BWR OWNERS' GROUP

Douglas W. Coleman
BWROG Chairman
Tel: (509) 377-4342
Fax: (509) 377-2354
dwcoleman@energy-northwest.com

c/o Energy Northwest, Mail Drop PE04, P.O. Box 968, Richland, WA 99352-0968

Project Number 691

BWROG-08070
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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

**SUBJECT:** Responses to Requests for Additional Information (RAIs) Regarding the Submittal of BWROG Licensing Topical Report (LTR) NEDO-33349, Revision 1, "BWR Application to Regulatory Guide 1.97, Revision 4," (TAC No. MD6697)

**REFERENCES:**
1. BWR Owners' Group Letter from R.C. Bunt dated August 31, 2007 Containing Licensing Topical Report NEDO-33349
2. Letter dated August 4, 2008 from NRC Containing Request For Additional Information
3. Letter dated August 19, 2008 from NRC Containing Request For Additional Information

Please find enclosed the BWROG response to the NRC Requests for Additional Information on the subject Licensing Topical Report NEDO-33349, Revision 1 (submitted via Reference 1). NRC provided the RAIs for this report via references 2 and 3. The enclosure contains three (3) attachments.

Attachment 1 contains RAI responses which are organized into four (4) sections. Sections 1 and 2 provide responses to Instrumentation & Controls Branch RAIs; section 3 contains responses to Nuclear Security and Incident Response Branch RAIs; and section 4 provides responses to Reactor Systems Branch RAIs.

Additionally, attachment 1 contains two (2) sections (Sections 5 and 6) dedicated to specific issues. Section 5 is related to the reconciliation of Type B and Type C Variables with respect to Regulatory Guide 1.97 Revision 3 design requirements; and section 6 is related to the declaration of Primary Containment Isolation Valve Position Indication as a Type D Variable.

Attachment 2 is a NEDO-33349 mark-up describing proposed changes to the LTR based on BWROG responses to the respective RAIs. Upon receipt of a Safety Evaluation (SE) issued by the Commission, NEDO-33349 will be revised and released as an approved LTR.

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Attachment 3 is a cross-reference table which links the BWROG RAI responses to the proposed changes contained within Attachment 2 (NEDO-33349 mark-up).

A substantial portion of the BWROG membership has endorsed this letter. However, majority endorsement should not be interpreted as a commitment of any individual member to a specific course of action.

We look forward to your timely review of these responses, and would be happy to meet with you to discuss any remaining issues. Should you have additional questions regarding this submittal, please contact Michael Iannantuono (BWROG - Project Manager) at 910-819-1956.

Sincerely,

Douglas W. Coleman

Douglas W. Coleman, Chairman
BWR Owners' Group

Enclosure
Attachments: 1. BWR Owners' Group Response to NRC Requests for Additional Information
2. BWR Owners' Group Proposed Revision to NEDO-33349
3. RAI Summary and NEDO-33349 Mark-up Changes

cc: M.C. Honcharik, NRC
F.P. Schiffley, BWROG Vice Chairman
K.A. McCall, BWROG Program Manager
BWROG Primary Representatives
A. Klemptner, DTE
M.A. Iannantuono, GEH

## IMPORTANT NOTICE REGARDING THE CONTENTS OF THIS REPORT

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**ATTACHMENT 1**
**BWR Owners' Group Response to NRC Requests for Additional Information**

***Introductory Response***

Licensing Topical Report (LTR) NEDO-33349 - Revision 1, was prepared at the direction of the BWR Owners' Group (BWROG) to identify a methodology for adaptation to Regulatory Guide 1.97 Revision 4 for BWR accident monitoring instrumentation. The LTR includes recommendations for alignment with RG 1.97 Revision 4 positions, and to support equipment modifications at operating BWRs.

This RAI response document has been organized into six sections. Sections 1 and 2 contain questions and responses to RAI from Instrumentation and Control Branch (Lead Reviewer); Section 3 from Nuclear Security and Incident Response Branch; Section 4 from Reactor Systems Branch; Section 5 contains supporting information for the reconciliation of Type B and Type C Variables with respect to Regulatory Guide 1.97 Revision 3 design requirements; and Section 6 contains supporting information related to the declaration of Primary Containment Isolation Valve Position Indication as a Type D Variable.

The document format contains a restatement of the NRC RAI followed by the BWROG response.

## Section 1: Instrumentation and Controls Branch RAI

***RAI 1-I&C-1***

On Page 2-1, Section 2.2 includes Level Control as a Type B function instead of the Type B Core Cooling Function in Revision 3 of RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants." Discuss the differences between the RG 1.97 Revision 3 Type B, Core Cooling Function and LTR NEDO-33349 Type B, Level Control function.

***RAI 1-I&C-1 Response***

Reactor Water Level is the same Type B variable as Coolant Level in the RPV for monitoring the core cooling function. The terminology difference is due to naming conventions associated with BWR Emergency Procedure Guidelines (EPG).

The core cooling function relates to fuel cladding barrier integrity. The fuel cladding remains intact when water level in the RPV is maintained above a predetermined level. Coolant Level in Reactor Vessel and BWR Core Thermocouples (an approved deviation per BTP 7-10), are listed in RG 1.97 R3 as Type B variables for Core Cooling function.

RG 1.97 R4 (IEEE Standard 497-2002) provides design criteria for the accident monitoring instrumentation. Type B variable instrumentation is a source of the primary information to operators for assessing plant critical safety functions (CSF) in response to Anticipated Operational Occurrences (AOO), accidents, or to achieve a safe shutdown. Consistent with this approach, appropriate BWR related Critical Safety Functions (CSF) were identified in the Topical as Reactivity Control, Pressure Control, Level Control, and Primary Containment Control.

The corresponding Type B variables determined from EPG entry conditions are Reactor Power/Neutron Flux, Reactor Pressure, Reactor Water Level, and Suppression Pool Temperature, Suppression Pool Water Level and Drywell Pressure Control.

***RAI 1-I&C-2***

On Page 2-4, Section 2.6 should discuss the qualification and design of Type E instrumentation that monitors radiological releases, in terms of being qualified to operate in the environment present when the instrument would be called upon to operate.

***RAI 1-I&C-2 Response***

Section 2.6 of the LTR will be revised to indicate the following:

Type E variables are used for monitoring and assessment of release magnitude, environmental monitoring, and radiation level monitoring in plant environs. Operational specifications of the instrumentation (range and ruggedness) will be consistent with the associated service environment. Type E variable qualification and design requirements are defined in IEEE Std. 497, which states, "Instrument channels that monitor systems are not required to be environmentally or seismically qualified."

Type E instruments should be of high-quality commercial grade, and should be selected to withstand the specified service environment. This is consistent with provisions of RG 1.97 R2 & R3 for Category 3 variables.

Instrumentation related to Type E variables, will be used by station operators to evaluate mitigation strategies associated with Emergency Planning (EP).

