September 17, 2004
WOG-04-474

WCAP-15981-NP, Rev. 0
Project Number 694

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

Subject: Westinghouse Owners Group
Transmittal of WCAP-15981-NP (Non-Proprietary), Rev. 0, "Post Accident Monitoring Instrumentation Redefinition for Westinghouse NSSS Plants" (MUHP-3036)

This letter transmits WCAP-15981-NP (Non-Proprietary), Rev. 0, entitled "Post Accident Monitoring Instrumentation Redefinition for Westinghouse NSSS Plants," dated August 2004. The Westinghouse Owners Group (WOG) is submitting WCAP-15981 in accordance with the Nuclear Regulatory Commission (NRC) licensing topical report program for review and acceptance for referencing in licensing actions. WCAP-15981 provides the technical justification for identifying the Post Accident Monitoring (PAM) instrumentation that should be included in the Technical Specifications for the Westinghouse NSSS plants. The associated changes to NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," will be included in a Technical Specifications Task Force (TSTF) Traveler that will be transmitted separately at a later date.

An initial pre-application meeting was held on December 10, 2003. At that meeting, the background and proposed approach (methodology) to redefine the PAM instruments to be included in the Technical Specifications was presented. As discussed at the meeting, the proposed approach is based on how Regulatory Guide 1.97 instrumentation is currently utilized in accident management. The NRC Staff had no objections to the proposed approach; however, they did mention that IEEE Std 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," was approved by the IEEE in 2002 and that they were considering revising Regulatory Guide 1.97 to incorporate IEEE Std 497-2002.

Subsequent to the meeting, a review of IEEE Std 497-2002 determined that it was developed primarily for the use of microprocessor based instrumentation for the next generation of advanced nuclear plant designs, and can also be utilized by current plants considering digital upgrades. IEEE Std 497-2002 discusses that the use of PAM instrumentation in severe accidents and for determining emergency action levels will be considered in the future. The use of PAM instrumentation, as utilized in severe

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accidents and for determining emergency action levels, is included in the proposed methodology contained in WCAP-15981. Therefore, the application of the proposed methodology contained in WCAP-15981 does not require a revision to Regulatory Guide 1.97.

Consistent with the Office of Nuclear Reactor Regulation, Office Instruction LIC-500, "Processing Request for Reviews of Topical Reports," the WOG requests that the NRC schedule a second pre-application meeting. The purpose of the meeting would be for the WOG to present and discuss the methodology and conclusions contained in WCAP-15981. The WOG requests that the NRC schedule the pre-application meeting prior to any decision that might be made with respect to accepting WCAP-15981 for review and approval.

Following NRC acceptance of WCAP-15981 for review and approval, the WOG requests that the NRC provide target dates for the issuance of any Request(s) for Additional Information and for the issuance of the Safety Evaluation (SE). The WOG requests that the NRC complete their review and issue an SE for WCAP-15981 and the associated TSTF by September 2005 in order to support the implementation of the Technical Specification changes at Beaver Valley Units 1 and 2, which is the lead plant for the WOG. First Energy Nuclear Operating Company plans to submit a License Amendment Request based on the methodology contained in WCAP-15981, following the Topical Report pre-application meeting.

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Sincerely,

Steven M. DiTommaso for

Frederick P. "Ted" Schiffley, II
Chairman, Westinghouse Owners Group

Enclosure

WOG-04-474
September 17, 2004

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Westinghouse Non-Proprietary Class 3

WCAP-15981-NP

August 2004

Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants

Westinghouse

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15981-NP

Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants

Robert J. Lutz, Jr.
Reliability and Risk Assessment

August 2004

Reviewer: James D. Andrachek
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Work Performed under Shop Order MUHP-3036

