed from the database information, as shown in Tables A-1, A-2 and A-3. Using these-standard operator actions, the risk importance of the operator actions over all Westinghouse NSSS plants is shown in Tables A-1 through A-3. Table A-1 summarizes the F-V importance measure results from information provided in the 2002 database update. Table A-2 provides the F-V importance measures from the 1999 database; Table A-3 provides the RAW values from the 1999 database. The information provided in Tables A-1 through A-3 shows the maximum and minimum values for the risk importance measures for each operator action reported in the databases, along with the median value based on all of the plants that provided a value. The mean value is not included, because it is typically skewed by one or two very high RAW and F-V values in the database.

A.4 CRITERIA FOR RISK IMPORTANCE

The EPRI PSA Applications Guide (Reference 18) suggests that a component has a high risk significance if the F-V value is greater than 0.05 or the RAW value is greater than 2.0. The use of F-V and RAW to identify risk important systems structures and components has been used in Section 5.1 of NEI 00-04 (Reference 19), which is endorsed by the NRC in Regulatory Guide 1.201 (Reference 20). The system level importance measure criteria for F-V from Reference 18 are applicable here based on two observations: 1) the operator action failure disables an entire system, as opposed to a component failure that may only contribute to a system failure, and 2) instrumentation failures are only a portion of the operator action failure, with the remainder being made up of operator errors of omission and commission in reading instrumentation and taking the appropriate actions. However, risk importance thresholds cannot be used as absolute criteria above which SSCs can be considered to be clearly risk-significant and below which they can be accepted as low in safety-significance. Rather, they are screening devices that provide insights as to what may or may not be important to safety for any given plant or system design.

At this point, a discussion of the common usage of the risk importance measures is in order. Risk Achievement Worth defines the importance of a PRA parameter by comparing the overall risk results (e.g., overall core damage) with the parameter at its nominal value, to the overall risk if the parameter is totally unreliable (e.g., always failed). Fundamentally, the Risk Achievement Worth has little to do with the design or reliability of a component itself, but relies heavily on the defense-in-depth available in the form of redundant SSCs to mitigate the effects of the loss of the component. On the other hand, the F-V more directly relates to the reliability of a component by suggesting the impact on risk from improvements in reliability.

Table A-1 2002 PRA Survey Results Operator Action Risk Reduction Worth *			
Operator Action	**Max**	**Min**	**Median**
Align Alternate Cooling to Charging Pumps	1.53	1.008	1.05
Restore AC Power	1.59	1.00	1.04
Restore Equipment Following AC Power Recovery	1.17	1.02	1.05
Re-Align AFW	1.10	1.00	1.02
Align Alternate Feedwater Source	1.10	1.01	1.05
Perform Remote Shutdown	1.20	1.10	1.12
Perform Bleed and Feed	1.20	1.00	1.02
Restore CCW	1.06	1.01	1.03
Restore Instrument Air	1.08	1.00	1.03
Align Emergency Boration	1.02	1.00	1.01
Transfer to Cold Leg Recirculation	1.59	1.03	1.05
Isolate Stuck Open Pressurizer PORV	1.04	1.004	1.01
Isolate Ruptured SG	1.06	1.007	1.03
Reactor Shutdown for ATWS	1.11	1.005	1.02
Manual SI	1.23	1.006	1.02
Establish Normal RHR	1.06	1.006	1.03
RCS Cooldown and Depressurization	1.19	1.00	1.04
Refill CST	1.09	1.07	1.08
Refill RWST	1.12	1.01	1.07
RCS Cooldown for SGTR	1.08	1.004	1.06
Restore Service Water	1.17	1.001	1.03
Control AFW Flow to Maintain SG water level	1.06	1.007	1.04
Terminate SI for SS Break	1.07	1.001	1.02
Terminate SI for SGTR	N/R	N/R	N/R

Note:

N/R = Not reported in the 2002 survey.

* Values in the database are given as Risk Reduction Worth which is related to F-V by: RRW = 1 - (1 - F-V).

Table A-2 1999 PRA Survey Results Operator Action Risk Reduction Worth*			
Operator Action	**Max**	**Min**	**Median**
Align Alternate Cooling to Charging Pumps	1.29	1.000	1.042
Restore AC Power	1.085	1.002	1.023
Restore Equipment Following AC Power Recovery	1.10	1.000	1.012
Re-Align AFW	1.22	1.000	1.005
Align Alternate Feedwater Source	1.10	1.005	1.019
Perform Remote Shutdown	1.29	1.020	1.125
Perform Bleed and Feed	1.29	1.008	1.015
Restore CCW	1.06	1.000	1.015
Restore Instrument Air	N/R	N/R	N/R
Align Emergency Boration	1.03	1.000	1.000
Transfer to Cold Leg Recirculation	1.59	1.000	1.029
Isolate Stuck Open Pressurizer PORV	N/R	N/R	N/R
Isolate Ruptured SG	1.38	1.000	1.005
Reactor Shutdown for ATWS	1.04	1.000	1.000
Manual SI	1.02	1.005	1.008
Establish Normal RHR	1.03	1.013	1.015
RCS Cooldown and Depressurization	1.13	1.000	1.016
Refill CST	N/R	N/R	N/R
Refill RWST	1.42	1.001	1.071
RCS Cooldown for SGTR	N/R	N/R	N/R
Restore Service Water	1.45	1.000	1.02
Control AFW Flow to Maintain SG water level	N/R	N/R	N/R
Terminate SI for SS Break	1.05	1.000	1.000
Terminate SI for SGTR	1.10	1.000	1.002

Note:

N/R = Not reported in the 1999 survey.

* Values in the database are given as Risk Reduction Worth which is related to F-V by: RRW = 1 - (1 - F-V).

Table A-3 1999 PRA Survey Results Operator Action Risk Achievement Worth (Using RAW Importance Measure)

Operator Action	Max	Min	Median
Align Alternate Cooling to Charging Pumps	481	1.18	2.24
Restore AC Power	26.3	1.33	2.88
Restore Equipment Following AC Power Recovery	4.50	1.15	1.40
Re-Align AFW	88	1.00	2.46
Align Alternate Feedwater Source	342	1.00	1.53
Perform Remote Shutdown	3.32	1.62	2.49
Perform Bleed and Feed	6.60	1.00	2.46
Restore CCW	15.4	1.01	2.46
Restore Instrument Air	N/R	N/R	N/R
Align Emergency Boration	3.0	1.00	1.10
Transfer to Cold Leg Recirculation	513	1.00	10.35
Isolate Stuck Open Pressurizer PORV	N/R	N/R	N/R
Isolate Ruptured SG	925	1.00	1.68
Reactor Shutdown for ATWS	11.5	1.00	1.02
Manual SI	68.8	1.7	3.05
Establish Normal RHR	N/R	N/R	N/R
RCS Cooldown and Depressurization	22.8	1.00	4.05
Refill CST	N/R	N/R	N/R
Refill RWST	5.25	1.00	1.20
RCS Cooldown for SGTR	N/R	N/R	N/R
Restore Service Water	954	1.00	1.41
Control AFW Flow to Maintain SG water level	N/R	N/R	N/R
Terminate SI for SS Break	20.2	1.00	1.10
Terminate SI for SGTR	23.1	1.00	1.16

Note:
N/R = Not reported in the 1999 survey.

In PRA applications, the RAW is typically used to assess the conditional risk during the time that a component is assumed to be removed from service. If the component is in service, then the components with the highest Risk Achievement Worth are those that should be considered for protecting against failure or avoiding additional activities that could remove them from service or render them inoperable. Risk Achievement Worth can be a useful tool in configuration risk management in this regard. In this application, the Risk Achievement Worth measure of importance can be an indicator for maintaining the current reliability of the instrumentation under consideration. Components ranking high in Risk Achievement Worth are those which potentially can result in the greatest increase in risk if their reliability is allowed to degrade. These components should be focused on in the monitoring of reliability and availability efforts, as well as other potential special treatment requirements. Less benefit is expected to be derived by focusing on systems and components ranking low in Risk Achievement Worth, since greater uncertainty can be tolerated in their performance due to the limited impact they are likely to have on risk.

Application of the Fussell-Vesely measure of importance includes the identification of SSCs that may be candidates for modification or improvement such that the overall risk can be lowered if the failure probability were reduced. Components ranking high in Fussell-Vesely are those at which efforts to improve the reliability or redundancy may have the greatest benefit. Components ranking low in Fussell-Vesely importance are not necessarily the best components on which to focus such efforts, since even if they were to be made completely reliable, they would only have a limited impact on overall risk.

It must be recognized when calculating either of these importance measures that it is physically impossible to make a component perfectly reliable (as is assumed for the Fussell-Vesely measure of importance) and it is highly unlikely that a component will always fail when called upon to perform its function or will always be out of service (as is the case for Risk Achievement Worth). In this regard, the values derived for each of these measures of importance should be considered as extremes or at least bounding in their characterization of the impact of the individual component or system on risk.

A.5 ASSESSMENT OF OPERATOR ACTION IMPORTANCES

The data summary in Tables A-1 through A-3 reveals that there is significant variability in the risk significance of many operator actions from plant to plant. That is, the risk importance of a particular operator action, based on RAW or F-V, may be significantly different from one plant to another. There are a number of reasons for this, including:

- Differences in the HRA models, including differences in the human error probabilities assigned to various actions,
- Differences in the manner in which operator actions are grouped in the HRA model, and
- Differences in the contribution to core damage for a given operator action due to plant design and plant specific equipment reliability factors.

A comparison of the F-V values reported in the 1999 and the 2002 surveys shows that there is not a significant difference in the results. That is, the operator actions with high F-V values in the 1999 survey also had high F-V values in the 2002 survey. The same conclusion can be drawn with respect to the low

F-V values; those operator actions with low F-V values in the 1999 survey also had low F-V values in the 2002 survey. Although RAW was not reported in the 2002 survey, it is assumed that the RAW values would also follow this same trend.

Therefore, the use of the operator action RAW values from the 1999 PRA survey, as shown in Table A-3 are a valid basis for assessing the importance of instrumentation for accident management.

From Table A-3 the operator actions with the highest RAW values, in descending order based on the median values for Westinghouse NSSS plants, are:

- Transfer to Cold Leg ECC Recirculation,
- RCS Cooldown and Depressurization,
- Manual Safety Injection,
- Restore AC Power,
- Perform Remote Shutdown,
- Re-align Auxiliary feedwater,
- Perform Bleed and Feed,
- Restore Component Cooling Water,
- Align Alternate Cooling to Charging Pumps,
- Isolate Ruptured SG,
- Align Alternate Feedwater Source
- Restore Service Water,
- Restore Equipment Following AC Power Recovery,
- Refill RWST,
- Terminate SI (SGTR and Secondary Side Breaks),
- Align Emergency Boration, and
- Reactor Shutdown for ATWS.

From these operator actions identified above, several can be eliminated based on the lack of instrumentation required to successfully complete the actions. The operator actions eliminated from further consideration are:

- Restore AC Power – This action is based on plant Abnormal Operating Procedures. The only instrumentation required for this action is the emergency bus voltage, which is an indicator that the action has been successfully completed. Since the successful completion of the operator action for restoration of AC power is not dependent on a specific indication that is provided by plant instrumentation, there is no potential post accident monitoring implication.

- Perform Remote Shutdown – The requirements for instrumentation at the remote shutdown panel are contained in the Remote Shutdown System Technical Specification, and are not PAM instrumentation.

- Restore Component Cooling Water (CCW) – This action is based on plant Abnormal Operating Procedures. The diagnosis of a fault in the CCW system and subsequent operator actions to restore CCW are based on the failure in a normally operating system. The failure of the system would be indicated in the control room by multiple indication and alarms. As such, no essential "key" parameter indication exists, since the operator action is based on the status of the entire system. The only instrumentation required for this action is the CCW flow and temperature, which is an indicator that the action has been successfully completed. Since this is not an action required to diagnose a condition that could lead to core damage that has a high risk significance, it does not satisfy Criterion 4 of 10 CFR 50.36 (c)(2)(ii) and therefore should not be included in the PAM Technical Specification.

- Align Alternate Cooling to Charging Pumps – This action is based on plant Abnormal Operating Procedures for loss of Component Cooling function to the charging pumps. The diagnosis of the loss of CCW and subsequent recovery actions are discussed above. The re-alignment of cooling to the charging pumps is a direct consequence of the diagnosis of a loss of CCW and is not based on any specific additional instrumentation indications. Since this is not an action required to diagnose a condition that could lead to core damage that has a high risk significance, it does not satisfy Criterion 4 of 10 CFR 50.36 (c)(2)(ii) and therefore should not be included in the PAM Technical Specification.

- Restore Service Water (SW) – The diagnosis of a fault in the SW system and subsequent operator actions to restore SW are based on the failure in a normally operating system. The failure of the system would be indicated in the control room by multiple indication and alarms. As such, no essential "key" parameter indication exists, since the operator action is based on the status of the entire system. The only instrumentation required for this action is the SW flow and temperature, which is an indicator that the action has been successfully completed. Since this is not an action required to diagnose a condition that could lead to core damage that has a high risk significance, it does not satisfy Criterion 4 of 10 CFR 50.36 (c)(2)(ii) and therefore should not be included in the PAM Technical Specification.

- Restore Equipment Following AC Power Recovery – This action is based on the plant Emergency Operating Procedures for Loss of All AC Power. This operator action follows the operator actions to restore AC power to the vital bus(es). An indication of successful restoration of AC power to a vital bus is the bus voltage. Various instrumentation are also available to indicate that actions to restore equipment have been successfully completed (e.g., pump amperage and flow). The only unique indication that equipment can be restored to a vital AC bus is the bus voltage. The vital bus voltage requirements are addressed by the Distribution Systems Technical Specification and are not PAM instrumentation.

It is also noted that although reactor coolant pump (RCP) seal LOCAs are an important contributor to core damage for Westinghouse PWRs, there are no operator actions modeled in the PRA to protect the RCPs from an RCP seal LOCA. The plant specific off-normal / abnormal procedures provide guidance for restoring seal cooling for those sequences that are susceptible to RCP seal LOCAs (which are all sequences involving a loss of all RCP seal cooling). However, if RCP seal cooling is not quickly re-established, then the procedures typically instruct the operators not to re-establish RCP seal cooling in order to avoid additional RCP seal damage due to thermal shock. Thus, there are no risk significant operator actions for preventing an RCP seal LOCA.

The next step in the assessment is to relate the PAM instrumentation to the operator actions modeled in the PRA. The instrumentation utilized for each operator action was identified by reviewing the detailed

PRA models for several plants and confirming these results with an independent review of the generic WOG Emergency Response Guidelines, upon which all of the WOG plant Emergency Operating Procedures are based. The results of this assessment are shown in Table A-4.