***RAI 1-I&C-3***

On Page 3-1, Section 3.1, second paragraph, second sentence, the phrase "but total conversion may be considered as part of plant control room upgrades including the use of digital systems," may lead to a conclusion that total conversion would only be part of a control room upgrade. This conclusion would be incorrect. A plant could convert to RG 1.97 Revision 4 without a control room upgrade. Section 3.1 should be revised to allow conversions to RG 1.97 R4 without a control room upgrade.

***RAI 1-I&C-3 Response***

LTR Section 3.1 will be revised to remove references to control room upgrades, and be replaced with non-specific language to describe modifications and upgrades.

***RAI 1-I&C-4***

On Page 3-2, Section 3.6 should discuss the applicability of RGs that are referenced by RG 1.97 Revision 4 that also reference industry codes and standards.

***RAI 1-I&C-4 Response***

LTR Section 3.6 will be revised to reference applicable RGs.

***RAI 1-I&C-5***

On Page 4-4, Section 4.1.3 should discuss the impact of common cause failures of digital systems.

***RAI 1-I&C-5 Response***

Section 4.1.3 will be revised to indicate the following:

Section 6.2 of IEEE Std. 497 addresses common cause failures related to microprocessor-based instrumentation software for Type A, B, and C variables.

10CFR50 single failure criteria guidance is provided in IEEE Std. 379, "Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Class 1E Systems," as endorsed by the NRC in RG 1.53 R2.

Requirements for consideration of common cause failures in a single failure analysis are contained in Section 5.5 of IEEE Std. 379-1988. Design qualification and quality assurance programs are intended to afford protection from design deficiencies and manufacturing errors. This approach is also appropriate for potential common cause failures associated with computer hardware and software that have been developed under the requirements of IEEE Std. 7-4.3.2, "Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations."

***RAI 1-I&C-6***

On Pages 4-6 thru 4-17, Sections 4.1.5 thru 4.5.2 call for variables that may not be the most direct variable for monitoring each function. Discuss the selection of non-direct variables in relation to the IEEE-497, Clause 6.9 statement, "To the extent practical, a direct variable shall be selected to monitor the related function. A less direct variable may be substituted for the most direct variable if justified by analysis. The analysis shall account for misinterpretation of the less direct variable as well as availability of reliable instrumentation, by direct variables."

***RAI 1-I&C-6 Response***

Section 4 will be revised to indicate the following:

Section 6.9 of IEEE 497, provides guidance on use of a less direct variable which may be substituted for the most direct variable if supported by analysis.

The LTR evaluation chooses the most direct variable for monitoring each Type variable. Consistent with RG 1.97 Rev 4, the typical BWR/4 and BWR/6 safety analyses are used to determine a generic list of Type A variables. BWR EPGs are used to determine the generic list of Type B variables. Type C variables are based on safety analysis, EPGs, and engineering analysis. A typical BWR/4 and BWR/6 were used to establish Type D and Type E variables.

The LTR requires plant specific evaluations to be performed in cases where the most direct variable is not available, or has been substituted by an alternate variable. For example, a plant specific evaluation would be performed with respect to a Type D variable substitution, if direct system flow instrumentation was substituted with pump valve position indication and/or indication of pump energization. In this case, the plant would rely on a specific analysis to determine if the choice of less direct instrumentation provides sufficient information to the operator monitoring system performance and required functions.

***RAI 1-I&C-7***

On Page 4-9, Section 4.2.3 should provide greater detail for the justification for how Suppression Pool Temperature fulfills the NEDO-33349 Type B key variable for the Primary Containment Control function.

***RAI 1-I&C-7 Response***

RAI Response Section 5 provides information regarding reconciliation of RG 1.97 R3 to the results of the LTR. Section 4 of the LTR addresses the methodology and the results of the BWR analysis.

Section 4.2.3 reflects how RG 1.97 R4 (IEEE Std. 497-2002) was applied in the LTR, to determine BWR/4 & /6 Type B variables – variables that provide information to plant operators assessing critical safety function execution. BWR Type B variables are related to EPG entry conditions which support plant mitigation strategies related to Primary Containment Control functions.

The Suppression Pool Temperature parameter is a Type B variable / entry condition for the Primary Containment Control function. It is monitored and controlled below the Technical Specification limits for protection of equipment in the primary containment, and to ensure containment integrity for the duration of an accident. If Suppression Pool Temperature exceeds the value for Emergency Operating Procedure (EOP) entry, operators initiate mitigating action to restore the suppression pool temperature parameter.

***RAI 1-I&C-8***

On Page 4-9, Section 4.2.3 should provide greater detail for the justification for how Suppression Pool Water Level fulfills the NEDO-33349 Type B key variable for the Primary Containment Control function.

***RAI 1-I&C-8 Response***

RAI Response Section 5 provides information regarding reconciliation of RG 1.97 R3 to the results of the LTR. Section 4 of the LTR addresses the methodology and the results of the BWR analysis.

Section 4.2.3 reflects how RG 1.97 R4 (IEEE Std. 497-2002) was applied in the LTR, to determine BWR/4 & /6 Type B variables – variables that provide information to plant operators assessing critical safety function execution. BWR Type B variables are related to EPG entry conditions which support plant mitigation strategies related to Primary Containment Control functions.

Suppression Pool Water Level monitoring is required to provide information to the operator, independent of actions associated with the Primary Containment Control function. It is monitored and controlled within Technical Specification limits to ensure adequate quench volume to absorb heat capacity associated with Emergency Depressurization (low level); provide RPV makeup via Emergency Core Cooling Systems (ECCS); and to minimize hydraulic-mechanical loading of equipment and structures located in primary containment and/or the suppression chamber (high level). If Suppression Pool Water Level exceeds values (high or low) which direct EPG entry, operators will initiate action to restore the suppression pool water level parameter.

***RAI 1-I&C-9***

On Page 4-9, Section 4.2.3 should provide greater detail for the justification for Primary Containment Isolation Valve Position no longer being a Type B key variable for the RG 1.97 Revision 3 Maintaining Containment Integrity function.

***RAI 1-I&C-9 Response***

LTR Section 4.2.3 reflects the conclusion that primary containment isolation valve (PCIV) position indication is a Type D variable for typical BWR/4 & /6 designs. RG 1.97 Revision 3 listed PCIV position indication as a Type B variable under the Maintaining Containment Integrity Function.

Type B variables provide information to plant operators assessing critical safety function execution. The BWR Type B variables have been selected from BWR EPG entry conditions identified in four top-level guidelines, including the Primary Containment Control Function.

The Type B Primary Containment Control function variables are Primary Containment Pressure, Suppression Pool Water Level and Suppression Pool Temperature. PCIV position indication is not a Critical Safety Function as it is not relied upon in safety analysis or in BWR EPGs. PCIV position indication meets the criteria of a Type D variable, as it provides verification that the containment isolation valve safety system has performed its function.

Further justification is included in RAI Response Section 6; and NEDO-33349 Section 7.4.1.