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LIST OF ACRONYMS

AC	Alternating Current
AFW	Auxiliary Feedwater
ASME	American Society of Mechanical Engineers
AOP	Abnormal Operating Procedure
ATWS	Anticipated Transient Without Scram
BAT	Boric Acid Tank
CA	Computational Aid (in SAMG)
CCW	Component Cooling Water
CDA	Core Damage Assessment
CDF	Core Damage Frequency
CET	Core Exit Thermocouples
CFR	Code of Federal Regulations
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
EFW	Emergency Feedwater (equivalent to AFW)
EOP	Emergency Operating Procedures
EPIP	Emergency Plan Implementing Procedures
ERG	Emergency Response Guidelines
ESFAS	Engineered Safety Feature Actuation System
FRG	Functional Restoration Guidelines
FV	Fussell-Vesely
HPME	High Pressure Melt Ejection
HRA	Human Reliability Analysis
IPE	Individual Plant Examination
LCD	Licensee Controlled Document
LCO	Limiting Conditions for Operation
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
NEI	Nuclear Energy Institute
NSSS	Nuclear Steam Supply System
ODCM	Offsite Dose Calculation Manual
NR	Narrow Range
NRC	Nuclear Regulatory Commission
PAM	Post Accident Monitoring
PASS	Post Accident Sampling System
PORV	Power Operated Relief Valve (refers to pressurizer)
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor

LIST OF ACRONYMS (cont.)

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
RVLIS	Reactor Vessel Level Instrumentation System
RWST	Refueling Water Storage Tank
SAMG	Severe Accident Management Guidance
SAT	Spray Additive Tank
SBO	Station Blackout
SDP	Significance Determination Process
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SLB	Steam Line Break
SSC	Systems, Structures and Components
STS	Standard Technical Specifications
SW	Service Water
TDAFW	Turbine Driven Auxiliary Feedwater
TSTF	Technical Specification Traveler
UFSAR	Updated Final Safety Analysis Report
WOG	Westinghouse Owners Group
WR	Wide Range

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 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 and which functions or actuates to mitigate a DBA or Transient the 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 a 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' 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 choose 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
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

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 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 Valves (PORV) and PORV Block Valve Position	Turbine Driven Auxiliary Feedwater (TDAFW) Pump Exhaust Radiation

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) use PRA as an input in determining the safety importance of components and systems,

  - The Maintenance Rule (10 CFR 50.65) requires the use of PRA results in determining the safety importance of components,

  - Regulatory Guide 1.174 (Reference 5) 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 relied heavily on PRA results to determine the safety importance of systems and components to measure and control post accident hydrogen in the containment.

The NRC Staff Requirements (Reference 6) regarding the acceptable PRA scope and quality to support regulatory applications has also been considered in developing the recommendation for the appropriate PAM instrumentation. Phase 1 of Reference 6 allows the use of Regulatory Guides 1.174 and 1.177 to be used. PRA quality is addressed through the PRA peer reviews, the American Society of Mechanical Engineers (ASME) PRA Standard (Reference 7), Nuclear Energy Institute (NEI) PRA Peer Review Process Guidance (Reference 8) and the draft Regulatory Guide 1.200 (Reference 9). The purpose of these efforts is to assure that the PRA is of sufficient quality to be used in regulatory applications. 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 and late fission product releases (a Level 2 PRA) and at least a qualitative assessment of seismic, fire and shutdown risks. The qualitative assessment of external events and shutdown risks will generally result in a more conservative approach to 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 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. Therefore, licensees using the methodology in this report need to provide evidence that their PRA meets the NRC Staff Requirements for Phase 1 of Reference 6. That is, the licensee's PRA should be based on the PRA scope and quality for Westinghouse NSSS plants that meet the current industry quality requirements for risk informed applications, i.e., that have addressed their PRA Peer Review findings.

3.2 METHODOLOGY

An overall methodology for using PRA results as an input to changes in the plant licensing basis is detailed in Regulatory Guide 1.174. This regulatory guide is written at a high level and additional guidance is required to develop a methodology for specific applications such as to the PAM Technical Specification. For example, a detailed methodology (Reference 10) was 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.

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 a 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 high risk significant function based on instrumentation indications? A high risk significance is defined from CDF and LERF Risk Achievement and Risk Reduction metrics per Reg. Guide 1.174. 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 risk significant indication 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.

The basis for determining risk significance is that described in Regulatory Guide 1.174. In compliance with the regulatory position in Regulatory Guide 1.174, 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 Tech Specs 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 as well as insights from mode transition (startup and shutdown transition), the Level 2 PRA containment integrity assessment, the Severe Accident Management Guidance and the Site Emergency Plan. The risk importance measures, as described in Appendix A of Regulatory Guide 1.174 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 removed from the PAM Technical Specifications and relocated 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.

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 operators 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.

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
- 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.