Table A-4 Instrumentation Required for Operator Actions Modeled in the PRA		
Operator Action	**Applicable EOPs**	**Associated Instrumentation**
Transfer to Cold Leg Recirculation	E-1, "Loss of Reactor or Secondary Coolant" E-3, "Steam Generator Tube Rupture"	RWST Level
RCS Cooldown and Depressurization	ES-1.2, "Post LOCA Cooldown and Depressurization"	SG Wide or Narrow Range Level, RCS Subcooling or RCS Pressure and Temperature
RCS Cooldown and Depressurization	ES-3.3, "Post SGTR Cooldown Using Steam Dump"	SG Wide or Narrow Range Level, RCS Subcooling or RCS Pressure and Temperature, RCS Pressure, SG Pressure
Manual SI	E-1, "Loss of Reactor or Secondary Coolant" E-3, "Steam Generator Tube Rupture" FR-C.1, "Response to Inadequate Core Cooling"	RCS Pressure; High Head SI Flow Rate, Pressurizer Level
Re-Align AFW	E-0, "Reactor Trip or Safety Injection" FR-H.1, "Response to Loss of Secondary Heat Sink"	SG Wide or Narrow Range Level; AFW Flow Rate
Perform Bleed and Feed	FR-H.1, "Response to Loss of Secondary Heat Sink"	SG Wide Range Level
Isolate Ruptured SG	E-3, "Steam Generator Tube Rupture"	SG Wide or Narrow Range Level
Refill RWST	ECA-1.1, "Loss of Emergency Coolant Recirculation"	RWST Level, Containment Pressure or Containment Sump Level
Terminate SI for SGTR and SS Break	E-1, "Loss of Reactor or Secondary Coolant" ES-1.1, "SI Termination"	Pressurizer Level; RCS Subcooling or RCS Pressure and Temperature
Reactor Shutdown for ATWS	FR-S.1, "Response to Nuclear Power Generation/ATWS"	Power Range Monitor
Align Alternate Feedwater Source	FR-H.1, "Response to Loss of Secondary Heat Sink"	SG Wide or Narrow Range Level; AFW Flow Rate
Align Emergency Boration	FR-S.1, "Response to Nuclear Power Generation/ATWS"	Power Range Monitor
Note: When multiple "Applicable EOP" or multiple "Applicable Instrumentation" entries appear in the table, the multiple entries are applicable to all conditions for that operator action.		

From this assessment, it is apparent that only a limited number of instruments are important in the PRA model to enable the operator to take the appropriate actions to prevent core damage. The key instrumentation is:

- RWST Level,
- SG Wide or Narrow Range Level,
- RCS Subcooling,
- RCS Temperature,
- RCS Wide Range Pressure,
- Pressurizer Level,
- SG Pressure,
- High Head SI Flow,
- Power Range Neutron Flux Monitor,
- SG Wide Range Level, and
- AFW Flow.

It is noteworthy that this list of instrumentation applies to all of the operator actions modeled in the PRA that are not eliminated from further consideration as discussed earlier in this section. Thus, the key instrumentation identified above is independent of the exact numerical value for risk significance of the operator actions in the PRA and is truly a generic conclusion.

As previously noted, the use of RAW importance measures only represents a screening assessment of those components that could be important. The RAW calculation assumes that the component is removed from service completely and is therefore unavailable in the PRA model. In reality, a reduced test and maintenance regime for a particular instrument might, in the extreme, reduce its reliability, but would not cause the instrument to become unavailable with a 100% certainty.

A.6 LERF ASSESSMENT

No quantification of the important equipment or operator actions to prevent large early releases is provided in the PRA database discussed above. However, some insights into the important operator actions for preventing large early releases may be obtained from PRA LERF assessments. A PRA LERF assessment models the consequences of core damage accidents and provides a quantification of the large early release frequency, or LERF. No operator actions to prevent fission product releases are explicitly modeled in most current LERF assessments. From a wide range of PRA studies for PWRs, it is known that there are three types of operator actions that can impact LERF: 1) operator actions to preserve the remaining fission product barriers after core damage has occurred, per the plant Severe Accident Management Guidance, or SAMG; 2) operator actions taken before core damage, per the plant EOPs, that do not impact the core damage frequency but that help to preserve the remaining fission product barriers; and 3) operator actions that are taken to prevent core damage for containment bypass sequences that, in turn, also impact LERF, because a large fraction of bypass core damage sequences are LERF sequences.

It is generally accepted that LERF is dominated by containment bypass sequences. For a typical Pressurized Water Reactor (PWR), 95 to 99% of the LERF is from SGTR core damage sequences and interfacing system LOCA core damage sequences. Containment isolation failures and early containment failures generally contribute less than 1% to LERF.

A qualitative assessment of the impact of the SAMG actions for each of the LERF contributors was completed based on engineering judgment. Each of these is examined separately for insights into instrumentation importance. Late containment failures are also examined based on their inclusion in Regulatory Guide 1.174 for maintaining defense in-depth.

In the following discussions, it is important to understand that while high Core Exit Temperature is an EOP indication, the PRA success criteria commonly use high Core Exit Temperature as the definition of core damage. Thus, high Core Exit Temperatures are not high risk significant for prevention of core damage (since it has already occurred when high Core Exit Temperatures are indicated). However, high Core Exit Temperatures are the primary indication used to transition from the EOPs to the SAMG and are therefore considered in the prevention of large early releases.

Bypass from SGTR – The important operator actions to prevent a LERF condition should be nearly identical to those required to prevent core damage, since many SGTR core damage sequences can be binned as LERF states. There are two unique SAMG strategies for the mitigation of fission product releases for the SGTR that are not included in the strategies for preventing core damage: reducing RCS pressure (per the FR-C.1 procedure in the EOPs or the SAMG) to minimize or terminate releases and filling the SG to cover the tube rupture location to scrub fission products coming from the RCS. Entry to the SAMG is also uniquely based on a high Core Exit Temperature indication. Therefore, the Core Exit Temperature, RCS pressure and SG level are important instrumentation obtained from risk insights for the SGTR bypass LERF sequences.

Bypass from ISLOCA – The important operator actions to prevent a LERF condition should be nearly identical to those required to prevent core damage since most ISLOCA core damage sequences can be binned as LERF states. The only unique SAMG strategy for the mitigation of fission product releases for the ISLOCA that is not included in the strategies for preventing core damage is reducing RCS pressure (per the FR-C.1 procedure or the SAMG) to minimize or terminate releases. Entry to the SAMG is also uniquely based on a high Core Exit Temperature indication. Therefore, Core Exit Temperature and RCS pressure are important instrumentation obtained from risk insights for the ISLOCA bypass LERF sequences.

Unisolated Containment – The important operator action to prevent a LERF condition for an unisolated containment LERF state is the manual closure of any containment isolation valves that failed to close automatically. Therefore, containment isolation valve position indication is important instrumentation obtained from risk insights for the unisolated containment LERF sequences.

Early Containment Failure – Early containment failures for PWRs are typically very small or negligible contributors to LERF and are driven by the union of the "tails" of high pressure melt ejection (HPME) containment pressure loads and the containment fragility (containment pressure capability) estimates. The only unique SAMG strategy for mitigation of fission product releases for the early containment failure states that is not included in the strategies for preventing core damage is reducing RCS pressure (per the FR-C.1 procedure) to minimize containment loads from HPME events. Therefore, RCS pressure is important instrumentation obtained from risk insights for the early containment failure LERF sequences.

Late Containment Failure – Late containment failures states are primarily driven by slow containment overpressurization by steam or noncondensible gases. The only unique SAMG strategy for the mitigation of fission product releases for the late containment failure states that is not included in the strategies for preventing core damage is venting the containment to prevent a catastrophic failure of the containment. Containment venting is reserved as a "last resort" action and is only implemented when containment pressures approach the point where the containment integrity may be challenged. Therefore, the ability to measure containment pressure well beyond the design basis value is important instrumentation obtained from risk insights for the late containment failure LERF sequences.

The only other late containment failure mode of any significance is from a hydrogen burn many hours after core damage has occurred. A further investigation of this contributor shows that it is almost exclusively associated with station blackout initiating events with no long term power recovery. For these cases, all DC power for instrumentation would also be lost due to battery depletion. Thus, a late containment failure mode is not subject to instrumentation importance considerations.

A.7 EXTERNAL EVENTS ASSESSMENT

Risk assessments for each plant will include the consideration of external events, internal flooding and shutdown. In some case, the external events risk assessments are based on detailed PRA models similar to the internal events assessments. However, many plants rely on screening risk assessment techniques that conservatively identify safe shutdown equipment lists. The Fire Induced Vulnerability Evaluations (FIVE) and Seismic Margins Analyses (SMA) are typical of this approach. In these cases, risk importance measures (e.g., RAW values) for the operator actions to use the equipment modeled in those risk assessments are not available. However, this information is not required for the evaluation of the PAM instrumentation as discussed below.

The dominant core damage sequences for fire initiating events are typically fires that lead to a loss of emergency switchgear and/or loss of all emergency AC and all emergency DC power buses. For sequences involving the failure of emergency AC power, the important operator actions are the same as for internal initiating events involving the loss of AC power. For the loss of DC power, this would result in the unavailability of all control room instrumentation that is important for diagnosing and responding to the event. From the perspective of important instrumentation for operator actions, there would typically be no unique operator actions for fire initiating events based on instrumentation. Therefore, there is no instrumentation importance input from the fire initiating event PRA. For some plants, a dominant fire initiated core damage event is a control room fire that results in the loss of all secondary system decay heat removal capability and therefore requires bleed and feed cooling. In this case, the operator actions are identical to those already considered for the internal initiating events. That is, the risk important operator action for this event is based on SG wide range level to initiate bleed and feed cooling.

For seismic risk assessments, the dominant seismic core damage sequences typically involve either a loss of the ultimate heat sink, a loss of all emergency AC and DC power, or a station blackout. In the case of a loss of all emergency DC power, there is no instrumentation to guide operator actions and therefore no risk importance for instrumentation. For the loss of the ultimate heat sink or a loss of all emergency AC power, the sequences are very similar to a station blackout already considered in the internal events PRA. In this case operator actions to control SG level and initiate a cooldown and depressurize the RCS are risk important operator actions. These actions are already designated as risk important operator actions from the internal events PRA.

Based on the above discussions, the fire and seismic risk assessments do not identify any new insights with respect to instrumentation risk importance.

A.8 CONCLUSIONS

From a risk perspective, the following instrumentation has been determined to have a high degree of importance (i.e., RAW > 2.0) for preventing core damage, according to a composite PRA model of all Westinghouse NSSS plants (only the highest RAW operator action value is shown):

- RWST Level (median RAW = 10.35),
- SG Wide or Narrow Range Level (median RAW = 4.05),
- RCS Subcooling (median RAW = 4.05),
- RCS Temperature (median RAW = 4.05),
- RCS Pressure (median RAW = 4.05),
- Pressurizer Level (median RAW = 4.05),
- SG Pressure (median RAW = 4.05),
- High Head SI Flow (median RAW = 3.05),
- Power Range Neutron Flux Monitor (RAW = 2.49),
- SG Wide Range Level (median RAW = 2.46), and
- AFW Flow (median RAW = 2.46).

The following instrumentation has been determined to have a relatively high degree of importance for preventing or mitigating a large early release, according to the assessment of LERF contributors for Westinghouse NSSS plants:

- RCS Pressure,
- Containment Isolation Valve Position, and
- Containment Pressure

All other instrumentation has a negligible risk importance based on the PRA results.

APPENDIX B
NRC/PWROG CORRESPONDENCE

NRCs Requests For Additional Information

PWROGs Responses To The NRCs Requests For Additional Information

PWROG Presentation Material from Sept. 2007 Meeting with NRC

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

March 20, 2006

WOG-06-104

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

Subject: Pressurized Water Reactor Owners Group
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," (LSC-0072 R1/MUHP-3038)

In September 2004, the Pressurized Water Reactor Owners Group (PWROG) (formerly the Westinghouse Owners Group) submitted WCAP-15981-NP (Non-Proprietary), Rev. 0, "Post Accident Monitoring Instrumentation Redefinition for Westinghouse NSSS Plants" for review and approval (Ref. 1). In April 2005 and May 2005, the NRC provided Requests for Additional Information (RAIs) for WCAP-15981 (Ref. 2, 3 and 4). Attachment1 to this letter provides the RAI responses and Attachment 2 provides the changes to WCAP-15981 based on the RAI responses. As noted in the very last RAI response in Attachment 1, the PAM Technical Specification mark-ups will be provided to the NRC by April 14, 2006. Following receipt of the Safety Evaluation for WCAP-15981, the WCAP changes contained in Attachment 2 will be incorporated into the approved version and will be issued as WCAP-15981-NP-A, Revision 1.

It should be noted that these RAI responses and WCAP markups being transmitted are identical to the draft RAI responses and WCAP mark-ups that were provided to the NRC on November 15, 2005 (Ref. 5). Per the NRC/WOG monthly Topical Report status call on February 21, 2006, the NRC (G. Shukla) indicated that the Staff had no comments on the draft RAI draft responses and the WCAP mark-ups, and requested that the PWROG formally submit the RAI responses and WCAP mark-ups to the NRC.

These RAI responses and WCAP mark-ups are being provided to support issuance of the draft Safety Evaluation for WCAP-15981 by August 1, 2006 which is consistent with the current Topical Report schedule on the NRC Topical Report web page and the latest communication from the NRC (G. Shukla) to the PWROG PM.

Document Control Desk
U. S. Nuclear Regulatory Commission
WOG-06-104

March 20, 2006

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If you have any questions concerning this matter, please feel free to call Tom Laubham at 412-374-6788.

Sincerely yours,

Tom Laubham approving for T. Schiffley
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in the Electronic Document Management System

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

FPS:TJL:mjl

Attachments

cc: Licensing Subcommittee
Steering Committee
R. A. Gramm, NRC
G. S. Shukla, NRC (via FedEx)
J. D. Andrachek
K. Vavrek
J. Duryea
C. B. Brinkman
J. A. Gresham
PMO

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 to S. DiTommaso, "RAIs on WCAP-15981 - Post Accident Monitoring Instrumentation Re-Definition," April 11, 2005.

3. NRC E-Mail, G. Shukla to S. DiTommaso, "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 to S. DiTommaso, "RAIs on WCAP-15981, Post Accident Monitoring Instrumentation Redefinition " May 26, 2005.