***RAI 1-I&C-10***

On Page 4-9, Section 4.2.3 should provide greater detail for the justification for the use of Drywell Pressure instead of Primary Containment Pressure as a Type B key variable for the Containment Control function.

***RAI 1-I&C-10 Response***

RAI Response Section 5 provides information regarding reconciliation of RG 1.97 R3 to the results of the LTR. Section 4 of the LTR addresses the methodology and the results of the BWR analysis.

Drywell Pressure and Primary Containment Pressure are essentially the same instruments for BWR containment designs, with both terms used interchangeably. Drywell Pressure indication is used as the EPG entry condition which supports plant mitigation strategies related to Primary Containment Control functions.

***RAI 1-I&C-11***

On Page 4-9, Section 4.2.3 should describe the difference between the RG 1.97 Revision 3 Type B, Maintaining RCS Integrity function and the NEDO-33349 Type B, Pressure Control function.

***RAI 1-I&C-11 Response***

RAI Response Section 5 provides information regarding reconciliation of RG 1.97 R3 to the results of the LTR. Section 4 of the LTR addresses methodology and results of the BWR analysis.

Section 4.2.3 reflects how RG 1.97 R4 (IEEE Std. 497-2002) was applied in the LTR, to determine BWR/4 & /6 Type B variables – variables that provide information to plant operators assessing critical safety function execution. BWR Type B variables are related to EPG entry conditions which support plant mitigation strategies related to RPV Control functions.

The Reactor Coolant Pressure Boundary provides a barrier to the release of primary coolant to the Primary Containment. The EPG for RPV Control uses pressure, level and power (neutron monitoring) control in an integrated manner to fulfill the RCS Integrity function. Specifically, the Reactor Pressure Control function is accomplished by controlling the RPV pressure through operation of Safety Relief Valves (SRV).

RPV pressure, drywell pressure, and drywell sump level, are variables selected in RG 1.97 R3 to fulfill the Maintaining Reactor Coolant System Integrity function. For BWRs, the Drywell Drain Sump is isolated during an accident, preventing the transfer of the Drywell Drain Sump inventory. Consequently, leakage trending analysis cannot be performed by operators under accident conditions. Drywell Sump Level is only effectively monitored during normal plant operations. Drywell Pressure is identified as a Type B variable in RG 1.97 R3 and is listed as a Type B variable in the LTR using provisions of RG 1.97 R4 (IEEE Std. 497-2002) under Primary Containment Control.

Monitoring RPV Pressure instrumentation satisfies requirements for both functions - Maintaining RCS Integrity function (RG 1.97 R3) and Reactor Pressure Control function (NEDO-33349 Type B variables).

***RAI 1-I&C-12***

On Page 4-9, Section 4.2.3 should provide greater detail for the justification of Drywell Pressure no longer being a Type B key variable for the RG 1.97 Revision 3 Maintaining RCS Integrity function or the NEDO-33349, Type B, Pressure Control function.

***RAI 1-I&C-12 Response***

RAI Response Section 5 provides information regarding reconciliation of RG 1.97 R3 to the results of the LTR. Section 4 of the LTR addresses methodology and results of the BWR analysis.

Section 4.2.3 identifies the Reactor Coolant System (RCS) Pressure Boundary as providing a barrier for the release of reactor coolant to the Primary Containment. The LTR lists Drywell Pressure as a Type A, B and C variable for multiple reasons. The Reactor Control EPGs include RPV Pressure monitoring for the manual control of Safety Relief Valves (SRV) to support RCS integrity.

Drywell Pressure is included as a Type B variable for Primary Containment control and also is an entry condition for the RPV Control EPG.

***RAI 1-I&C-13***

On Page 4-11, Section 4.3.4 should provide greater detail for the justification for how Reactor Water Level fulfills the Type C key variable for the Fuel Cladding function.

***RAI 1-I&C-13 Response***

RAI Response Section 5 provides information regarding reconciliation of RG 1.97 R3 to the results of the LTR. Section 4 of the LTR addresses methodology and results of the BWR analysis.

For BWRs, the parameter providing the most direct indication of fission product barrier integrity is RPV water level indication. Section 4.3.4 concludes that Reactor Water Level is the primary Type C variable associated with the Fuel Cladding function. The Fuel Cladding function of RG 1.97 R3, lists Type C variables which were later determined not to be required for BWRs (radioactivity concentration, analysis of primary coolant, and BWR core thermocouples). The LTR methodology included an engineering evaluation with safety analysis insights and EPG-based conclusions that RPV water level is the most-direct parameter reflective of fuel cladding integrity.

The integrity of the fuel cladding barrier is determined by the status of core cooling. The fuel cladding barrier is maintained intact when the core remains adequately cooled. Breach of the fuel cladding barrier is assumed when adequate core cooling cannot be restored or has not been maintained. RPV water level instrumentation is the most directly indicative parameter in determining adequate core cooling effectiveness.

Loss of RPV coolant inventory during an accident results in a corresponding decrease of RPV water level, and an increase in fuel temperature. If actions to restore RPV water level are not effective, rising fuel temperature can cause fuel cladding damage and release of radioactivity. The operator is guided by EPG/SAG-based procedures to consider breach of the fuel cladding barrier, and to initiate mitigating action to protect the fuel cladding barrier.

Other RG 1.97 R4 Type E radiation detection instruments (off-gas radiation monitoring, containment radiation monitoring, and RPV sampling) and hydrogen monitors, provide necessary information to the operator, which enables the assessment of potential core damage. Additional variables, not related to EPG assessments of fission product barrier integrity, are used during Emergency Planning (EP).

***RAI 1-I&C-14***

On Page 4-11, Section 4.3.4 should provide greater detail for the justification for how Reactor Water Level fulfills Type C key variable for the Reactor Coolant Pressure Boundary function.

***RAI 1-I&C-14 Response***

RAI Response Section 5 provides information regarding reconciliation of RG 1.97 R3 to the results of the LTR. Section 4 of the LTR addresses methodology and results of the BWR analysis.

Section 4.3.4 concludes, Reactor Water Level to be the Type C variable most-directly associated with the Reactor Coolant Pressure Boundary and Fuel Cladding functions.

Integrated RPV Control EPGs, consider the RPV Water Level parameter in addressing several critical safety functions. RPV Water Level is a key parameter controlled through several EPG strategies. As required, RPV Water Level is restored and maintained within acceptable limits using EPG-based Emergency Operating Procedures (EOP).

***RAI 1-I&C-15***

On Page 4-11, Section 4.3.4 should provide greater detail for the justification for how Suppression Pool Temperature fulfills the Type C key variable for the Reactor Coolant Pressure Boundary function.

***RAI 1-I&C-15 Response***

RAI Response Section 5 provides information regarding reconciliation of RG 1.97 R3 to the results of the LTR. Section 4 of the LTR addresses methodology and results of the BWR analysis.

Section 4.3.4 concludes, Suppression Pool Temperature to be the Type C variable most-directly associated with the Reactor Coolant Pressure Boundary function.