Other Design Basis Accidents

All of the remaining DBA analyses 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. 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. 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 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

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 PROCEDURES

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. 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. The FRGs are of particular relevance in this assessment since the FRGs also have links to EALs for implementation of offsite radiological protective actions as discussed Section 4.5.1. The response in the FRGs is classified according to the severity of the deviation of the symptom from an expected condition. The most severe classification is called a "red path," which signifies a significant deviation that requires an immediate response to attempt to bring the value of that parameter closer to the expected value. The FRG red paths, along with the instrumentation used to diagnose the red path, are:

- Subcriticality – Power Range Neutron Flux
- Core Cooling – Core Exit Thermocouples (CETs) (primary indication), Reactor Vessel Level and RCS Subcooling (secondary indication)
- Heat Sink – SG Wide Range Level (primary indication), Total Feedwater Flow (secondary indication)
- Integrity – RCS Wide Range Pressure, RCS Cold Leg Resistance Temperature Detectors (RTDs)
- Containment – Containment Wide Range Pressure

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 PRA and EAL discussions (Sections 4.5.1 and 4.5.2). Separate consideration of the impact of instrumentation in the EOPs is redundant to the assessment of the importance of instrumentation in the PRA and EALs, and is therefore not necessary.

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.

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 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 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.

- 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 OCDM 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.

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. 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							
Component Cooling Water Flow Rate		✓					
Condenser Air Ejector (High Range)							✓
Containment Enclosure Negative Pressure							
Containment Sump Level (Narrow Range)							
Containment Pressure (Narrow Range)							
Containment Water Level (Wide 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 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					✓

Table 8 (cont.) Summary of Important Indications for Accident Management

Indication/Purpose	DBA	EOP	SAMG	PRA	E-Plan
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.

The assessment described in this section of the report focuses on that instrumentation that is relied upon in plant safety analyses, accident management and offsite protective actions. 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 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 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. 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. 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. It is used in the SAMG to assure that adequate water is available in the containment sump(s) for ECC recirculation, should that 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. CST refill is a long-term action that is not credited in the UFSAR analyses. In the PRA models, operator action 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 a low risk significance. Additionally, CST refill is typically not required in the first 16 to 20 hours after an accident. 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 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 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 heart 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 Water Level (Wide Range)

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 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-14986-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. 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. 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 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 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.

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 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 Level (Narrow Range)	✓		
RWST Level (Wide Range)	✓		✓
Steam Generator Water Level (Narrow Range)	✓		
Spent Fuel Pool Exhaust Radiation	✓		
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	✓		
Safety Valve Position	✓		

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 Tech Spec Instrumentation

Function	Typical Reg. Guide 1.97 Variable Type/Category	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.
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 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: Only the highest Reg. Guide 1.97 classification is shown in this table			

Table 11 Regulatory Guide 1.97 Classification for PAM Relocated to LCD			
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 used 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. These alternate indications can also provide the information necessary for the operators to determine the need to initiate a manual reactor trip.

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.

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

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 (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 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 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, 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 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.