5. WOG E-Mail, T. Laubham to G. Shukla, "Draft RAI Responses and WCAP-15981 Markups,' November 15, 2005.

Attachment 1 to WOG-06-104

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

PLANT SYSTEMS BRANCH RAIs

(Received April 11, 2005 via an e-mail from G. Shukla (NRC) to S. DiTommaso (W))

General Comments:

1. Generic insights from the Westinghouse PRA database are useful, but potentially incomplete, e.g., the database does not reflect risk achievement worth (RAW) rankings based on the latest plant-specific PRA results, plant-specific details regarding the relationship between important operator actions and associated instrumentation, and the resolution of peer review comments. Although generic risk insights could be used in a limited manner, i.e., to add instruments to the standard technical specification, additional assessments at a plant-specific level would be required if instruments are to be removed from technical specifications on the basis of risk. The discussion on methodology implementation needs to better describe the plant-specific risk assessments that are expected to be performed by the utility, and the manner in which the results of these assessments are to be used in the implementation process.

Response:

The significant operator actions, determined by risk importance will be determined from the plant specific PRA when WCAP-15981 is implemented on a plant-specific basis to identify the instrumentation utilized for those operator actions. The generic risk insights were only utilized to identify the key operator actions based on risk importance in the generic methodology and are used to develop the proposed generic PAM Technical Specification contained in NUREG-1431. The plant specific PAM instrumentation will be determined utilizing the plant specific PRA.

WCAP-15981 Sections 3.1, 4.6, 5, 7, and 8 were revised to clarify that the plant specific PRA will be utilized to determine the plant specific PAM instrumentation and the scope and technical adequacy of the PRA information used in the determination.

(The revisions to Sections 3.1, 4.6, 5, 7, and 8 are provided in Attachment 2.)

2. The conclusion of the study regarding the specific instruments that should be added to or removed from the PAM technical specification appears to have been based largely on a qualitative, reclassification of the key instruments (as summarized in Tables 10 and 11) rather than on the basis of the importance of the instrumentation to risk, EOPs, or other factors. Thus, this methodology does not appear to be "risk-informed", and should not be characterized as such. If the report is modified to more clearly de-emphasize the role of risk information in supporting the conclusions, certain information/assessments requested below may not be needed.

Response:

The plant instrumentation was evaluated with respect to the Criteria of 10 CFR 50.36, specifically Criterion 3 and Criterion 4, which are the only criteria applicable to the identifying Post Accident Monitoring (PAM) instrumentation that should be included in the Technical Specifications.

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

The plant instrumentation was also evaluated in WCAP-15981 from a risk perspective based on risk insights obtained from the PRA in terms of the important operator actions identified based on risk importance. Also included under the broad category of risk insights is the use of instrumentation in the Severe Accident Management Guidance (SAMG), and Emergency Plan Implementing Procedures (EPIP). If the plant instrumentation was shown to be important to risk mitigation in the PRA, SAMG, or EPIP, it is concluded that it satisfies Criterion 4 of 10 CFR 50.36 and should be included in the PAM Technical Specification. Therefore risk insights were used solely for the purpose of identifying the instrumentation that satisfied Criterion 4 of 10 CFR 50.36.

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. As discussed above, 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 satisfies Criterion 3 of 10 CFR 50.36 should be classified as Regulatory Guide 1.97 Type A instrumentation. Instrumentation that satisfies Criterion 4 of 10 CFR 50.36 should be classified as Regulatory Guide 1.97 Category I instrumentation. All other instrumentation not included in the PAM Technical Specification should have a lower Regulatory Guide 1.97 classification.

WCAP-15981 Section 3.2 and Appendix A were revised to clarify that the proposed approach is not risk informed in accordance with Regulatory Guide 1.174, but rather uses PRA insights and other applicable information in determining the instrumentation that should be included in the PAM Technical Specification.

(The revisions to Sections 3.2 and Appendix A are provided in Attachment 2.)

3. The addition of three instruments to the PAM technical specification in accordance with the topical report conclusions (steam generator pressure, refueling water storage tank level, and high head safety

injection flow) is consistent with the importance of these instruments in DBAs, EOPs, PRAs, as well as other applications, such as the Emergency Response Data System (ERDS). (Although the conclusion appears to have been based on a reclassification of these instruments, rather than on the importance of the instrumentation to risk, etc., as stated above.) Accordingly, the NRC staff would concur with this conclusion.

Response:

No response required, the Staff agrees with adding these instruments to the PAM Technical Specification.

4. Five of the six instruments that would be relocated from the PAM technical specification in accordance with the topical report conclusions (source range neutron flux, RCS hot and cold leg temperature, reactor vessel water level, and containment sump water level) are significant in EOPs, EALs, and SAMG, as well as required parameters for ERDS. (Again, the conclusion appears to have been based on a reclassification of these instruments, rather than on the importance of the instrumentation to EOPs, etc.). The justification for removing these instruments from the PAM technical specification appears inadequate, given the rôle of these instruments as potentially-important indicators of plant status and event progression.

Response:

See the response to General Comment RAI Number 2 above regarding the Regulatory Guide 1.97 reclassification of the instrumentation and evaluation of the instrumentation with respect to Criteria 3 and 4 of 10 CFR 50.36 to determine whether it should be included in the PAM Technical Specification.

There are two types of instrumentation utilized in the EOPs, PRA, SAMG, and EPIP; "key" instrumentation that is necessary for the operator to effectively diagnose, and mitigate accidents, and "backup" instrumentation that supplements the "key" instrumentation that supports operator actions to recover the plant.

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

- Perform the diagnosis, in accordance with the plant EOPs, of plant conditions required to initiate manual actions required to bring the plant to a safe stable state for DBAs (discussed in the UFSAR) as well as the wider range of potential accident sequences included in the PRA,

- Perform the pre-planned manual actions in accordance with the plant EOPs, for which no automatic control is provided, that are required for safety systems to accomplish their safety function to mitigate DBAs,

- Perform the pre-planned manual actions in accordance with the plant EOPs to bring the plant to a safe stable state for a wide range of accidents included in the PRA, and

- Diagnose plant conditions that may pose a threat to the health and safety of the general public in accordance with the plant SAMG and EPIP.

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

- Verify the indications of the key instrumentation,

- Operate plant systems utilized to achieve a safe shutdown, including the verification of the automatic actuation of safety systems, and

- Operate other systems normally utilized for achieving a safe shutdown condition.

The instrumentation utilized in the EOPs to cue operator actions for which no automatic control is provided satisfies Criteria 3 of 10 CFR 50.36. All other plant instrumentation was evaluated from a risk perspective based on its use in the PRA, SAMG, and EPIP, to determine whether it satisfies Criteria 4 of 10 CFR 50.36. If the instrumentation does not satisfy Criteria 3 or 4 of 10 CFR 50.36, it can be relocated out of the PAM Technical Specification to a licensee controlled document.

The ERDS is covered by another regulation, Appendix E to 10 CFR 50, and is not affected by the evaluation of the PAM instrumentation with respect to Criterion 3 or 4 of 10 CFR 50.36. If instrumentation that is required by Appendix E to 10 CFR 50 for ERDS does not satisfy Criterion 3 or 4 of 10 CFR 50.36, it should not be included in the PAM Technical Specification. While ERDS provides a significant amount of information to the NRC, and may provide information to other licensee offsite facilities, it is not used directly by the plant operators in their role in mitigating the consequences of an accident.

Specific Information Requests:

1. The assessment of PAM instrumentation considered instrumentation important to design basis accidents (DBAs), probabilistic risk assessments (PRAs), emergency operating procedures (EOPs), severe accident management guidance (SAMG), and Emergency Plan Implementing Procedures (EPIPs), but failed to consider those parameters that are required to be transmitted to the NRC via the Emergency Response Data System (ERDS). Although ERDS is not a safety system, consideration of these parameters within the topical report framework would provide additional insights into a decision on whether certain instruments should be added to or removed from technical specifications. Please expand the assessment to include consideration of the ERDS parameters, and reassess the recommendations for relocation of certain instrumentation in view of the role of these instruments in ERDS.

Response:

The ERDS is covered by another regulation, Appendix E to 10 CFR 50, and is not affected by the evaluation of the PAM instrumentation with respect to Criterion 3 or 4 of 10 CFR 50.36. If instrumentation that is required by Appendix E to 10 CFR 50 for ERDS does not satisfy Criterion 3 or 4 of 10 CFR 50.36, it should not be included in the PAM Technical Specification. While ERDS provides a significant amount of information to the NRC, and may provide information to other licensee offsite facilities, it is not used directly by the plant operators in their role in mitigating the consequences of an accident.

2. Several instruments are listed in Table 4 but not included in Table 9, e.g., auxiliary feedwater valve position, containment water level (wide range), residual heat removal flow. Also, high head safety injection is identified in Table 9 but not in Table 4. Please update the tables so they are consistent.

Response:

WCAP-15981 Table 9 was revised to include all of the instruments that are contained in Table 4. Table 4 was revised to include High Head SI Flow.

(The revisions to Tables 4 and 9 are provided in Attachment 2.)

3. In Table 5, it is stated that high risk significance is defined from CDF and LERF risk achievement and risk reduction metrics per Regulatory Guide (RG) 1.174. Although the RG discusses importance measures, it does not define specific values for these metrics for screening purposes. Please provide a more appropriate reference for the selected screening values.

Response:

WCAP-15981 Section A.4 was revised to reference the EPRI PSA Application Guide, NEI-00-04, and Regulatory Guide 1.201 on the use of risk importance measures to determine risk important systems, structures and components.

(The revisions to Section 8 and Appendix A.4 are provided in Attachment 2.)

4. The identification of risk-significant operator actions is based on PRA information compiled within the proprietary Westinghouse PRA database. Please provide a general description of this database, including the type of information contained in the database, the number of plants represented (e.g., total, by RCS design, by containment type), and the vintage/pedigree of the data (e.g., the portion of the data that is based on IPEs, pre-peer-reviewed updates of the IPE, and post-peer-reviewed updates that address peer review findings). Identify and describe any previous applications where insights/results from this database were also used to support the application.

Response:

The PAM instrumentation proposed to be included in Technical Specification 3.3.3, "PAM Instrumentation," of NUREG-1431 was determined based on generic insights obtained from the Westinghouse NSSS PRA database. The PAM instrumentation that will be included in the plant specific PAM Technical Specifications will be determined utilizing the plant specific PRA to determine the plant specific risk significant operator actions. The Westinghouse NSSS PRA database was only used for demonstrative purposes to identify the instrumentation that would be included in the PAM Technical Specification for a generic, reference plant.

WCAP-15981 Sections 7 and 8 were revised to clarify that the plant specific PRA will be utilized to determine the plant specific PAM instrumentation that should be included in the PAM Technical Specification.

(The revisions to Sections 7 and 8 are provided in Attachment 2.)

5. Please explain why operator actions related to prevention of reactor coolant pump (RCP) seal LOCAs are not among the set of important operator actions identified in Appendix A, given the large contribution to core damage frequency from RCP seal LOCA sequences in some Westinghouse plant PRAs, including those employing the latest ("WOG 2000") seal LOCA methodology.

Response:

Although reactor coolant pump (RCP) seal LOCAs are an important contributor to core damage for Westinghouse PWRs, there are no operator actions modeled in the PRA to protect the RCPs from a seal LOCA. The plant specific abnormal/off-normal procedures provide guidance for restoring RCP seal cooling for those sequences that are susceptible to RCP seal LOCAs (which are sequences involving a loss of all RCP seal cooling). However, if RCP seal cooling is not quickly re-established, then the abnormal/off-normal procedures typically instruct the operators not to re-establish RCP seal cooling in order to avoid additional RCP seal damage due to thermal shock. The preferred recovery strategy in this case is to use an aggressive RCS cooldown to cool down the RCP seals. The time available for restoration of RCP seal cooling is very short (e.g., on the order of minutes) and therefore recovery from a loss of RCP seal cooling event for accident initiators modeled in the PRA is very unlikely. The PRA modeling of a recovery from a loss of RCP seal cooling does not include the diagnosis of the loss of RCP seal cooling and the subsequent unique recovery strategies; the nominal strategy to bring the plant to a safe stable state would provide an adequate RCP seal cool down. Thus, there are no risk significant operator actions for preventing an RCP seal LOCA.

WCAP-15981 Section A.5 was revised to include a discussion of RCP seal LOCAs.

(The revisions to Section A.5 are provided in Attachment 2.)

6. Section 4.2 provides a list of instruments determined from the PRA to be important for preventing core damage. However, this list does not include several additional instruments that are indicated as important to risk in Table 7, i.e., containment sump water level (wide range), containment pressure

(wide range), containment isolation valve position, and component cooling water flow rate. The list/section also does not include several instruments important to LERF, as discussed in Appendix A, e.g., core exit temperature and containment isolation valve position indication. Please provide a more complete accounting and discussion of the risk-significant instruments.

Response:

The instruments listed in Section 4.2 on page 15 are the same instruments that are listed in Appendix A on page A-11. These instruments are risk important in the PRA to cue operator actions necessary to prevent core damage. Tables 7 and 8 were revised to be consistent with Appendix A and Section 4.2. Appendix A also discusses several operator actions that can be important to risk but for which there is no unique set of instrumentation to cue the diagnosis and subsequent implementation of recovery strategies.

The instruments that were determined to be important to cue operator actions to prevent large early releases, as discussed in Appendix A are:

- Core Exit Temperature
- RCS Pressure
- Steam Generator Level
- Containment Isolation Valve Position
- Containment Pressure

The above instruments are included in Table 7 to show their risk importance.

WCAP-15981 Section 4.2 was revised to include a discussion of LERF that is contained on pages A-11 to A-13 of Appendix A.

(The revisions to Section 4.2 and Tables 7 and 8 are provided in Attachment 2.)

7. The identification of risk-significant operator actions and associated instrumentation is based exclusively on consideration of the Westinghouse database/survey for internally-initiated at-power events. Although the impact of externally-initiated events on instrument identification is addressed qualitatively in Appendix A, it appears that no consideration has been given to instrumentation that is important in events during low power operation and shutdown. Under such conditions, certain systems may require manual actuation, and possibly additional instrumentation to provide the necessary cues/information to operators. Please provide an expanded assessment that includes consideration of instrumentation needs during low power operation and shutdown.

Response:

The PAM Technical Specification is applicable in Modes 1, 2, and 3, which ranges from hot full power in Mode 1, down to a keff < 0.99 and an RCS Tavg > 350°F in Mode 3, to provide the indications necessary to mitigate DBAs occurring in these modes. Section 4.1 of WCAP-15981 discusses the operator actions that are assumed in the DBA analysis that are cued from PAM instrumentation. The evaluation of the DBAs considers the Modes of Applicability, i.e., Modes 1 through 3 of the PAM Technical Specification, which includes low power operation. The DBA analyses performed at hot full power bound those analyses at low power, with a few exceptions, such as, the main steam line break at hot zero power core response. Section 4.3 ~~of~~ WCAP-15981 discusses the instrumentation utilized in the EOPs, which are entered following a reactor trip or safety injection. The DBA analysis, EOPs, at-power PRA, SAMG, and EPIP are sufficient to determine the appropriate PAM instrumentation that should be included in the PAM Technical Specification for these Modes. The at-power PRA analyses also bound the PRA analyses at low power.