BWR design relates the Suppression Pool Temperature parameter, to Reactor Coolant Pressure Boundary integrity (also used in the BWR EPGs for Primary Containment control). Significant Suppression Pool Temperature increases are indicative of SRV leakage / operation, or of Reactor Coolant Pressure Boundary integrity loss. BWR Technical Specifications impose operational limits to control Suppression Pool Temperature. Beyond Tech Spec values, EPGs provide strategies to avoid approaching containment design limitations.

***RAI 1-I&C-16***

On Page 4-11, Section 4.3.4 should provide greater detail for the justification for Drywell Drain Sump Level no longer being a Type C key variable for the Reactor Coolant Pressure Boundary function.

***RAI 1-I&C-16 Response***

RAI Response Section 5 provides information regarding reconciliation of RG 1.97 R3 to the results of the LTR. Section 4 of the LTR addresses methodology and results of the BWR analysis.

The LTR concludes Drywell Drain Sump Level is not a Type C key variable for the Reactor Coolant Pressure Boundary function.

For BWRs, Drywell Drain Sump Level is used during normal operation to assess leakage to the Primary Containment as provided in RG 1.45. Trending Drywell Drain Sump Level variations allows operators to calculate a leakage rate from the Reactor Coolant Pressure Boundary into the Primary Containment. If defined values are exceeded, operator actions are required to mitigate the breach of the Reactor Coolant Pressure Boundary.

The Drywell Drain Sump Level neither automatically initiates nor alerts the operator to initiate a safety system as a result of an accident to ensure RCS integrity. The Drywell Drain Sump is isolated during an accident, preventing the transfer of the Drywell Drain Sump inventory. Consequently, leakage trending analysis cannot be performed by operators under these circumstances.

This assessment is germane to guidance provided in Branch Technical Position 7-10, Revision 5 – March 2007, Table 1, For BWRs: Acceptable Deviations and Clarifications to Revisions 2 and 3 of Regulatory Guide 1.97. Additional conditions were imposed, with implementation executed by most BWRs in the form of plant specific deviations.

***RAI 1-I&C-17***

On Page 4-11, Section 4.3.4 should provide greater detail for the justification for RCS Pressure no longer being a Type C key variable for the Primary Containment function.

***RAI 1-I&C-17 Response***

RAI Response Section 5 provides information regarding reconciliation of RG 1.97 R3 to the results of the LTR. Section 4 of the LTR addresses methodology and results of the BWR analysis.

Section 4.3.4 concludes that RCS Pressure is not a Type C variable associated with the Primary Containment function, while RG 1.97 R3 lists RCS Pressure as a Type C variable for the Primary Containment function.

For BWRs, the LTR concludes that RCS pressure is not a direct or indirect indicator of a breach in the Primary Containment Integrity barrier. Also included in the LTR, is a list of Type C variables aligned with RG 1.97 R4 related to Primary Containment Integrity, Drywell Pressure, Suppression Pool Level, and Suppression Pool Temperature.

RCS Pressure is included as a Type C variable supporting the Reactor Coolant System Integrity function, and as a Type A and B variable supporting the RCS Integrity function.

***RAI 1-I&C-18***

On Page 4-11, Section 4.3.4 should provide greater detail for the justification for the use of Drywell Pressure instead of Primary Containment Pressure as a Type C key variable for the Primary Containment function.

***RAI 1-I&C-18 Response***

RAI Response Section 5 provides information regarding reconciliation of RG 1.97 R3 to the results of the LTR. Section 4 of the LTR addresses the methodology and the results of the BWR analysis.

Drywell Pressure and Primary Containment Pressure are essentially the same instruments for BWR containment designs, with both terms used interchangeably. Drywell Pressure indication is used as the EPG entry condition which supports plant mitigation strategies related to Primary Containment Control functions. Drywell pressure is used for determination of Primary Containment integrity.

***RAI 1-I&C-19***

On Page 4-11, Section 4.3.4 should provide greater detail for the justification for Suppression Pool Water Level as a Type C key variable for the Primary Containment function.

***RAI 1-I&C-19 Response***

RAI Response Section 5 provides information regarding reconciliation of RG 1.97 R3 to the results of the LTR. Section 4 of the LTR addresses methodology and results of the BWR analysis.

Section 4.3.4 concludes that Suppression Pool Water Level indication is a Type C variable in accordance with RG 1.97 R4 (based on provisions of IEEE Standard 497).

The Suppression Pool Water Level parameter, is an EPG entry condition for the Primary Containment Control function, and is monitored and controlled below the Technical Specification limits for protection of equipment in the primary containment. Parameter monitoring and control is used to maintain containment integrity for the duration of an accident. If Suppression Pool Water Level exceeds values (high or low) which direct EPG entry, the operator will initiate action to restore the suppression pool water level parameter.

***RAI 1-I&C-20***

On Page 4-11, Section 4.3.4 should provide greater detail for the justification for Suppression Pool Temperature as a Type C key variable for the Primary Containment function.

***RAI 1-I&C-20 Response***

RAI Response Section 5 provides information regarding reconciliation of RG 1.97 R3 to the results of the LTR. Section 4 of the LTR addresses methodology and results of the BWR analysis.

Section 4.3.4 concludes that Suppression Pool Temperature indication is a Type C variable in accordance with RG 1.97 R4 (based on IEEE Standard 497).

Suppression Pool Temperature monitoring is provided to detect conditions leading to containment breach; and to verify effectiveness of ECCS action to prevent containment breach. Maintaining Suppression Pool Temperature within limits is a Technical Specifications requirement and an EPG action (heat capacity limits) to ensure the containment pressure suppression function is maintained during accident conditions.

If Suppression Pool Temperature exceeds limits, EPG-based procedures are employed to manage and maintain Primary Containment integrity.

***RAI 1-I&C-21***

On Page 4-15, Section 4.4.5 should include statements concerning the EQ of position switches for containment isolation valves located inside and outside of containment in response to a pipe break outside of containment.

***RAI 1-I&C-21 Response***

Section 4.4.5 summarizes the EQ requirements for containment isolation valve position switches located inside and outside containment. Included in Section 4.4.5 is the need to consider pipe breaks outside containment.

Section 4.4.5 includes statements that conclude containment isolation valves are only required to provide isolation of the containment for a LOCA. Containment isolation valve position switches are not required to be qualified for pipe breaks outside containment because the containment is isolated before fuel uncovery occurs.

***RAI 1-I&C-22***

On Page 5-1, Section 5, add to the last sentence of the discussion on SQ the words, "following a seismic event."

***RAI 1-I&C-22 Response***

LTR Section 5 will be revised to include the phrase "...following a seismic event."

***RAI 1-I&C-23***

On Page 5-1, Section 5, the last sentence alludes to the concept that the NEDO-33349 methodology could be implemented consistent with the provisions of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.59. This section should address the fact that a plant-specific conversion to RG 1.97 Revision 4 would be a change in a licensee's commitment and would need to be reviewed by the NRC staff. Additionally, any plant-specific deviations from either NEDO-33349 or RG 1.97 Revision 4 would need to be submitted along with detailed justifications for those deviations for NRC staff review.