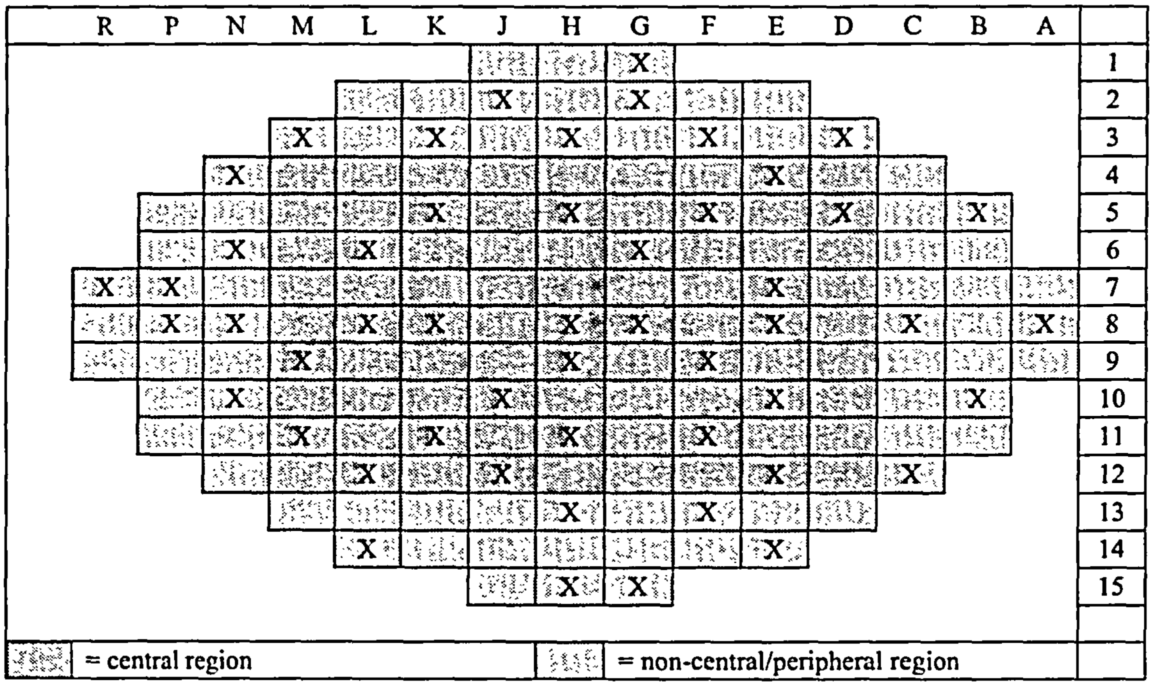

Figure 1 Typical Core Exit Thermocouple Locations for a Three Loop Plant

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. 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.

8 IMPLEMENTATION

The plant specific implementation of this methodology only requires a confirmation of the generic evaluations contained in this report based on 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.

For the PRA portion of the assessment, the licensee should ensure that the PRA Peer Review findings have been addressed. In addition, the generic assessment PRA assessment presented in Appendix A of this report indicates that only a detailed quantitative PRA for internal initiating events from the at-power condition is required to support the plant specific PAM Technical Specification assessments. The assessment presented in Appendix A for the other important initiating events (e.g., fire, seismic) indicates that only a qualitative PRA (e.g., FIVE and SMA) is adequate to identify any unique operator actions that could impact the PAM Technical Specification instrument determination.

Also, the generic determination in this report for the Core Damage Assessment determination was based on the use of the approved methodology in WCAP-14696. If a licensee has used a different methodology then an assessment of the key indications to support the core damage assessment should be performed based on the actual methodology used.

9 REFERENCES

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11. "Westinghouse Owners Group Emergency Response Guidelines," Revision 1C, 1997.

12. "Westinghouse Owners Group Severe Accident Management Guidance," Revision 0, June 1994.

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15. "Requirements for Emergency Response Capability," NUREG-0737 Supplement 1, December 1982.

16. "Westinghouse Owners Group Post Accident Core Damage Assessment Guidance," WCAP-14696-A, Revision 1, December 1999.

17. 10 CFR 50.44, "Combustible Gas Control in Containment," Federal Register, September 16, 2003, Volume 68, Number 179, pp. 54123 – 54142.

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 (FV) measure is a derivation of the RRW and is defined as, FV=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 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 FV) 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 FV values for operator actions modeled in the PRA. However, the PRA information collected in the 2002 survey only included FV 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 FV importance measure results from information provided in the 2002 database update. Table A-2 provides the FV 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 FV values in the database.

A.4 CRITERIA FOR RISK IMPORTANCE

From Regulatory Guide 1.174 and the draft Regulatory Guide 1.201 for the 10 CFR 50.69 rulemaking, it can be concluded that a component has a high risk significance if the FV value is greater than 1.05 or the RAW value is greater than 2.0. 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 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 FV 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 (Using FV Importance Measure)

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

Table A-2 1999 PRA Survey Results Operator Action Risk Reduction Worth (Using FV Importance Measure)			
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			

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 the FV 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 FV values in the 1999 survey also had high FV values in the 2002 survey. The same conclusion can be drawn with respect to the low

FV values; those operator actions with low FV values in the 1999 survey also had low FV 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 a.c. 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 a.c. power to the vital bus(es). An indication of successful restoration of a.c. 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 a.c. bus is the bus voltage. The vital bus voltage requirements are addressed by the Distribution Systems Technical Specification and are not PAM instrumentation.

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 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 d.c. 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 a.c. and all emergency d.c. power buses. For sequences involving the failure of emergency a.c. power, the important operator actions are the same as for internal initiating events involving the loss of a.c. power. For the loss of d.c. 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 a.c. and d.c. power, or a station blackout. In the case of a loss of all emergency d.c. 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 a.c. 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
TECHNICAL SPECIFICATIONS TASK FORCE TRAVELER

(To be provided separately)