The PAM Technical Specification is not applicable in the shutdown and refueling Modes (4, 5 and 6), therefore a shutdown PRA is not required to determine the PAM instrumentation required in those Modes.

The third bullet, "Risk Impact," on page 10 of WCAP-15981 was be revised to delete the text "... as well as insights from mode transition (startup and shutdown transition)."

(A revision to page 10 is provided in Attachment 2.)

8. The methodology assumes that all EOP operator actions that are important for preventing core damage are modeled in the plant PRA. The converse would also appear to be true, i.e., any instrument identified in the topical report as important to risk would relate to an important operator action in the EOPs (or in the SAMG). This would imply that any instrument identified in Table 7 as significant to risk should also be identified (in Table 7) as significant to EOPs (or SAMG). This concept has not been consistently applied in Table 7. For example, pressurizer level and RWST level instruments are both indicated to be significant to risk, but neither instrument is indicated to be significant to EOPs or SAMG. A similar relationship may also exist between EOPs and EALs, i.e., an instrument identified in the topical report as important to an EAL might also relate to an important operator action in the EOPs or SAMG. Please provide a more consistent accounting of the significance of the various instrumentation to the EOPs, considering the relationship between the EOP operator actions and the PRA and EALs.

Response:

As stated in Section 4.3 of WCAP-15981, all of the important EOP actions would be identified as important PRA actions or important for the declaration of Emergency Action Levels in the EPIPs. It should also noted that all of the operator actions assumed in the DBA analyses for which no automatic actuations are available should also be associated with important EOP actions.

Table 7 of WCAP-15981 was revised to properly reflect this relationship. Section 4.3 of WCAP-15981 was also revised to reflect the relationship between the DBA assessment and the EOPs.

(The revisions to Section 4.3 and Table 7 are provided in Attachment 2.)

9. Although a high instrument importance in the PRA might constitute a basis for including an instrument in technical specifications, a low importance in the PRA would not necessarily constitute a basis for removing an instrument from technical specifications. Several aspects of the PRA model would need to be critically assessed at a plant-specific level before using a PRA to support such a relaxation. These include the resolution of all relevant PRA quality issues, the relationship between the instrument and the associated human actions, the completeness of the human reliability and systems models in areas related to the instrument/operator actions, and the quantification of these models. Each of these aspects of the PRA would need to be assessed on a plant-specific basis, rather than generically, based on the Westinghouse PRA database/survey. Accordingly, the topical report should be modified to include a clear statement of the plant-specific assessments and reviews of the PRA that each utility would be expected to perform if instrumentation is to be relocated from the technical specifications on the basis of risk significance.

Response:

The plant specific PAM instrumentation to be included in the plant specific PAM Technical Specification will be determined utilizing the plant specific PRA which meets certain requirements for technical adequacy to ensure that the operator actions are adequately considered in the PRA model. Specifically, the licensee's PRA should be based on a PRA whose scope and technical adequacy meets the current industry requirements for risk informed applications.

WCAP-15981 Sections 3.1, 7, and 8 were revised to clarify that the plant specific PRA, which meets certain technical adequacy requirements, will be utilized to determine the plant specific instrumentation that will be included in the plant specific PAM Technical Specifications.

(The revisions to Sections 3.1, 7, and 8 are provided in Attachment 2.)

10. Justify why RCS Subcooling Margin should not be included within the PAM technical specification, given that the topical report found this instrument to be important in DBAs, PRAs, EOPs, and EALs, and that the parameter is also required for ERDS.

Response:

RCS subcooling was found to be an important parameter for diagnosing challenges to core cooling and for ensuring that adequate margins for core cooling were maintained while taking actions to mitigate the consequences of accidents. As discussed on page 38 of WCAP-15981, the RCS Subcooling Monitor indication is derived from a correlation using the core exit temperature and RCS pressure. While the RCS Subcooling Monitor is typically the primary indication relied upon by the plant operating staff in the EOPs, the operators are trained to independently validate the RCS

Subcooling Monitor indication from the RCS temperature (via the core exit thermocouples) and RCS pressure indications. Since both the core exit temperature and RCS pressure indications are already considered to be important instruments and are proposed to be included in the PAM Technical Specification, the operators have a highly reliable and available means of determining RCS subcooling. Since a means of determining RCS subcooling is available using the instrumentation included in the proposed PAM Technical Specification, it is not necessary to include the indication provided by the RCS subcooling instrumentation in the PAM Technical Specification.

Additionally, see the responses to General Comment RAI Number 4 and Specific Information Request RAI Number 1 regarding ERDS.

11. The discussion in Section 8 states that plant-specific implementation of the topical report methodology only requires a confirmation of the generic evaluations contained in the report. For the PRA portion of the assessment, the only guidance provided is that the utility should ensure that the PRA Peer Review findings (for the internal events) have been addressed. In the NRC staff's view, a more comprehensive evaluation would be needed at the plant-specific level if instrumentation is to be relocated on the basis of risk significance. To determine whether additional instrumentation should be included in technical specifications, the evaluation would include: (1) generation and evaluation of plant-specific importance listings (the generic assessment described in the topical report may be incomplete since the 2002 survey did not include "RAW" importances, and since important human actions from the latest plant-specific PRA may not have been captured in the Westinghouse PRA database), (2) identification of instrumentation associated with any additional important operator actions, and (3) confirmation that the instrumentation needed to support risk important operator actions has been appropriately considered. To determine if specific instrumentation can be relocated to licensee controlled documents, the evaluation would include an assessment of: (1) the relationship between the instrument and the associated human actions, (2) the completeness of the human reliability and systems models in areas related to the instrument/operator actions, and (3) the adequacy of the quantification of these models. The utility evaluation would also include consideration of the plant-specific risk analyses for external events and low power/shutdown. The discussion on methodology implementation needs to better describe the plant-specific risk assessments that are expected to be performed by the utility, and the manner in which the results of these assessments are to be used in the implementation process.

Response:

Sections 3.2, 7, and 8 were revised to clarify that a plant specific assessment of the DBA analyses, PRA, EOPs, SAMG and EPIPs, using the methodology described in this report is required, to determine the plant specific PAM Technical Specification instrumentation. Thus, the quantitative and qualitative assessments in this report only serve to demonstrate the generic methodology to be used in a plant specific evaluation.

Sections 3.1 and 8 were revised, as discussed in the response to Specific Information Request RAI Number 9 above, to include requirements on PRA technical adequacy for assessing the instrumentation to be included in the PAM Technical Specification.

Section 8 was revised to identify the details of a plant specific PAM instrumentation evaluation by adding a flowchart that illustrates the process that a licensee would use to determine whether a specific instrument should be included in the PAM Technical Specification or relocated to a licensee controlled document.

As discussed in the response to RAI Number 7, the PAM Technical Specification is not applicable in Modes 4, 5 and 6. Therefore, insights from shutdown or transition PRA assessments do not need to be considered in the evaluation of the PAM instrumentation.

(The revisions to Sections 3.1, 3.2, 7 and 8 are provided in Attachment 2.)

12. Please provide a flowchart or logic diagram depicting the process that a utility would be expected to follow to determine whether a specific instrument should be included in the PAM technical specification or licensee controlled documents. Important considerations within the process should include: (1) how the instrument relates to Criterion 3 and 4 of 10 CFR 50.36 (c)(2)(ii), (2) whether the instrument supports important operator actions in the plant-specific internal events PRA, external events risk assessment, and shutdown risk assessment, (3) whether the instrument is important for other purposes, such as EOPs, EALs, etc. (if this is in fact a consideration in the decision), (4) how the instrument would be classified using the RG 1.97 classification approach (if this is in fact a consideration in the decision), and (5) whether alternate indications are available in lieu of the specific instrument (if this is in fact a consideration in the decision). The flowchart should also depict the process that a utility would be expected to follow to: (1) confirm that the plant-specific risk models (for internal events, external events, and shutdown events) are of suitable quality for this application (i.e., peer review findings that could impact the identification and ranking of important operator actions and associated instrumentation have been resolved), (2) confirm that the plant-specific risk models are of sufficient detail to reflect the risk significance of the specific operator actions and instruments, and (3) document the results of the risk evaluation.

Response (Overall):

A flowchart that illustrates the process that a licensee would use to determine whether a specific instrument should be included in the PAM Technical Specification or relocated to a licensee controlled document was provided in the revised Section 8.

Response to 12 (1):

See the response to General Comment Number 2 regarding how it is determined whether the PAM instrumentation satisfies to Criteria 3 and 4 of 10 CFR 50.36 (c)(2)(ii).

Response to 12 (2):

See the discussion on pages A-7 through A-11 of WCAP-15981 regarding the instrumentation that supports important operator actions in the internal events PRA.

See the discussion in the second paragraph on page 56 and pages A-13 and A-14 of WCAP-15981 regarding external events.

See the response to Specific Information Request RAI Number 7 above regarding a shutdown risk assessment.

Response to 12 (3):

See the response to General Comment RAI Number 4 regarding the use of instrumentation in the EOPs, PRA, SAMG, and EPIP. See the discussion on pages 21 through 23 of WCAP-15981 regarding the use of EALs to determine what PAM instrumentation should be included in the PAM Technical Specification.

Response to 12 (4):

There is no Regulatory Guide 1.97 classification that is used as input to determine whether the PAM instrumentation should be included in the PAM Technical Specification. The assessment described in this report is an independent assessment of instrumentation importances based on current accident management understanding and knowledge. After the assessment with respect to Criteria 3 and 4 of 10 CFR 50.36 is complete, the Regulatory Guide 1.97 classification should be updated for consistency with the assessment performed. Those instruments that satisfy Criterion 3 of 10 CFR 50.36 would then be classified as a Regulatory Guide 1.97 Type A variable. Similarly, those instruments that satisfy Criterion 4 of 10 CFR 50.36 would then be classified as a Regulatory Guide 1.97 Category I variable.

Response to 12 (5):

The alternate indications that were identified for the PAM instrumentation proposed to be included in the PAM Technical Specification are discussed in Table 13 on page 49 of WCAP-15981. The identification of alternate indications does not input into the determination of whether PAM instrumentation should be included in the PAM Technical Specification. The alternate indications were identified to allow unit operation to continue beyond 30 days with one inoperable PAM channel or beyond 7 days with two inoperable PAM channels, in lieu of a unit shutdown for those PAM functions that were determined to have alternate indications. This provision is allowed by Required Action B.1 for one inoperable PAM channel, and Required Action F.1 for two inoperable Reactor

Vessel Water Level or Containment Area Radiation (High Range) channels in Technical Specification 3.3.3, "PAM Instrumentation," in NUREG-1431.

(The revisions to Section 8 are provided in Attachment 2.)

ELECTRICAL AND INSTRUMENTATION AND CONTROLS BRANCH RAIs

(Received May 16, 2005 via an e-mail from G. Shukla (NRC) to S. DiTommaso (W))

1. Regulatory Guide (RG) 1.97 grouped the variables to be monitored during and after an accident into five types. RG 1.97 recommends that Category 1 instrumentation provide the operator with information on the key variables for (1) plant specific Type A variables, (2) the accomplishment of four Type B plant safety functions (reactivity control, core cooling, reactor coolant system (RCS) integrity, and containment integrity), (3) the potential for breach or actual breach of three Type C fission product barriers (fuel cladding, reactor coolant pressure boundary, and containment), (4) the operation of three Type D safety systems and other systems important to safety (primary containment system, secondary system, and auxiliary feedwater system), and (5) the magnitude of release of radioactive materials of one Type E variable (containment radiation). These functions, potential for breach, system status, and magnitude are referred to as functions in this request for additional information.

   WCAP-15981 examines each variable that is in the NUREG-1431 (STS) post accident monitoring (PAM) technical specifications (TSs) or in plant specific PAM TSs and determines the highest function that each variable served, but does not examine each RG 1.97 function and how the RG 1.97 variables serve each function. The WCAP appears to not consider that some variables serve multiple functions and multiple types and therefore might also fall into multiple categories.

   The format of the WCAP should be revised to address each RG 1.97 function under each type, address the key variables for each function, and appropriately categorize each key variable. If RG 1.97 designated a variable as a key variable for a particular function, but the WCAP analysis suggests it should not be a key variable, the WCAP should identify the key variables for that function with appropriate justification. Additionally, the WCAP should provide the new categorization of any proposed downgrade of variables, along with appropriate justification as it relates to each function.

Response:

Technical Specification 3.3.3, "PAM Instrumentation," in NUREG-1431, "Standard Technical Specifications Westinghouse Plants," contains a Reviewer's Note that states: "Table 3.3.3-1 shall be amended for each unit as necessary to list: 1) All Regulatory Guide 1.97, Type A instruments and 2) All Regulatory Guide 1.97, Category 1, non-Type A instruments in accordance with the unit's Regulatory Guide 1.97, Safety Evaluation Report."

As discussed on pages 3 and 4 of WCAP-15981, WCAP-15981 was prepared to evaluate the non-Type A, Category 1 instrumentation to determine whether it should be included in the PAM Technical Specification based on the staff's conclusion in a 1988 NRC letter which states: "the staff is unable to confirm the Owners Groups' conclusion that Category 1 Post-Accident Monitoring Instrumentation is not of prime importance in limiting risk (Criterion 4)."

The PAM instrumentation was evaluated in WCAP-15981 with respect to the Criteria of 10 CFR 50.36 (specifically Criteria 3 and 4), which are the only Criteria applicable to the PAM instrumentation Technical Specification. The EOPs were utilized in the methodology contained in WCAP-15981 to determine the instrumentation that is used to perform specific manual actions assumed in the DBA analyses for which there is no automatic actuation of equipment provided. The instrumentation utilized in the EOPs to perform specific manual actions that are assumed in the DBA analyses satisfy Criterion 3 of 10 CFR 50.36, and should be included in the PAM Technical Specification.

The PAM instrumentation was also evaluated in WCAP-15981 from a risk perspective based on risk insights obtained from the PRA in terms of the important operator actions identified based on risk importance, instrumentation utilized in key SAMG operator actions, and in key operator actions from the EPIP. If the PAM instrumentation was shown to be important to risk mitigation in the PRA, SAMG, or EPIP, it satisfies Criterion 4 of 10 CFR 50.36 and should be included in the PAM Technical Specification.