***RAI 1-I&C-23 Response***

LTR Section 5 (last paragraph) is to be revised to delete the reference to 50.59, and to address the implementation concerns raised by the NRC staff.

Section 5 of the LTR will be revised to indicate the following:

Tables are generic in nature and are intended for illustration purposes only. A plant specific review – consistent with Section 4 evaluation methodology – is required when implementing changes which impact accident monitoring capabilities. Additionally, significant plant modifications, and subsequent impact to the station's accident monitoring program, may require NRC approval prior to implementation.

***RAI 1-I&C-24***

Section 5 should address the impact of NUREG-0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements," on items that are included in both NUREG-0737 and RG 1.97.

***RAI 1-I&C-24 Response***

Section 5 of the LTR will be revised from what is described in the RAI 1-I&C-23 response to reflect the following related to items included in both NUREG 0737 and RG 1.97:

NRC acceptance of this LTR methodology will be used as a basis for plant specific reviews of their post accident monitoring requirements and licensing commitments including but not limited to items included in both NUREG 0737 and RG 1.97 Revision 2 or 3.

In addition to the above, it is expected that the NRC will include conditional provisions in the NRC's acceptance of this LTR to address implementation of LTR methodology similar to what has been done before on other BWR LTRs, such as NEDO-33160-A "Regulatory Relaxation for the Post- Accident SRV Position Indication System" (Reference 13 in LTR).

***RAI 1-I&C-25***

On Pages 5-1 and 5-7, Tables 5-1 and 5-2 should explain the concept of a generic list of Type A variables in consideration of the definition of Type A variables that indicates that Type A variables are plant specific.

***RAI 1-I&C-25 Response***

The LTR uses a methodology which enables development of a generic Type A variables list based on BWR safety analysis. Although most BWRs possess similar methods in determining Type A variables, minor differences exist. The LTR offers guidance in performing necessary plant specific analyses, in order to expand or contract a station's list of Type A variables.

For example, if a station desired to remove Drywell Temperature as a Type A variable, the LTR would be supportive in concluding Drywell Temperature is not a Type A variable.

Section 7 of the LTR will be revised to indicate the following to include language addressing changes for Type A variables, as compared to the approach described in RG 1.97 R2 & R3:

Section 4 of the LTR describes the methodology to establish Type A variables in compliance with provisions of RG 1.97 R4 for generic BWRs. The methodology defines a generic list of Type A variables based on accident analysis supported by the EPGs. The generic list requires review and approval on a plant specific basis, and should additionally reflect Type A variables for most BWRs. BWRs did not provide a generic list of Type A variables for NRC review – compliance with RG 1.97 R3 was categorized through plant specific evaluations.

***RAI 1-I&C-26***

On Pages 5-1 and 5-7, Tables 5-1 and 5-2 should provide greater detail for the justification for Suppression Pool Water Level no longer being a Type D key variable for the status of Containment Related Systems.

***RAI 1-I&C-26 Response***

LTR Section 7 of the LTR will be revised to include the following related to Type D variables:

The LTR concludes that Suppression Pool Water Level is a Type A, B, and C variable. Type D has a broad definition encompassing all Type A, B, and C variables. Type D imposes reduced design requirements, bounded by the requirements for Type A, B, and C variables. Suppression Pool Water Level has not been listed as a Type D variable to ensure the highest design requirements are applied.

***RAI 1-I&C-27***

On Pages 5-1 and 5-7, Tables 5-1 and 5-2 should provide greater detail for the justification for Suppression Pool Temperature no longer being a Type D key variable for the status of Containment Related Systems.

***RAI 1-I&C-27 Response***

LTR Section 7 of the LTR will be revised to include the following related to Type D variables:

The LTR concludes that Suppression Pool Temperature is a Type A, B, and C variable. Type D has a broad definition encompassing all Type A, B, and C variables. Type D imposes reduced design requirements, bounded by the requirements for Type A, B, and C variables. Suppression Pool Temperature has not been listed as a Type D variable to ensure the highest design requirements are applied.

***RAI 1-I&C-28***

On Pages 5.1 and 5.7, Tables 5-1 and 5-2 should provide greater detail for the justification of Neutron Flux becoming a Type D variable to monitor Safety System Performance for the Reactor Protection System and the Control Rod Drive System.

***RAI 1-I&C-28 Response***

LTR Section 7 of the LTR will be revised to provide detail on Neutron Flux (Neutron monitoring)

LTR Tables 5-1 and 5-2, document variable determination results. The Tables indicate that Neutron Monitoring is a Type B variable supporting the Reactivity Control function, and a Type D variable for the Reactor Protection System and Control Rod Drive System.

RG 1.97 Revisions 2 and 3 include Neutron Monitoring as a Type B variable. Category 1 requirements were imposed which BWR design did not fully meet. NEDO-31558-A "BWROG Proposed Neutron Monitoring System Post-Accident Monitoring Functional Criteria," established alternate design criteria for existing BWRs.

The LTR concludes that Neutron Monitoring is a Type B variable and that the requirements imposed in NEDO-31558-A are appropriate in establishing design requirements for RG 1.97 Revision 4. Additionally, the LTR concludes that Neutron Monitoring meets the criteria as a Type D variable, used to monitor performance of the Reactor Protection and Control Rod Drive Systems using NEDO-31558-A design requirements.

***RAI 1-I&C-29***

On Pages 5-1 and 5-7, Tables 5-1 and 5-2 should provide greater detail for the justification for Drywell Pressure no longer being a Type D key variable for the status of Containment Related Systems.

***RAI 1-I&C-29 Response***

LTR Section 7 will be revised to include the following related to Type D variables:

The LTR concludes that Drywell Pressure is a Type A, B, and C variable, and can also be considered as Type D. Type D has a broad definition encompassing all Type A, B, and C variables. Type D imposes reduced design requirements, bounded by the requirements for Type A, B, and C variables. Drywell Pressure has not been listed as a Type D variable to ensure the highest design requirements are applied.

***RAI 1-I&C-30***

On Pages 5-1 and 5-8, Tables 5-1 and 5-2 should provide greater detail for the justification for Drywell Spray Flow no longer being a Type D key variable for the status of Containment Related Systems.

***RAI 1-I&C-30 Response***

Residual Heat Removal (RHR) system valve position, and RHR system flow are listed as Type D variables for all RHR operating modes including drywell sprays.

This assessment is germane to guidance provided in Branch Technical Position (BTP) 7-10, Revision 5 – March 2007, Table 1. For BWRs: Acceptable Deviations and Clarifications to Revisions 2 and 3 of Regulatory Guide 1.97.

The BWR RHR system provides multiple operating modes including Low Pressure Coolant Injection (LPCI), suppression pool cooling, shutdown cooling, and drywell spray. The system uses common pumps and piping and does not have the means to directly measure drywell spray flow. The tables are based on generic BWR/4 & /6 plants, which will use RHR system flow and drywell spray valve position, in lieu of drywell spray flow which is consistent with BTP 7-10.

***RAI 1-I&C-31***

On Pages 5-2 and 5-8, Tables 5-1 and 5-2 should provide greater detail for the justification of Control Rod Position becoming a Type D variable to monitor Safety System Performance for the Reactor Protection System and the Control Rod Drive System.

***RAI 1-I&C-31 Response***

RAI Response Section 5 provides information regarding reconciliation of RG 1.97 R3 to the results of the LTR. Section 4 of the LTR addresses methodology and results of the BWR analysis

The LTR determined that Control Rod Position, as referenced in Tables 5-1 and 5-2, meets the criteria of a Type D variable to monitor Safety System Performance for the Reactor Protection System and the Control Rod Drive System. The Control Rod Position indication system is used to determine that the

Reactor Protection system has performed its safety function by control rod insertion following an accident. The Control Rod Position system confirms that the control rod drive safety system has performed its safety function.

***RAI 1-I&C-32***

On Pages 5-3, 5-4, 5-9, and 5-10, Tables 5-1 and 5-2 should provide greater detail for the justification for the addition of Safety System Performance position switches as Type D variables.

***RAI 1-I&C-32 Response***

Tables 5-1 and 5-2 reflects the conclusions of the LTR concerning Safety System Performance position switches that includes Reactor Pressure Vessel (RPV) and Primary Containment Isolation Valve (PCIV) position indication. The Tables indicate that RPV and PCIVs position indication are Type D variables for typical BWR/4 & /6 designs. RG 1.97 Revision 3 listed PCIV position indication as a Type B variable under Maintaining Containment Integrity Function. PCIV position indication is not a Critical Safety Function as it is not relied upon in safety analysis or in BWR EPGs. PCIV position indication meets the criteria of a Type D variable, as it provides verification that the containment isolation valve safety system has performed its function.

Further justification is included in RAI Response Section 6; NEDO-33349 Section 7.4.1

***RAI 1-I&C-33***

On Pages 5-4 and 5-9, Tables 5-1 and 5-2 should explain the BWR/4 and BWR/6 differences in the Classification Basis for Other RPV Normally Closed Isolation Valve Position Switches on valves that do not require opening for either a LOCA or pipe break outside of containment.

***RAI 1-I&C-33 Response***

There are no classification differences between BWR/4 and /6, with respect to Normally Closed Isolation Valve Position Switches – those that do not require opening during a LOCA or during a pipe break outside of containment.

Table 5-1 will be revised aligning it with Table 5-2 section for Classification Basis:

"Not required for safety system performance indication."

***RAI 1-I&C-34***

On Pages 5-4 and 5-9, Tables 5-1 and 5-2 should explain the BWR 4 and BWR 6 differences in the Classification Basis for Normally Closed Containment Isolation Valve Position Switches on valves inside or outside containment that do not require opening for a loss-of-coolant accident.

***RAI 1-I&C-34 Response***

There are no classification differences between BWR/4 and /6, with respect to Normally Closed Containment Isolation Valve Position Switches – on valves inside or outside containment, that do not require opening during a LOCA.

Table 5-1 will be revised aligning it with Table 5-2 section for Classification Basis:

"Not required for safety system performance indication."

***RAI 1-I&C-35***

On Pages 5-5 and 5-11, Tables 5-1 and 5-2 should detail information concerning a generic alternate means for providing Cooling Water Temperature to Engineering Safety Feature (ESF) System Components as a Type D key variable to monitor operation of the Cooling Water System. Otherwise the review of alternate means should be plant specific. On Page 7-2, Section 7.1 address plant specific deviations for Cooling Water Temperature to ESF System Components and Cooling Water Flow to ESF System Components from Category 2 to Category 3 when ESF Room Temperature and essential service water (ESW) Pump Running are used as Category 2 variables. On Pages 5-5 and 5-11, Tables 5-1 and 5-2 list Equipment Area Cooling System Cooling Water Temperature and Essential Service Water System Flow, but Cooling Water Temperature to ESF System Components and Cooling Water Flow to ESF System Components are not listed. Is this supposed to be a generic change? If so, provide detailed information concerning this generic change. Otherwise Cooing Water Temperature to ESF System Components and Cooling Water Flow to ESF System Components should be included in Tables 5-1 and 5-2.

***RAI 1-I&C-35 Response***

The LTR is not intended to seek generic approval of plant specific deviations, alternate means should be addressed on a plant specific basis. The LTR addresses generic design features for BWR systems. A specific BWR/4 and

BWR/6 plant design was used to assist in this effort, addressing areas such as cooling water systems which have differences throughout the BWR fleet.

Both Section 7.1 and Table 5.1 and 5.2 of the LTR will be revised. The list of plant specific deviations included in Section 7.1 will be removed from the LTR.

Table 5.1 and 5.2 will be revised to list generic terminology used in RG 1.97 Rev 3 related to ESF cooling water temperature and flow instrumentation. Also included is RHR Service water which is a common BWR system providing cooling water to ESF components. Emergency Service Water is a common term but has system differences in the BWR fleet.

***RAI 1-I&C-36***

On Pages 5-5 and 5-11, Tables 5-1 and 5-2 should explain the apparent discrepancy between the EQ and the SQ for DC Power Status between BWR/4 and BWR/6. For BWR/4 EQ and SQ are listed as 'Yes' and for BWR/6 EQ and SQ are listed as 'No'.

***RAI 1-I&C-36 Response***

LTR Table 5-2 for BWR/6 – Variable for DC power status, will be revised to include (Y = Yes):

EQ = Y
SQ = Y

***RAI 1-I&C-37***

On Pages 5-6 and 5-12, Tables 5-1 and 5-2 should provide greater detail for the justification for the addition of Off Gas System Release Point Radiation Level, Ambient Air Temperature, and Control Room Area Radiation Monitors as Type E variables.

***RAI 1-I&C-37 Response***

LTR Table 5-1 and 5.2 "Comments" section will be revised to indicate the following:

Variable: Offgas system release point –
Comments: Included as a Type E variable, consistent with identified pathways contained in RG 1.97 R4.

Variable: Ambient air temperature –
Comments: Included as a Type E variable, consistent with release magnitude determination parameters contained in RG 1.97 R4.

Variable: Control room area radiation monitors –
Comments: Included as a Type E variable, consistent with plant recovery access parameters contained in RG 1.97 R4.

***RAI 1-I&C-38***

On Page 6-1, Section 6 discusses guidelines for application to specific plants. Although the accident monitoring instrumentation in currently licensed plants meet the plants licensing basis, licensees converting to RG 1.97 Revision 4 will need to clearly identify design differences and equivalent variables in their plant specific applications for use of RG 1.97 Revision 4 and document any deviations. This topic should be addressed in Section 6.