The 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. The evaluation of the current instrumentation utilized in accident management is consistent with the reclassification of the containment hydrogen monitors based on their use in accident management as discussed in the 10 CFR 50.44 rulemaking. Therefore a complete reclassification of the PAM instrumentation that identifies all of the functions and types is not necessary to evaluate the PAM instrumentation to determine whether it satisfies Criteria 3 and/or 4 of 10 CFR 50.36. The only changes to the Regulatory Guide 1.97 classifications that are necessary are for the instrumentation whose assessment with respect to Criteria 3 and 4 of 10 CFR 50.36 differs from that in the original Regulatory Guide 1.97 classification. It should be noted that no instrumentation upgrade due to the reclassification to Regulatory Guide 1.97 Type A or Category I is required to include this instrumentation in the PAM Technical Specification.

2. The WCAP recommends that RCS pressure, core exit temperature, pressurizer level, steam generator level (wide range), and steam generator pressure be reclassified as Type A. However, since Type A variables are plant specific, it is not clear how the WCAP justifies the recommendation that a variable be a generic Type A. Please explain the concept of generic Type A variables and how it would be applied on a plant specific basis.

Response:

The instrumentation that should be included in the PAM Technical Specification based on 10 CFR 50.36 Criterion 3 were determined based on a generic evaluation of the DBA analyses as discussed in

Section 4.1 of WCAP-15981 and a review of the WOG Emergency Response Guidelines (ERGs) as discussed in Section 4.3 of WCAP-15981. The evaluation that identified that the instrumentation satisfies Criterion 3 of 10 CFR 50.36 would also need to reflect that that the instrumentation should be classified as a Regulatory Guide 1.97 Type A variable for consistency. Therefore, any instrumentation that satisfies 10 CFR 50.36 Criterion 3 should also be identified as a Regulatory Guide 1.97 Type A variable. The plant specific evaluation to determine which instrumentation satisfies Criteria 3 and/or 4 of 10 CFR 50.36 and the Regulatory Guide 1.97 classifications will be performed on a plant specific basis based on a review of the plant specific DBA analyses and EOPs as part of the plant specific implementation of WCAP-15981.

WCAP-15981 Sections 7 and 8 were revised to clarify that the plant specific DBA analyses and EOPs will be utilized to determine the plant specific PAM instrumentation to be included in the plant specific Technical Specifications.

(The revisions to Sections 7 and 8 are provided in Attachment 2.)

3. Some of the variables that the WCAP recommends for inclusion in the Technical Specifications (refueling water storage tank (RWST) level, high head safety injection (SI) flow, and steam generator pressure) are not currently classified as Category 1 in RG 1.97. Therefore, at some plants these instruments might not meet the redundancy, environmental qualification, seismic qualification, and power source criteria of Category 1 instrumentation. Is the WCAP proposing that licensees be forced to upgrade this instrumentation or request a deviation?

If this instrumentation is not upgraded but is included in the Technical Specifications, changes in the number of channels or the listing of alternate channels of instrumentation must be written into plant specific Technical Specifications, which would deviate from the STS philosophy. Please discuss these topics.

Response:

NUREG-1431 currently requires that Regulatory Guide, 1.97, Type A instruments and Category I, non-Type A instruments be included in the PAM Technical Specification. Prior to the implementation of NUREG-1431, plant specific PAM Technical Specifications included/include Regulatory Guide 1.97, non-Category 1, non-Type A instruments and even some non-Regulatory Guide 1.97 instruments. The current PAM Technical Specifications for plants who have not converted to NUREG-1431 may also contain Regulatory Guide 1.97, non-Category 1, non-Type A instruments, as well as non-Regulatory Guide 1.97 instruments. The Technical Specification Actions address instrument inoperability for those Regulatory Guide 1.97, non-Category I, non-Type A instruments, non-Regulatory Guide 1.97 instruments, as well as the Regulatory Guide 1.97 Type A and non-Type A, Category I instruments.

Including these Regulatory Guide 1.97 instruments (refueling water storage tank (RWST) level, High Head safety injection (SI) flow, and steam generator pressure) in the PAM Technical Specification if they are not classified as Regulatory Guide 1.97, Category I instruments is acceptable, since the PAM Technical Specification Actions will address instrument inoperability for all of the PAM Instrumentation, including any non-Category I PAM Instrumentation.

The Bases for current Action (A.1) in Technical Specification 3.3.3, "PAM Instrumentation," of NUREG-1431 discusses PAM functions that only have one required channel and other non-Regulatory Guide 1.97 instrumentation available to monitor the function as a basis for the 30 day Completion Time. Additionally, the current Actions in Technical Specification 3.3.3 of NUREG-1431 allow unit operation to continue beyond 30 days with one PAM channel inoperable (Required Action B.1), and beyond 7 days with two PAM channels inoperable for the Reactor Vessel Water Level and Containment Area Radiation (High Range) functions (Required Action F.1) based on having alternate indications. The Bases for Required Action F.1 in Technical Specification 3.3.3 of NUREG-1431 discuss that alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation may be temporarily installed, and that unit operation may continue beyond 30 days with both PAM channels inoperable for these two functions. Specification 5.6.7, "Post Accident Monitoring Report," in NUREG-1431 discusses the use of pre-planned alternate methods of monitoring inoperable PAM instrumentation (Conditions B and F).

WCAP-15981 does not require an upgrade in classification, or a deviation to the Regulatory Guide 1.97 classification if any of these instruments are Regulatory Guide 1.97, non Category I, non-Type A instruments, since the Technical Specifications will address instrument inoperability.

4. For the reactivity control function the WCAP appears to disagree with the RG 1.97 statement: "If two or more instruments are needed to cover a particular range, overlapping of instrument span should be provided." RG 1.97 recommends Category 1 neutron flux instrumentation with a range of 10-6% to 100% full power for the reactivity control function. However the WCAP recommends that source range neutron flux, which monitors 10-6% to 1%, should be a backup variable because it provides diagnostics for maintaining subcriticality during RCS cooldown and depressurization. Isn't it important for the operator to determine that the reactor is actually shutdown?

How would the operator verify that the reactor is actually shutdown without information from the source range neutron flux instrumentation?

Response:

For all accident sequences, immediate subcriticality would be achieved without operator actions by the insertion of control rods into the core, which is part of the plant design basis for all DBAs except the large break LOCA. For the large break LOCA, the blowdown forces may result in an inability to insert control rods, and the rapid injection of large quantities of highly borated water from the accumulators and RWST ensures that subcriticality is achieved with no operator actions. Failure to achieve initial subcriticality by control rod insertion would be diagnosed by the plant operators using the power range neutron flux indication as prescribed by the plant EOPs. As discussed in Sections 4 and 5 of WCAP-15981, the generic assessment has determined that the power range neutron flux indication can be an important PAM instrument that satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii). Additionally, the power range neutron flux indication does not satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii), since automatic features ensure that initial subcriticality is achieved.

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 the RCS cooldown and depressurization. Thus, the optimal recovery guidelines in the EOPs do not rely on the source range monitor. The source range monitor is only used in the Functional Restoration Guideline portion of the EOPs for the diagnosis of a potential loss of core shutdown margin. The analyses and evaluations that support the PRA model typically show that this action is screened out of the PRA model based on the low probability for accident sequences in which recriticality could occur in a success path. Therefore, the source range neutron flux indication does not satisfy Criterion 4 of 10 CFR 50.36 (c)(2)(ii). Additionally, the source range neutron flux indication does not satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii), since automatic features ensure that initial subcriticality is achieved.

Therefore the source range neutron flux indication is not included in the generic list of instruments in the proposed PAM Technical Specification in NUREG-1431.

5. For the core cooling function the WCAP appears to disagree with the RG 1.97 statement: "The measurement of a single key variable may not be sufficient to indicate the accomplishment of a given safety function. Where multiple variables are needed to indicate the accomplishment of a given safety function, it is essential that they each be considered key variables and measured with high-quality instrumentation." RG 1.97 recommends Category 1 instrumentation for RCS hot leg temperature, RCS cold leg temperature, RCS pressure, and reactor vessel water level to monitor the core cooling function. However the WCAP recommends that RCS hot leg temperature, RCS cold leg temperature, and reactor vessel water level should be backup variables because they provide either backup core cooling information or diagnostic core cooling information and that core exit temperature as monitored by the core exit thermocouples, as monitored by the core exit thermocouples (CETs), should be used as the key variable for the core cooling function. The WCAP does not mention RCS pressure in the core cooling function discussion. The core cooling function discussion should discuss all of these variables.

   How would the operator verify that core cooling is taking place based on core exit temperature alone?

   How would the operator determine that the core is covered or uncovered without knowing the reactor vessel water level?

   The WCAP says of RCS subcooling, "The inputs to the RCS subcooling monitor are the CETs for RCS temperature and the wide range RCS pressure indication for RCS pressure." However, a number of plants use information from the CETs, RCS hot leg temperature, RCS cold leg temperature, and RCS pressure as inputs to RCS subcooling. Since RCS hot leg temperature and RCS cold leg temperature are input to RCS subcooling, shouldn't they be considered primary information and, therefore, be classified as Category 1?

Contrary to the WCAP assessment, RCS hot leg temperature and RCS cold leg temperature are currently classified as Type A at approximately 41 Westinghouse units. Please explain this contradiction.

Response:

Section 5.1 of WCAP-15981 discusses the evaluation of the Core Exit Temperature, RCS hot leg temperature, RCS cold leg temperature, and reactor vessel water level instrumentation with respect to whether it should be included in the PAM Technical Specification, by determining whether they satisfy Criterion 3 and/or Criterion 4 of 10 CFR 50.36. Only the Core Exit Temperature and RCS Pressure indications are proposed to be included in the PAM Technical Specification based on satisfying Criterion 4 of 10 CFR 50.36. This assessment concluded that only the core exit thermocouples can provide a direct indication of core cooling. Hot leg and cold leg RTD indications can be affected by accident conditions that may mask the actual core cooling condition (e.g., the impact of SG reflux cooling in the hot leg RTDs). Therefore, the RCS hot leg temperature, RCS cold leg temperature, and reactor vessel water level instrumentation are proposed to be relocated from the PAM Technical Specification to a licensee controlled document since they do not satisfy Criterion 3 or 4 of 10 CFR 50.36. Not all of the instrumentation utilized in the EOPs is included in the PAM Technical Specification.

The most direct inputs to determining RCS subcooling are the core exit thermocouples (CETs) for RCS Temperature, and wide range RCS Pressure indications. Both of these indications are proposed to be included in the PAM Technical Specification by WCAP-15981. Even if the actual plant RCS Subcooling Monitor uses the RCS hot leg or cold leg temperature inputs for determining RCS subcooling, the core exit thermocouples are an adequate alternate and preferred indication for this determination. The core exit thermocouples provide the highest indication of the RCS fluid temperature since they are located at the core outlet region of the reactor vessel, and therefore provide a minimum subcooling indication.

RCS hot leg temperature and RCS cold leg temperature are currently classified as a Regulatory Guide 1.97 Type A indication based on the current Regulatory Guide 1.97 evaluations of PAM Instrumentation that may not have determined how this instrumentation is utilized in the DBA analyses and EOPs. If it is determined that this instrumentation is a Regulatory Guide 1.97 Type A indication based on the plant specific DBA analyses and EOPs, then this instrumentation would be included in the PAM Technical Specification for that plant.

6. For the RCS integrity and reactor coolant pressure boundary functions the WCAP appears to disagree with the RG 1.97 statement: "The measurement of a single key variable may not be sufficient to

indicate the accomplishment of a given safety function. Where multiple variables are needed to indicate the accomplishment of a given safety function, it is essential that they each be considered key variables and measured with high-quality instrumentation." RG 1.97 recommends Category 1 RCS pressure, containment sump water level (wide range), and containment pressure instrumentation to monitor the RCS integrity function. RG 1.97 also recommends Category 1 RCS pressure, containment pressure, and containment sump water level (wide range) instrumentation to monitor the reactor coolant pressure boundary function. However, the WCAP recommends that containment sump water level (wide range) should provide information on the status of SI from the RWST and should, therefore, be Category 2. The WCAP should discuss the role of containment sump water level (wide range) instrument in the RCS integrity and reactor coolant pressure boundary functions.

How would the operator determine the amount of water in containment to support switchover to recirculation without knowing the containment sump water level?

How would the operator determine if a break in RCS piping is inside or outside containment without containment sump water level?

How is the WCAP recommendation for RWST level being included in the TSs affected based on automatic switchover to recirculation, semiautomatic switchover to recirculation, or manual switchover to recirculation?

Contrary to the WCAP assessment, containment sump water level (wide range) is currently classified as Type A at approximately 18 Westinghouse units. Please explain this contradiction.

Response:

The diagnosis of RCS pressure boundary integrity as a critical safety function in the EOP functional restoration guidelines does not rely on the containment water level indication. A loss or potential loss of RCS integrity is diagnosed in the EOPs from RCS pressure, RCS temperature (for pressurized thermal shock concerns only) and pressurizer level. All of these parameters are indicated by instrumentation that has been determined to satisfy Criterion 3 or 4 of 10 CFR 50.36 for other reasons (e.g., RCS pressure wide range, core exit temperature and pressurizer level) and proposed to be included in the PAM Technical Specification. Therefore, the PAM instrumentation provides a means for the diagnosis of a loss, or potential loss, of RCS pressure boundary integrity. However, since the automatic actuation of systems and components in response to a loss of RCS integrity is included in the design of Westinghouse PWRs, pressurizer level and RCS pressure are not important for diagnosing a loss of the RCS pressure boundary. Similarly, RCS pressure and RCS temperature are used in the EOPs as indicators of a potential loss of RCS integrity due to pressurized thermal shock concerns. However, the potential for a loss of RCS integrity due to pressurized thermal shock is not a DBA, and has been shown to be a negligible PRA contributor to risk (Reference: Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Criteria in the PTS Rule (10CFR50.61), Draft, December 2002, Accession Number ML030090632). Therefore, containment sump level wide range is not used as an indicator of RCS pressure boundary integrity.

The use of the containment sump level wide range indication in accident management is discussed on page 37 in Section 5.1 of WCAP-15981.

As discussed on page 38 of Section 5.1, the RWST level alarms and/or indication are utilized to obtain information regarding the switchover to the emergency core cooling and containment spray recirculation mode.