***RAI 1-I&C-38 Response***

LTR Page 6-1, second paragraph, will be revised to indicate:

In implementing the evaluation methodology, it is important to recognize the unique design features. These design features can have a significant impact on the result of the safety analysis and other equivalent parameters that can be used as an alternative to direct system performance measurements. Use of this LTR in adopting RG 1.97 R4, requires the licensee to identify design differences and Variable category equivalencies, where applicable The application to a specific plant is to be consistent with the plant's current licensing design basis, including the requirements for environmental and seismic qualification

***RAI 1-I&C-39***

On Page 6-4, Section 6.5 discusses compliance with IEEE-497 referenced standards. When IEEE standards are issued they reference the latest versions of issued IEEE standards. The NRC staff recognizes that existing plant's current licensing bases predate many of these referenced standards. While it is anticipated that existing plants will maintain their current licensing basis, it is expected that each licensee will document deviations from RG 1.97 Revision 4 in their plant-specific applications for the use of RG 1.97 Revision 4. This topic should be addressed in Section 6.5.

***RAI 1-I&C-39 Response***

Response to RAI 1-I&C-38 addresses need for plant specific reviews prior to use.

LTR Section 6.5 will be revised to indicate recommendations for documentation of codes and standards as compared to IEEE 497-2002, with respect to plant specific applications.

***RAI 1-I&C-40***

On Page 7-1, Section 7.1 references Standard Review Plan Branch Technical Position (BTP) HICB 10-5. The SRP was updated in March 2007. As part of this update, BTP HICB 10-5 was renumbered as BTP 7-10. Section 7.1 should be updated to change "HICB-10-5" to "7-10 (Reference 5)," and to reference the information in the March 2007 update of the SRP.

***RAI 1-I&C-40 Response***

Section 7.1, NRC Approved Deviations to Regulatory Guide 1.97 Revisions 2 and 3, will be revised to indicate the following:

Included within the results are variables which have resulted in NRC approval of plant specific deviations. Justification for the deviations can be found in Standard Review Plan NUREG 0800, Section 7, BTP 7-10 - Guidance on Acceptance of RG 1.97. NUREG 0800 Section 7, BTP 7-10 provides criteria for use of RG 1.97 R4, and includes Table 1 to BTP HICB-10-5 Revision 5 (March 2007) for deviations approved by the NRC for BWRs.

***RAI 1-I&C-41***

On Page 7-1, Section 7.1 discusses the acceptance of RG 1.97 Revision 3 Category 3 alternate instrumentation in lieu of Category 1 Drywell Sump and Drywell Drain Sump Level instrumentation. This acceptance was based on meeting certain conditions as listed in Table 1 of SRP BTP 7-10. Section 7.1 should either include these conditions or include a determination that all existing BWRs meet these conditions, if that is the case.

***RAI 1-I&C-41 Response***

See response to RAI 1-I&C-16 concerning drywell drain sump level as a Type C variable.

The LTR concludes Drywell Drain Sump is not a post accident instrument required for a generic BWR. The Drywell Drain Sump is isolated on a loss of coolant accident. The instrumentation neither automatically initiates nor alerts the operator to initiate operation of a safety system in a post-accident situation. That is, the Drywell Drain Sump level indication is used to determine potential degradation in the Reactor Coolant Pressure Boundary during normal plant operation only, so that plant repairs can be made promptly prior to any potential failure. The generic BWR design meets all the conditions established in BTP 7-10. Plants adopting the LTR for changes to drywell drain sump level indication requirements would need to perform plant reviews to confirm the conditions are met.

***RAI 1-I&C-42***

On Page 7-1, Section 7.1 should include details concerning the basis for each of the deviations and clarifications listed in this section.

***RAI 1-I&C-42 Response***

The list of plant specific deviations has been removed from the LTR.

***RAI 1-I&C-43***

On Page 7-2, Section 7.1 includes a list of plant-specific deviations. If these deviations are applicable to the majority of BWRs, the justification for applicability should be included along with references to plant-specific safety evaluations where these deviations were approved. If these deviations are not applicable to the majority of BWRs, why is a list of plant-specific deviations included?

***RAI 1-I&C-43 Response***

The list of plant specific deviations has been removed from the LTR.

***RAI 1-I&C-44***

On Page 7-3, Section 7.2 should clarify the conclusion that the five variables listed should be in the TSs along with any additional plant-specific Type A, Type B, or Type C variables.

***RAI 1-I&C-44 Response***

LTR Section 7.3 (previously 7.2) will be expanded to provide additional information concerning Technical Specifications.

The LTR provides a generic list of Type A variables that includes Reactor Water Level, Reactor Pressure, Drywell Pressure, Suppression Pool Temperature, and Suppression Pool Level instrumentation. In addition, the LTR concludes that neutron flux to be a Type B variable. The current list of standard BWROG post accident monitoring (PAM) instrumentation in the Technical Specifications includes Type A and non-Type A, Category 1 variables which is based on RG 1.97 R3 requirements. Prior NRC agreement in a separate BWROG effort contained in NEDO-31558 provided conditions for plant exclusion of neutron monitoring from Technical Specifications.

The LTR establishes a generic basis Type A variables was based on RG 1.97 R4 (IEEE Standard 497-2002) selection criteria. RG 1.97 R4 eliminates instrument qualification categories, and, thus, non-Type A Category 1 variables. The BWROG intends to pursue Technical Specification changes consistent with conclusions of the LTR.

The intended Technical Specification changes will address the guidance provided in the NRC letter dated May 9, 1988 from Thomas Murley to NSSS Owners Groups – containing the original inclusion of RG 1.97 into PAM Standard Technical Specifications, and what is required for changes related to risk information.

A preliminary review concludes that RG 1.97 R4, Type A, B and C variables will meet the criteria for inclusion in NUREG 1433 and NUREG 1434 Standard Technical Specifications. Several RG 1.97 R3, non-Type A, Category 1 variables currently included in Technical Specifications may not meet the criteria for inclusion. A separately proposed PAM Technical Specification change will be processed through the Technical Specification Task Force (TSTF) process for changes to BWROG Standard Technical Specifications (NUREG 1433/1434).

***RAI 1-I&C-45***

On Page 7-4, Section 7.3.2 lists Containment Area High-Range Radiation as a RG 1.97 Revision 3 Type C Category 1 variable. This is not correct. Containment Area Radiation is a RG 1.97 Revision 3 Type C Category 3 variable and Containment Area High-Range Radiation is a RG 1.97 Revision 3 Type E Category 1 variable. Licensees should be reminded that even if RG 1.97 Revision 4 allows different design and qualification criteria for Containment Area High-Range Radiation instrumentation, the criteria of NUREG-0737 Item II.F.1 are still required to be addressed.

***RAI 1-I&C-45 Response***

Containment radiation monitors which were used in BWR plant designs prior to NUREG 0737, imposed requirements listed in RG 1.97 R3 as Type C, Category 3. Section 7.4.2 (previously 7.3.2) will be corrected. The need to address NUREG-0737 criteria has been included in several sections of the LTR and is specifically addressed in the response to RAI 1-I&C-24 which indicates NRC acceptance of this LTR methodology will be used as a basis for plant specific reviews of their post accident monitoring requirements and licensing commitments including but not limited to items included in both NUREG 0737 and RG 1.97 Revision 2 or 3.