For the diagnosis of a LOCA inside containment, the containment sump water level is used in the EOPs, as the third indication (after the containment pressure and containment radiation indications). Following a reactor trip and automatic initiation of safety injection (based on low pressurizer pressure) any one of the three indications is sufficient for the operators to diagnose a LOCA condition; the other two indications may provide confirmation of the LOCA. For example, in the case of a small LOCA (e.g., less than 2 inch equivalent diameter) that results in an SI signal, the first indication would likely be a small increase in containment radiation levels and potentially a slight increase in containment pressure (e.g., less than 1 psig). Depending on the break location, the break water might not quickly reach the containment sump and provide an indication of a containment sump level increase. Additionally, for the diagnosis of a LOCA outside containment, following reactor trip and automatic initiation of safety injection the containment sump water level indication (e.g., no increase in containment sump level) is the third indication utilized in the EOPs (after a decreasing RCS pressure indication and an increasing auxiliary building radiation indication). For the most probable LOCA outside containment, according to PRA studies, is the interfacing system LOCA outside containment in the low pressure RHR piping connected to the RCS. This would likely result in a containment sump water level indication, since the flow from the relief valves on the RHR piping is routed back to pressurizer quench tank whose rupture disk would quickly open and release fluid to the containment. Thus "no containment sump level" is not a reliable indicator of a LOCA outside containment. In summary, the diagnosis of RCS pressure boundary integrity as a critical safety function in the EOP functional restoration guidelines does not rely on the containment sump water level indication.

Depending on the plant design, some plant designs provide automatic switchover to the recirculation mode, while other designs provide semi-automatic switchover to the recirculation mode, and some designs require manual switchover. The automatic or semi-automatic switchover function is provided by the Engineered Safety Features Actuation System (ESFAS) function based on RWST level, and is included in the ESFAS Instrumentation Technical Specification for those plant designs. Some of the automatic switchover to the recirculation mode designs also require an RWST Low-Low Level signal coincident with a Containment Sump Level- High signal to prevent spurious switchover. The RWST Low-Low Level signal coincident with a Containment Sump Level-High Function is included in the ESFAS Instrumentation Technical Specification for those plant designs. If the switchover to the recirculation mode is provided manually, it is first cued from the RWST alarm and not the RWST level indication, nor the Containment Sump Water Level (Wide Range) indication. If the switchover to the recirculation mode is provided manually utilizing the RWST Level indication, the RWST level indication would be included in the PAM Technical Specification as proposed by WCAP-15981, and identified as satisfying Criterion 3 of 10 CFR 50.36 and identified as a Regulatory Guide 1.97, Type A variable. Table 10 was revised to reflect that the RWST Level is a Regulatory Guide 1.97, Type A variable for those plants that utilize its indication for manual or semi-automatic switchover to the recirculation mode.

WCAP-15981 evaluated the Containment Sump Water Level (Wide Range) instrumentation with respect to whether it should be included in the PAM Technical Specification by determining whether it satisfies Criterion 3 and/or Criterion 4 of 10 CFR 50.36.

The Containment Sump Water Level (Wide Range) instrumentation is proposed to be relocated from the PAM Technical Specification to a licensee controlled document since it does not satisfy Criterion 3 or 4 of 10 CFR 50.36, however, it will still be available for monitoring purposes. Not all of the instrumentation utilized in the EOPs is included in the PAM Technical Specification.

Containment sump water wide range is currently classified as a Regulatory Guide 1.97 Type A variable based on the current Regulatory Guide 1.97 evaluations of PAM Instrumentation that may not have determined how this instrumentation is utilized in the DBA analyses and EOPs. If it is determined that this instrumentation is a Regulatory Guide 1.97 Type A indication based on the plant specific DBA analyses and EOPs, then this instrumentation would be included in the PAM Technical Specification for that plant.

WCAP-15981 Sections 4.1, 5.1 and Table 10 were revised to be consistent with the information provided in this response.

(The revisions to Sections 4.1, 5.1 and Table 10 are provided in Attachment 2.)

7. For the operating status of the auxiliary feedwater (AFW) system, the WCAP appears to disagree with the RG 1.97 statement: "The measurement of a single key variable may not be sufficient to indicate the accomplishment of a given safety function. Where multiple variables are needed to indicate the accomplishment of a given safety function, it is essential that they each be considered key variables and measured with high-quality instrumentation." RG 1.97 recommends Category 1 condensate storage tank level instrumentation to monitor the operating status of the AFW system. However, the WCAP has recommended that condensate storage tank level should be downgraded to Category 2 and should provide information to indicate whether a continued steam generator heat sink can be maintained and long term AFW system operating status.

It appears that the WCAP justifies using AFW flow in lieu of condensate storage tank level as the key variable for the AFW system status. Please explain the relationship between these variables with respect to AFW system status.

How would the operator verify that there is sufficient water to feed the AFW system without knowing the condensate storage tank level?

Contrary to the WCAP assessment, condensate storage tank level is currently classified as Type A at approximately 16 Westinghouse units. Please explain this contradiction.

Response:

As discussed in Section 5.1 of WCAP-15981, the AFW Flow and Steam Generator Level instrumentation provide the indications used in the EOPs for diagnosing issues related to the performance of the AFW system. The primary diagnosis used in the EOPs for inadequate AFW performance is the AFW flow rate indication. The second symptom used in the EOPs to diagnose inadequate AFW performance is a decreasing SG water level. CST level is not used in the diagnosis of inadequate AFW performance.

While the CST level instrumentation is the primary indication of the ability to continue to provide AFW flow to the steam generators to provide a secondary side heat sink for decay heat removal from the reactor core, CST refill is a long-term action that is typically not required in the first 16 to 20 hours after an accident. For the DBA events, the plant would either be on normal RHR cooling (for non-LOCA events) or ECC recirculation in a time frame well before the CST inventory is exhausted. Operator actions to refill the CST based on low CST level indication are modeled in some PRA models. The results of the PRA assessment in Appendix A of WCAP-15981 show that the CST level indication has a low risk significance. The low risk significance is based on: a) the low probability that natural circulation would be the only means of maintaining the core in a safe stable state following an accident, and b) the long time to deplete the CST during which CST refill would have begun and been monitored based on the available CST level indications (including local indications) if long term natural circulation decay heat removal was selected as the long term safe stable state.

WCAP-15981 evaluated the condensate storage tank level instrumentation with respect to whether it should be included in the PAM Technical Specification by determining whether it satisfies Criterion 3 and/or Criterion 4 of 10 CFR 50.36. Since it does not satisfy Criterion 3 or 4 of 10 CFR 50.36, the condensate storage tank level instrumentation is proposed to be relocated from the PAM Technical Specification to a licensee controlled document. If it is determined that the condensate storage tank level indication is risk important form a plant specific PRA, then this instrumentation should be included in the PAM Technical Specification for that plant.

The condensate storage tank level is currently classified as a Regulatory Guide 1.97 Type A indication based on the current Regulatory Guide 1.97 evaluations of PAM Instrumentation that may not have determined how this instrumentation is utilized in the DBA analyses and EOPs. If it is determined that this instrumentation is a Regulatory Guide 1.97 Type A indication based on the plant specific DBA analyses and EOPs, then this instrumentation would be included in the PAM Technical Specification for that plant.

WCAP-15981 Section 5.1 was revised to be consistent with the information provided in this response.

(The revisions to Section 5.1 are provided in Attachment 2.)

8. The WCAP includes a discussion of alternate instrumentation for power range neutron flux, high head SI flow, containment area radiation (high range), steam generator water level (wide range), and auxiliary feedwater flow. Some of these alternate instruments either are not currently classified as Category 1, the WCAP recommends for downgrade from Category 1, or are not part of the RG 1.97 program. To take credit in the Technical Specifications for alternate instrumentation, the alternate instrumentation should also be in the Technical Specifications. Please revise the discussion of alternate instrumentation to include only Category 1 instruments that are included in the Technical Specifications.

Response:

Technical Specification 3.3.3, "PAM Instrumentation," in NUREG-1431 allows the use of Regulatory Guide 1.97 non-Category I instrumentation as alternate instrumentation for inoperable PAM instrumentation and the alternate instrumentation does not have to be included in the Technical Specifications as discussed below.

The Bases for current Action (A.1) in Technical Specification 3.3.3, "PAM Instrumentation," of NUREG-1431 discusses PAM Functions that only have one required channel and other non-Regulatory Guide 1.97 instrumentation available to monitor the function as a basis for the 30 day Completion Time. Additionally, the current Actions in Technical Specification 3.3.3 of NUREG-1431 allow unit operation to continue beyond 30 days with one PAM channel inoperable (Required Action B.1), and beyond 7 days with two PAM channels inoperable for the Reactor Vessel Water Level and Containment Area Radiation functions (Required Action F.1) based on having alternate indications. The Bases for Required Action F.1 in Technical Specification 3.3.3 of NUREG-1431 discuss that alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation (High Range) may be temporarily installed, and that unit operation may continue beyond 30 days with both PAM channels inoperable for these two functions. Specification 5.6.7, "Post Accident Monitoring Report," in NUREG-1431 discusses the use of pre-planned alternate methods of monitoring inoperable PAM instrumentation (Conditions B and F).

The justification for the use of alternate instrumentation contained in Section 6 of WCAP-15981 was revised to provide a more complete discussion of the adequacy of the instrumentation proposed to be used as alternate instrumentation for the PAM instrumentation included in the PAM Technical Specification.

(The revisions to Section 6 are provided in Attachment 2.)

TECHNICAL SPECIFICATION SECTION RAIs

(Received May 26, 2005 via an e-mail from G. Shukla (NRC) to S. DiTommaso (W))

1. "The STS (NUREG-1431) Post Accident Monitoring Instrumentation, Table 3.3.3-1 requires all Regulatory Guide 1.97, Type A instruments and all Regulatory Guide 1.97, Category 1, non-Type A instruments in accordance with the unit's Regulatory Guide 1.97, Safety Evaluation Report. WCAP-15981 proposes changes that will revise the list to include generic Type A instruments and revise the categorization of certain Category 1 instruments. Provide a markup of STS LCO 3.3.3 and the LCO 3.3.3 Bases to show the proposed changes to the staff precedent in the STS."

Response:

The Technical Specification and Bases markups that reflect the changes to Technical Specification 3.3.3 of NUREG-1431 that are proposed by WCAP-15981 will be provided by April 14, 2006.

PWROG
PWR Owners Group

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

August 10, 2006

OG-06-259

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

Subject: Pressurized Water Reactor Owners Group
<u>Additional Revisions to WCAP-15981-NP, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," (LSC-0072 R1/MUHP-3038)</u>

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 to S. DiTommaso, "RAIs on WCAP-15981 - Post Accident Monitoring Instrumentation Re-Definition," April 11, 2005.

3. NRC E-Mail, G. Shukla to S. DiTommaso, "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 to S. DiTommaso, "RAIs on WCAP-15981, Post Accident Monitoring Instrumentation Redefinition," May 26, 2005.

5. WOG 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 to R. Lutz, "WCAP-15981 (PAM)," May 10, 2006.

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 mark-ups were transmitted to the NRC on March 20, 2006 (Ref. 5).

Enclosure 1 to this letter provides additional revisions as mark-ups to WCAP-15981-NP that address the additional items contained in Ref. 6, that were discussed during a telecon held on May 25th, 2006 between Bob Palla (NRC) and Bob Lutz and Jim Andrachek (Westinghouse). Please provide these WCAP revisions to Bob Palla.

Following receipt of the Safety Evaluation for WCAP-15981, the WCAP changes contained in Enclosure 1 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 Tom Laubham at 412-374-6788.

Sincerely yours,

Tom Laubham approving for T. Schiffley
Electronically Approved Records Are Authenticated
in the Electronic Document Management System

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

FPS:TJL:mjl

Enclosure

cc: Licensing Subcommittee
Steering Committee
S. Peters, NRC (via FedEx)
Bob Palla, NRC
J. D. Andrachek
K. J. Vavrek
R. J. Lutz
C. B. Brinkman
J. A. Gresham
PMO

CDF (a Level 1 PRA), as well as early fission product releases (a LERF assessment). A ~~and at least a~~ qualitative assessment of late containment failures, and core damage risks from seismic, fire and ~~shutdown~~ other important external initiating events should also be performed for any PAM instrument proposed to be relocated from the Technical Specifications. The qualitative assessment of external events ~~and shutdown~~ risks will generally result in a more conservative approach in determining the safety significance of components, compared to a quantitative PRA assessment. A review of the important operator actions from several Westinghouse NSSS plants with a fully quantified external events PRA, as discussed in Appendix A, has shown that the important operator actions that are based on control room instrumentation in the external events PRA are the same as those already determined to be significant from the internal events PRA.

3.2 METHODOLOGY

The overall methodology used for assessing the importance of instrumentation to be included in the PAM Technical Specification is similar to the methodology (Reference 10) developed and used in the successful elimination of Post Accident Sampling System (PASS) requirements that specifically addressed offsite emergency radiological protection aspects important to safety.

Although the approach used in this report uses the results of PRA assessments, it is not a risk-informed application in accordance with Regulatory Guide 1.174. Rather than focusing on the five elements of a risk informed approach as specified in Regulatory Guide 1.174, this approach directly assesses the importance of instrumentation with respect to the Criteria of 10 CFR 50.36 (c)(2)(ii). This direct assessment uses the plant DBA analyses, PRA, EOPs, SAMG and EPIP as the basis for assigning importance to the instrumentation. Therefore, the methodology used in this report is more prescriptive than a risk informed approach.

PAM instrumentation is intended to provide indications of plant parameters that are the basis for important operator actions to bring the plant to a safe stable state in the event of an accident. The information available to make this determination includes:

- Design Basis Accidents – While most DBAs rely on instrumentation that provides a signal to automatically initiate systems and components to bring the plant to a safe stable state, there are also several key operator actions assumed in the DBA analyses.
- Probabilistic Risk Assessment – The PRA models a number of operator actions to bring the plant to a safe stable state and prevent core damage.
- Emergency Operating Procedures – The EOPs provide guidance for the operator response to an accident, based on instrumentation indications of plant parameters. The EOPs are the basis for the PRA and DBA operator action modeling.

8 IMPLEMENTATION

The plant specific implementation of this methodology contained in this report requires a plant specific evaluation of the accident management application of PAM instrumentation contained in the: 1) Design Basis Accidents, 2) Emergency Operating Procedures, 3) Probabilistic Risk Assessment, 4) Severe Accident Management Guidelines, and 5) Emergency Plan as discussed in this report.

The generic list of PAM instrumentation proposed to be included in Technical Specification 3.3.3 of NUREG-1431, and those instruments proposed to be relocated from the plant specific Technical Specifications to LCDs must be confirmed on a plant specific basis by reviewing the plant specific DBA analyses, PRA, EOPs, SAMGs, and EPIP.

The overall process to be used by licensees to identify the PAM instrumentation that should be included in the Technical Specifications is discussed in Table 14 and shown ~~provided~~ in Figure 2. This process is identical to that described in this report, except that plant specific information would be used in place of generic information to determine the PAM instrumentation to be included in the plant specific PAM Technical Specifications. As discussed in Section 3.2, this is not a risk informed application to be evaluated using the guidance in Regulatory Guide 1.174. The methodology directly assesses the importance of instrumentation with respect to the criteria of 10 CFR 50.36 (c)(2)(ii) rather than focusing on the five elements of a risk informed approach as specified in Regulatory Guide 1.174. The methodology uses risk assessment as one element of the overall method to determine the instrumentation to be included in the PAM Technical Specification.

The first step in the process (Step 1 in Table 14) is to identify the operator actions that are assumed in DBA analyses using the criteria in Table 5. These operator actions satisfy Criterion 3 of 10 CFR 50.36 because no automatic actuation of equipment is included in the plant design for these actions.

The next steps are to identify the risk important operator actions from the plant PRA using the criteria in Table 5. This part has two distinct steps: a) verification of the PRA technical adequacy for this application (Step 2 in Table 14), and b) use of the PRA information to identify operator actions based on instrumentation that satisfies Criterion 4 of 10 CFR 50.36 (Step 3 in Table 14).

As discussed in Section 3.1, the licensee should ensure that the internal events PRA is technically adequate for this application. PRA technical adequacy of the internal events PRA for risk informed applications is typically addressed through the PRA peer reviews and self assessments using a variety of guidance, including the American Society of Mechanical Engineers (ASME) PRA Standard (Reference 7), Nuclear Energy Institute (NEI) PRA Peer Review Process Guidance (Reference 8) and/or Regulatory Guide 1.200 (Reference 9). Since this is not a risk informed application as described in Regulatory Guide 1.174, only a limited assessment of the ~~The more extensive~~ PRA technical

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adequacy is required for this ~~requirements contained in References 7 and 9, while assuring a more robust PRA, are not required for this applicatio~~application. The limited assessment of the PRA technical adequacy only needs to consider the areas of the accident sequence analysis and the human reliability analysis to assure that treatment of operator actions based on plant instrumentation is appropriate. In particular, the licensee should confirm that all operator actions potentially impacted by the subject instruments have been identified, that the treatment of these operator actions in the PRA is appropriate (including the human error probability values and dependencies), and that there are no peer review comments that can affect the conclusions regarding instrument importance.

~~n since the determination of the PAM Technical Specification instrumentation does not rely solely on the PRA and the CDF and LERF values determined from the PRA model.~~

~~The first step in the process is to identify all operator actions that are assumed in DBA analyses using the criteria in Table 5. These operator actions satisfy Criterion 3 of 10 CFR 50.46 in that no automatic actuation of equipment is included in the plant design for these actions.~~

~~The next step is to identify the risk important operator actions from the plant PRA using the criteria in Table 5.~~ The RAW and F-V risk importance measures ~~with appropriate numerical values are~~can ~~be~~ 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 ~~or safe shutdown equipment lists~~. As noted in Section 3.2 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.

~~The next step is to identify the instrumentation associated with the important design basis and PRA operator actions. This step establishes the relationship between the instrument and the associated human actions. This would typically involve the use of the plant emergency and off-normal / abnormal procedures to identify any instrumentation cues for initiating these actions, as well as instrumentation cues used to confirm that the operator action has been successfully completed.~~

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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 ~~to identify 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 set of instrumentation, while others can only be cued from specific instrumentation. This step would therefore focus on the minimum set required to support the key operator actions.~~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, can be relocated from the PAM Technical Specification to a licensee controlled document. At this point, 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 the 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.

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.

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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 ○ Operator actions for which no automatic actuation of equipment is provided
2	PRA technical adequacy	• Summary of PRA ○ Scope (Level 1, LERF, external events) ○ Peer reviews ○ Update history ○ PRA updating process • PRA reflects as-built, as-operated design ○ 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 • 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
6	Identification of instrumentation to be included or relocated from the PAM Technical Specification	• Focused evaluation of the adequacy of the PRA and HRA treatment of operator actions associated with any variables (instrumentation) to be relocated from the PAM Technical Specification • Identify appropriate changes to the Regulatory Guide 1.97 classifications to be consistent with the inclusion in, or relocation from, the PAM Technical Specification

56c

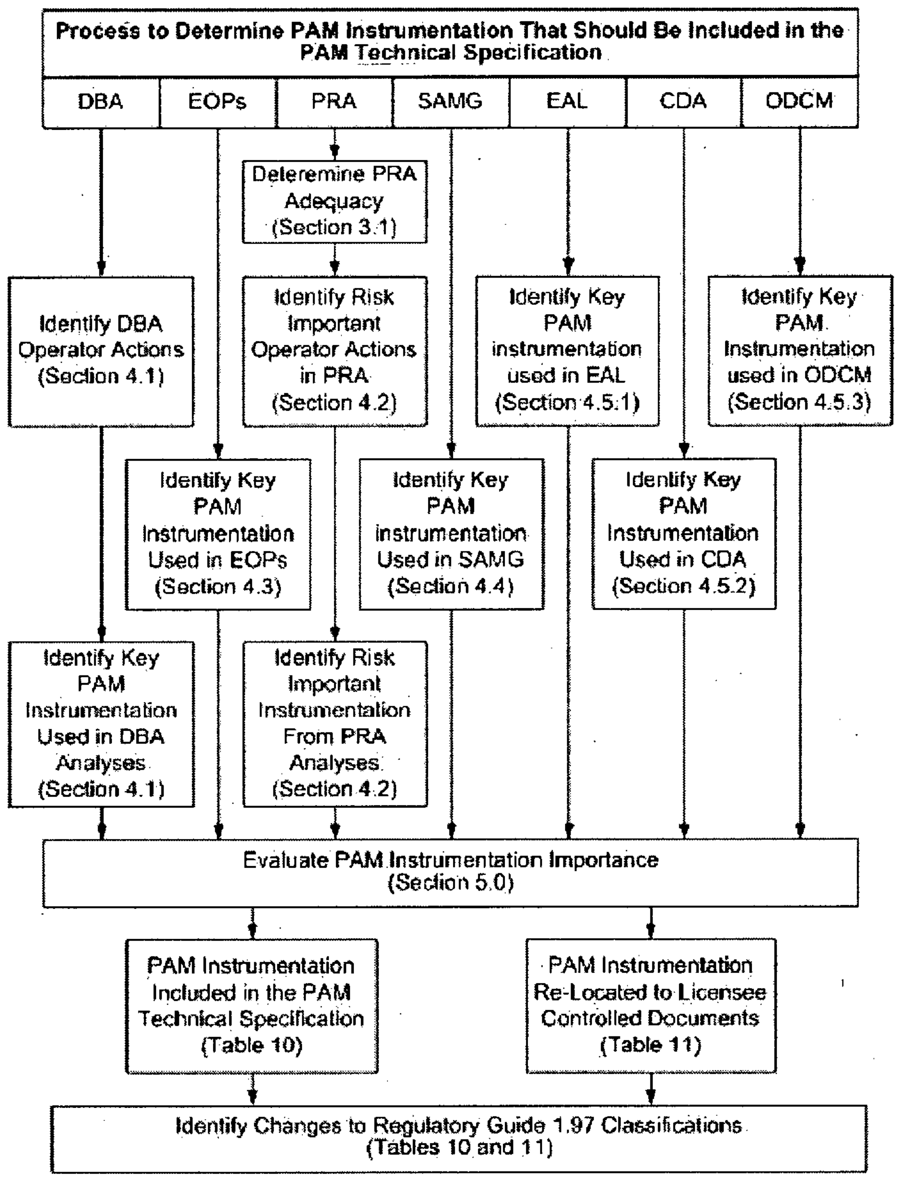

Figure 2 Process to Determine PAM Instrumentation That Should Be Included in the PAM Technical Specification

NRC Email, R. Palla to R. Lutz, "WCAP-15981 (PAM)," May 10, 2006

Follow-Up Comments on WCAP-15981 Revisions and RAI Responses

Response to General Comment 1 - The comment stated that additional assessments at a plant-specific level would be required if instruments are to be removed from technical specifications on the basis of risk. This has not been addressed in the revision. In addition to calling for an overall assessment of the adequacy of the PRA/HRA to support this application, where specific instruments are proposed for deletion from the technical specifications, the methodology should call for focused assessments of the adequacy of the PRA/HRA treatment with regard to those instruments. As part of the assessments, the licensee would confirm that all operator actions potentially impacted by the subject instruments have been identified, that the treatment of these operator actions in the PRA is appropriate (including the human error probability values and any dependencies on other failures or instruments), and that there are no peer review comments that can alter the finding regarding instrument importance. This should be considered for inclusion in Sections 3.2 and 8, and Appendix A of the WCAP.

Response to RAI 12 - The RAI requested that the flowchart/process description indicate whether/how the availability of alternate indications would be considered in the decision to relocate instruments from the technical specifications. Although clarified in the RAI response, this clarification has not been included in the revision, e.g., in Section 8 or on the flowchart.

Response to RAI 12 - The RAI requested that the flowchart depict the process that a licensee would be expected to follow to: (1) confirm that the plant-specific risk models are of suitable quality for this application, (2) confirm that the plant-specific risk models are of sufficient detail to reflect the risk significance of the specific operator actions and instruments, and (3) document the results of the risk evaluation. There is only a brief discussion of items 1 and 2 on revised p. 8a, and there is no discussion on documentation. Attachment 1 provides a summary of the types of information that should be considered for inclusion in a licensee submittal referencing this WCAP report. This should be considered for inclusion in Sections 3.2 and 8, and Appendix A of the WCAP.

p. 8b, line 3 - Mention of shutdown initiating events and shutdown risks should be deleted consistent with the response to RAI 7.

p. 56a, paragraph 4 - The last sentence, indicating that the more extensive PRA technical adequacy requirements contained in References 7 and 9 are not required for this application, conflicts with the first two sentences, indicating that the licensee should ensure that the PRA is technically adequate for this application using guidance including References 7 and 9. Further clarification is needed regarding which requirements must be met.

p. 56a, paragraph 6 - Mention of shutdown equipment lists should be deleted consistent with the response to RAI 7.

p. 56a and b - Several steps in the implementation process appear to be missing from the implementation description, e.g.:

- In Appendix A, there is a step to eliminate operator actions based on certain instrument considerations. It seems like this step should be included here.

- Actions/instruments related to LERF and external events are candidates for inclusion. A step addressing these additional items should be included here.

- Consistent with the above Response to General Comment 1, a step should be included to call for focused assessments of the adequacy of the PRA/HRA treatment for any instruments proposed for deletion from the technical specifications.

p. 56c - The two boxes on the flowchart related to the use of PRA should include a reference to Appendix A of the WCAP, as well as Section 4.2.

Attachment 1 to NRC Email, R. Palla to R. Lutz, May 10, 2006

PRA-Related Items for Inclusion in Applications Referencing WCAP-15981

1. A general PRA description addressing:

   - scope (Level 1, 2, 3, external events, margins versus PRA)
   - peer reviews
   - update history, including version peer reviewed, version(s) in which peer review comments were addressed, and version used for PAM application
   - licensee PRA updating and QA process

2. A description of the most relevant peer reviews, characterization of peer review findings, and status of resolution of peer review comments

   - listing of all unresolved F&Os that potentially impact PAM application

3. A conclusion regarding PRA quality assessment for PAM application, and verification that the quality is acceptable for the application

   - PRA reflects as-built, as-operated design; any recent plant modifications and operational changes not reflected in the PRA don't impact the plant-specific PAM application
   - peer review comments resolved or don't impact plant-specific PAM application
   - PRA and HRA is sufficiently complete and applicable for evaluating the risk associated with the PAM application (high level conclusion – more focused evaluation addressed under item 9)

4. Listings of the important human actions identified based on risk achievement worth (RAW) and Fussell-Vesely (FV) importance values for CDF and for LERF, along with these values

5. Additions to the list of important human actions based on review of results from the plant-specific external event assessments, or verification that the plant-specific risk assessments don't result in identification of additional risk-significant operator actions or variables/instruments and the rationale for this result (based on plant-specific considerations)

6. A listing of variables/instruments related to the important human actions

   - for each variable/instrument considered in the risk evaluation, an explanation of how it was related or mapped to a PRA model element

7. Summary tables showing important indications for accident management, and the context in which they are important, e.g., DBA, EOP, SAMG, PRA, E-Plan (similar to Tables 7 and 8 in WCAP-15981)

8. A summary table describing variables/instruments added to or relocated from the technical specifications, and the specific bases for each change (similar to Table 1 in Attachment D to the BVPS lead plant application)

9. For any variables/instruments to be deleted from the technical specifications based on their lack of risk significance, the results of a focused evaluation of the adequacy of the PRA and HRA treatment (or lack of treatment) of operator actions associated with those variables/instruments

10. For any variables/instruments to be deleted from the technical specifications based on their lack of risk significance, a discussion of how the reliability and availability of these instruments will be monitored and assessed, e.g., under the maintenance rule, other licensee program, or performance measurement strategy

11. A discussion of how the key RG 1.174 principles are impacted and maintained by the proposed changes (plant specific version of the 5 items discussed on p. 10 of Section 3.2)

    - meets regulatory requirement
    - consistent with the defense-in-depth philosophy
    - maintains sufficient safety margins
    - increases in CDF and risk are small
    - use of performance measurement strategy to monitor the change

PWROG
PWR Owners Group

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

June 28, 2007

OG-07-292

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

Subject: Pressurized Water Reactor Owners Group
<u>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)</u>

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.

U.S. NRC Document Control Desk June 28, 2007
OG-07-292 Page 2

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 DiMuzio approving 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"

1

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.

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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.

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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

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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.

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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.

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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.

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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)

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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.

Note: Attachment 2 contains a tracked-changes version of this WCAP, and was therefore omitted.

PAM NRC Meeting Presentation, September 20, 2007

NRC-PWROG Meeting to Discuss Determining the PAM Instrumentation that Satisfies the Criteria of 10CFR50.36

Denny Buschbaum, TXU
PWROG Vice-Chairman
Bert Yates, Ameren, PWROG
Licensing Subcommittee Chairman

September 20, 2007

NRC-PWROG Meeting to Discuss Determining the PAM Instrumentation that Satisfies the Criteria of 10CFR50.36

Agenda

- Background
- Objective of the Topical Report
- Correspondence Chronology
- Methodology to Determine the PAM Instrumentation that Satisfies Criterion 4 of 10CFR50.36
- HRA Evaluation of Operator Actions for PAM Instrumentation Relocated from the Tech Specs
- Instrumentation Utilized in ERG CSF Status Trees
- Plant Specific HRA Evaluation
- NUREG-1431 Changes
- Summary and Conclusions

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NRC-PWROG Meeting to Discuss Determining the PAM Instrumentation that Satisfies the Criteria of 10CFR50.36

Background

- WCAP-11618, "MERITS Program- Phase II Task 5 Criteria Application," documents the results of the application of the NRC's Interim Policy Statement Criteria to NUREG-0452
- NRC letter from Murley (NRC) to Newton (WOG) dated May 1988 documents the NRC staff review of the application of the NRC's Interim Policy Statement Criteria to NUREG-0452
- WCAP-11618 concluded that only Reg. Guide Type A instrumentation satisfied any of the NRC's Interim Policy Statement Criteria (Criterion 3)
- WCAP-11618 also concluded that Reg. Guide non-Type A, Category 1 instrumentation did not satisfy any of the NRC's Interim Policy Criteria including Criterion 4

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NRC-PWROG Meeting to Discuss Determining the PAM Instrumentation that Satisfies the Criteria of 10CFR50.36

Background (cont.)

- The Staff's review documented in the May 1988 letter agreed that Reg. Guide 1.97 Type A instrumentation satisfied Criterion 3 and concluded that the Staff was unable to confirm the OGs conclusion that Category 1 PAM instrumentation was not of prime importance in limiting risk (Criterion 4)
- The basis for the conclusion regarding Category 1 PAM instrumentation was that "recent" PRAs have shown the risk significance of operator recovery actions required a knowledge of Category 1 variables, and that "recent" severe accident studies showed significant potential for risk reduction from accident management

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Background (cont.)

- The Staff concluded that the OGs should develop further risk-based justification in support of relocating any or all Category 1 variables from the STS
- Tech Spec 3.3.3, "PAM Instrumentation," in NUREG-1431 contains a Reviewer's Note that requires Table 3.3.3-1 to include all Reg. Guide 1.97 Type A instruments, and all Reg. Guide 1.97 Category 1, non-Type A instruments in accordance with the unit's Reg. Guide 1.97 SER

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Objective of the Topical Report (WCAP-15981)

- The objective of the Topical Report is to determine which PAM instrumentation satisfies Criteria 3 and/or 4 of 10CFR50.36

- **Criterion 3:** An SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier
- The safety analyses were reviewed to determine which PAM instrumentation provides the primary information that is essential for the direct accomplishment of specified manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBA or transients
- This PAM instrumentation satisfies Criterion 3 of 10CFR50.36

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Objective of the Topical Report (cont.)

- **Criterion 4:** An SSC which operating experience or probabilistic risk assessment has shown to be significant to public health and safety
- A methodology was developed to address the Staff request contained in the 1988 NRC letter to develop a risk-based justification to support relocating Category 1 variables that do not satisfy Criterion 4 of 10CFR50.36 out of the STS
- This PAM instrumentation satisfies Criterion 4 of 10CFR50.36
- This Topical Report will support the addition of an additional Reviewer's Note to Tech Spec 3.3.3 of NUREG-1431 that will provide licensees with an option of revising their plant specific PAM Table to reflect the PAM instrumentation that satisfies Criteria 3 and/or 4 of 10CFR50.36 based on the methodology contained in WCAP-15981

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Correspondence Chronology

- The Topical Report was submitted for NRC review and approval on September 17, 2004 (WOG-04-474)
- Responses provided to the first set of RAIs on March 20, 2006 (WOG-06-104)
- WCAP revisions provided in response to a teleconference on August 10, 2006 (OG-06-259)
- Responses provided to the second set of RAIs on June 28, 2007 (OG-07-292)
- Responses provided to the third set of RAIs on August 22, 2007 (OG-07-376)

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Methodology to Determine the PAM Instrumentation that Satisfies Criterion 4 of 10CFR50.36

- Identification of variables that show significant potential for risk reduction from accident management comes from three sources
  - PRAs to date focus on prevention of core damage and therefore are useful in identifying variables that reduce risk by preventing core damage
  - SAMG is useful in identifying variables that reduce risk by preventing fission product releases to the environment following core damage
  - EPIP is useful in identifying variables that reduce risk by timely implementation of offsite emergency preparedness activities
- The focus of the remainder of this presentation is on the PRA and the prevent of core damage

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Methodology (Continued)

- The relationship of the PRA and PAM instrumentation must be understood to determine the risk significance
  - The PRA models all design basis and beyond design basis initiating events
  - The PRA accident scenarios go beyond design basis single failures and assumptions
  - The PRA models operator actions to mitigate the accident scenarios to determine a realistic CDF

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Methodology (Continued)

- Operator actions modeled in the PRA are typically based only on Emergency Operating Procedure (EOP) or Abnormal Operating Procedure (AOP) steps
  - Actions assumed in the DBA analyses, and
  - Actions to respond to beyond design basis events
- Not all EOP or AOP steps are modeled in the PRA
  - Highly reliable operator actions, e.g., control AFW to maintain SG level
  - Second and third level recovery actions based on their impact on risk
- All operator actions that have an impact on risk are modeled in the PRA
  - Owners Group Peer Reviews ensure technical adequacy of the HRA

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Methodology (cont.)

- The human reliability analysis (HRA) in the PRA assumes that instrumentation is available to cue the operator to take prescribed actions
  - The HRA typically does not assign an instrumentation failure rate
  - The HRA assumes that the instrumentation is available and accurate when required, whether it is included in the Tech Specs or not
    - The instrumentation is monitored once per shift by the operator viewing the control board, e.g., Channel Checks, etc.
    - Important instrumentation has multiple channels
    - The operators are aware of instrumentation out-of-service and the methods to compensate if an event occurs
    - Operator training stresses that instrumentation readings need to be validated by correlations with other instrumentation before any major actions are performed

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Methodology (cont.)

- The availability of instrumentation, in regards to relocation from PAM Tech Specs, to support operator actions modeled in the PRA is inherent in the PRA risk importance determination
  - The RAW value assumes that the operator actions always fails and measures the impact on risk
  - Therefore RAW provides an indication of the impact of instrument unavailability on risk
- If an operator action has a low RAW value then the instrumentation to support that action has low risk significance
- If an operator action is modeled in the PRA, then the RAW value can be used to determine the impact on risk of degraded availability

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HRA Evaluation of Operator Actions for PAM Instrumentation Relocated from the Tech Specs

- For the operator actions modeled in the PRA, the impact of potentially reduced availability is inherent in the risk importance determination
- For operator actions not explicitly modeled in the PRA but are considered to be highly reliable actions, an assessment would be required to determine that the operator action remains highly reliable, even in the event of reduced instrumentation unavailability
  - e.g., SG level is multi-channel and also narrow range and wide range
- For operator actions screened from the PRA based on contribution to risk, an assessment would be required to determine that the risk would not increase if the instrumentation unavailability increases
  - If it is determined that the increase in instrumentation unavailability for these operator actions would increase risk, the instrumentation would be retained in the Tech Specs

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Critical Safety Function (CSF) Status Trees

- CSF Status Trees provide a second level of defense against a transient or an accident progressing to core damage
  - The Optimal Recovery Procedures (the E-Series, including ES and ECA procedures for Westinghouse plants) provide the expected operator actions for design basis accidents and many beyond design basis events
  - The CSF Status Trees would typically only be used for beyond design basis events
  - There are four levels of priority in the CSF Status Trees
    - Only Red paths require immediate attention
    - Orange paths require immediate attention when no Red paths are available
    - Yellow and Green paths are to be addressed at the operator's discretion when time is available
  - Only Red and Orange CSF paths need to be considered

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Instrumentation Utilized in Critical Safety Function (CSF) Status Trees

- Subcriticality CSF Status Tree
  - PR neutron flux is used to indicate a Red Path to FR-S.1 - PR neutron flux is risk significant in the generic PRA and will be retained in PAM TS 3.3.3
  - IR SUR is used to indicate a Red Path to FR-S.1 – IR SUR is not risk significant in the generic PRA (Appendix A) and will not be retained in the PAM TS 3.3.3; IR SUR needs to be addressed in the plant specific implementation to determine its risk significance
  - COTs and CALs (SR 3.3.1.7, SR 3.3.1.8, SR 3.3.1.11) in TS 3.3.1 verify that IR and SR indications are available
  - IR and SR can be removed from TS 3.3.3 but will remain controlled by TS 3.3.1

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Instrumentation Utilized in Critical Safety Function (CSF) Status Trees

- Core Cooling CSF Status Tree
  - Core Exit Temperature (CET) is used to indicate a Red Path to FR-C.1 and Orange Path to FR-C.2 - CET is risk significant in the generic PRA and will be retained in PAM TS 3.3.3
  - Reactor Vessel Level (RVLIS) is used to indicate a Red Path to FR-C.1and an orange path to FR-C.2 when CET is less than the setpoint value – RVLIS is not risk significant in the generic PRA (Appendix A) and will not be retained in the PAM TS 3.3.3; RVLIS needs to be addressed in the plant specific implementation to determine its risk significance

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Instrumentation Utilized in Critical Safety Function (CSF) Status Trees

- Heat Sink CSF Status Tree
  - AFW Flow and SG level are used to indicate a Red Path to FR-H.1 – AFW Flow and SG level are risk significant in the generic PRA and will be retained in PAM TS 3.3.3
    - Heat Sink CSF Status Tree uses SG Narrow Range level as a primary indicator but operators are trained to also use SG Wide Range level
    - FR-H.1 procedure uses SG Wide Range level to initiate Bleed and Feed which is risk significant in the PRA
    - Only SG Wide Range level indication needs to be included in the PAM TS 3.3.3

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Instrumentation Utilized in Critical Safety Function (CSF) Status Trees

- Integrity CSF Status Tree
  - RCS Cold Leg Temperature and RCS Pressure are used to indicate a Red Path and an Orange path to FR-P.1
    - Pressurized Thermal Shock (PTS) is not risk significant in the generic PRA (Appendix A)
    - Additional PTS analyses performed for the upcoming PTS conclude that PTS is not risk significant
    - Cold Leg Temperature indication will not be retained in the PAM TS 3.3.3; Cold Leg Temperature indication needs to be addressed in the plant specific implementation to determine its risk significance
    - OTDT and OPDT COTs and CALs (SR 3.3.1.7, SR 3.3.1.10) verify that Tavg and ΔT indications are available

NRC-PWROG Meeting to Discuss Determining the PAM Instrumentation that Satisfies the Criteria of 10CFR50.36

Instrumentation Utilized in Critical Safety Function (CSF) Status Trees

- Containment CSF Status Tree
  - Containment Pressure is used to indicate a Red Path and an Orange path to FR-Z.1; Containment Pressure is risk significant in the generic PRA and will be retained in PAM TS 3.3.3
  - Containment Sump Level Wide Range is used to indicate an Orange Path to FR-Z.2; prevention of Containment Flooding is not risk significant in the generic PRA (Appendix A)
    - Containment Sump Level Wide Range indication will not be retained in the PAM TS 3.3.3; Containment Level Wide Range indication needs to be addressed in the plant specific implementation to determine its risk significance

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Instrumentation Utilized in Critical Safety Function (CSF) Status Trees

- Inventory CSF Status Tree
  - All Paths are Yellow and therefore do not need to be addressed immediately
  - Pressurizer Level and RVLIS are sued to indicate Yellow Paths; RCS Inventory using FR-I.1 through FR-I.3 are not risk significant in the generic PRA (Appendix A)
    - Pressurizer Level is retained in PAM TS 3.3.3 for other risk significant operator actions; RVLIS indication will not be retained in the PAM TS 3.3.3; RVLIS indication needs to be addressed in the plant specific implementation to determine its risk significance

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Plant Specific HRA Evaluation

- Refer to the handout

NRC-PWROG Meeting to Discuss Determining the PAM Instrumentation that Satisfies the Criteria of 10CFR50.36

NUREG-1431 Changes

- NUREG-1431 contains a Reviewer's Note that requires Table 3.3.3-1 to include all Reg. Guide 1.97 Type A instruments, and all Reg. Guide 1.97 Category 1, non-Type A instruments in accordance with the unit's Reg. Guide 1.97 SER, therefore this is a generic list of instruments that must be implemented on a plant specific basis
- The plant specific lists of instruments varies from the generic list
- An additional Reviewer's Note to Tech Spec 3.3.3 of NUREG-1431 will be added that will provide licensees with an option of revising their plant specific PAM Table to reflect the PAM instrumentation that satisfies Criteria 3 and/or 4 of 10CFR50.36 based on the methodology contained in WCAP-15981
- The plant specific lists determined with the methodology in WCAP-15981 will also vary from the generic list

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Summary and Conclusions

- The methodology in WCAP-15981 will be applied on a plant specific basis to determine the plant specific list of instrumentation to be included in Table 3.3.3-1 based on determining whether the instrumentation satisfies Criteria 3 and/or 4 of 10CFR50.36
- Applying the methodology in WCAP-15981 does not remove any PAM instrumentation from the plant design
- Applying the methodology in WCAP-15981 does not impact the plant specific Reg. Guide 1.97 analysis and SER
- The plant specific list of PAM instrumentation may be different from the generic list based on applying the methodology on a plant specific basis

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Summary and Conclusions (cont.)

- The relocation of PAM instrumentation for operator actions modeled in the PRA that are not risk significant and for highly for highly reliable operator actions that are not risk significant does not impact the HRA, since the RAW value considers a complete failure of the operator action, which corresponds to a complete failure of the instrumentation
- A qualitative assessment will be performed for operator actions not modeled in the PRA to determine whether other EOP or AOP procedures that are modeled in the PRA would be effective in mitigating risk, and those PAM instruments would be included in the Tech Specs
- **Need a summary of the CSF Status Tree review**
- **Need a summary of the plant specific HRA evaluation**

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PWROG PWR Owners Group

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

October 4, 2007

OG-07-443

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

Subject: **Documentation of Agreement Reached at September 20, 2007 Meeting Regarding WCAP-15981-NP, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," (LSC-0072 R1/MUHP-3038)**

References:
1. PWROG Letter, F. Schiffley to Document Control Desk, "Response to Requests for Clarification of June 28, 2007 RAI Responses for WCAP-15981-NP, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," (LSC-0072 R1/MUHP-3038)," OG-07-37, August 22, 2007.

The purpose of this letter is to document the agreement reached between the PWROG and the NRC staff at the September 20, 2007 public meeting regarding the information to be included in the Approved version of WCAP-15981-NP, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants."

Following receipt of the Safety Evaluation for WCAP-15981, the PWROG will revise page 16 of the approved version of WCAP-15981 to incorporate the information on the Westinghouse ERG Critical Safety Function Status Trees presented in the September 20, 2007 public meeting.

The PWROG is awaiting feedback from the NRC Special Projects Branch regarding the request from the NRC Tech Spec Branch to include a markup of Tech Spec 3.3.1, "PAM Instrumentation," in the approved version of WCAP-15981. If we have to include a sample markup of Tech Spec 3.3.1 in the approved WCAP per your request, the letter transmitting the approved version of the WCAP to the NRC will need to state that the sample Tech Spec markup included in the WCAP will be superseded by the associated Traveler when it is submitted for NRC review.

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

Sincerely yours,

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

FPS:CD:las

cc: Licensing Subcommittee
Steering Committee
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