***RAI 1-I&C-46***

On Page A-1, Appendix A should discuss that although Type E instrumentation is not required to be environmentally qualified, this instrumentation is expected to be designed to operate in the environment that it will see when it is needed to monitor its designated variable.

***RAI 1-I&C-46 Response***

Section 2.6 of the LTR will be revised to indicate the following:

Type E variables are used for monitoring and assessment of release magnitude, environmental monitoring, and radiation level monitoring in plant environs. Operational specifications of the instrumentation (range and ruggedness) will be consistent with the associated service environment. Type E variable qualification and design requirements are defined in IEEE Std. 497, which states, "Instrument channels that monitor systems are not required to be environmentally or seismically qualified."

Type E instruments should be of a high-quality commercial grade, selected to withstand the specified service environment. This is consistent with provisions of RG 1.97 R2 & R3 for Category 3 variables.

Instrumentation related to Type E variables, will be used by station operators for evaluating mitigation strategies associated with Emergency Planning (EP).

***RAI 1-I&C-47***

On Page A-1, Appendix A should explain the significance of the BWR/4 columns in Table A-1. Are these columns supposed to indicate if there is a difference between BWR/6 and BWR/4 designs for an individual variable?

***RAI 1-I&C-47 Response***

The intent of Appendix A of the LTR, is to assist with reconciliation of LTR results to RG 1.97 REV. 2 / REV. 3. A column was added to show how a typical BWR/4 currently lists RG 1.97 variables, which includes NRC approved plant specific deviations (for information purposes). The Table is not intended to address BWR/4 & /6 design differences, as it reflects a specific BWR 4. Tables 5-1 and 5-2, depicts results of the LTR for BWR/4 & /6 designs using RG 1.97 R4 criteria.

***RAI 1-I&C-48***

On Page A-2, Table A-1 shows that for the Core Cooling function, Reactor Water Level, in the BWR/4 column is listed as a Type A Category 1 variable. Is this supposed to indicate that for BWR/4 reactors the Reactor Water Level is a Type A variable and not a Type B variable? If so, what BWR/4 variable fulfills the Core Cooling function? If not, why is Coolant Level in Reactor not listed as a Type B variable in the BWR/4 column? If reactor Water Level is not shown in Table A-1 as a key variable for a Type C function, why is it listed in the IEEE-497 Type column as A, B, C?

***RAI 1-I&C-48 Response***

The BWR/4 column is intended to depict how a specific plant which is a BWR/4 currently identifies variables in compliance with RG 1.97 R2 (the plant has a licensing commitment to R2), including approved plant deviations. This information was included in the LTR for information purposes only. Further review, indicates that the table should be more consistent with RG 1.97 R2, except for deviations and to show reactor water level as Type A and B variables required by RG 1.97 R2.

The Table has been revised to indicate that Reactor Water Level is a Type A and B variable, consistent with R2 requirements. The conclusion of the LTR is depicted in the column titled "IEEE 497", which states per RG 1.97 R4 criteria, that Reactor Water Level is a Type A, B and C variable.

***RAI 1-I&C-49***

On Pages A-2 and A-5, Table A-1 show that for the Maintain RCS Integrity function, the Reactor Coolant Pressure Boundary function, and the Containment function, Reactor Pressure, in the BWR/4 column is listed as a Type A Category 1 variable. Is this supposed to indicate that for BWR/4 reactors the Reactor Pressure is a Type A variable and not a Type B or Type C variable? If so, what BWR/4 variables fulfill the Maintain RCS Integrity function, the Reactor Coolant Pressure Boundary function, and the Containment function? If not, why is RCS Pressure not listed as a Type B and Type C variable in the BWR/4 column?

***RAI 1-I&C-49 Response***

See response to RAI 1-I&C-48. Reactor Pressure should be listed as a Type A, B, and C variable in the BWR/4 column. RG 1.97 R2 listed Reactor Pressure as a Type B and C, and the plant concluded it met the criteria as a Type A variable. The LTR will be revised.

***RAI 1-I&C-50***

On Pages A-3, A-5, and A-6, Table A-1 shows that for the Maintain RCS Integrity function, the Reactor Coolant Pressure Boundary function, the Containment function, and the Primary Containment Related Systems function, Drywell Pressure, in the BWR/4 column is listed as a Type A Category 1 variable. Is this supposed to indicate that for BWR/4 reactors the Drywell Pressure is a Type A variable and not a Type B, Type C, or Type D variable? If so, what BWR/4 variables fulfill the Maintain RCS Integrity function, the Reactor Coolant Pressure Boundary function, the Containment function, and the Primary Containment Related Systems function? If not, why is Drywell Pressure not listed as a Type B, Type C, or Type D variable in the BWR/4 column?

***RAI 1-I&C-50 Response***

See response to RAI 1-I&C-48. Drywell Pressure should be listed as a Type A, B, C, and D variable in the BWR/4 column. RG 1.97 R2 listed Drywell Pressure as a Type B, C, and D variable and the plant concluded it met the criteria as a Type A variable. The LTR will be revised.

***RAI 1-I&C-51***

On Page A-6, Table A-1 shows that for the Primary Containment Related Systems function, Suppression Pool Temperature, in the BWR/4 column is listed as a Type A Category 1 variable. Is this supposed to indicate that for BWR/4 reactors the Suppression Pool Temperature is a Type A variable and not a Type D variable? If so, what BWR 4 variable fulfills the Primary Containment Related Systems function? If not, why is Suppression Pool Temperature not listed as a Type D variable in the BWR/4 column? If Suppression Pool Temperature is not shown in Table A-1 as a key variable for a Type B or C function, why is it listed in the IEEE-497 Type column as A, B, C?

***RAI 1-I&C-51 Response***

See RAI 1-I&C-48 response. Suppression pool temperature will be shown as Type A and D in the BWR 4 column. RG 1.97 R2 listed as a Type D, and the plant concluded it also was Type A. The LTR concludes suppression pool temperature is a Type A, B, and C variable.

***RAI 1-I&C-52***

On Page A-5 and A-6, Table A-1 shows that for the Reactor Coolant Pressure Boundary function and the Primary Containment Related Systems function, Suppression Pool Water Level, in the BWR/4 column is listed as Type A Category 1 variable. Is this supposed to indicate that for BWR 4 reactors the Suppression Pool Water Level is a Type A variable and not a Type C or D variable? If so, what BWR 4 variables fulfill the Reactor Coolant Pressure Boundary function and the Primary Containment Related Systems function? If not, why is Suppression Pool Water Level not listed as a Type C and Type D variable in the BWR/4 column? If Suppression Pool Water Level is not shown in Table A-1 as a key variable for a Type B function, why is it listed in the IEEE-497 Type column as A, B, C?

***RAI 1-I&C-52 Response***

See RAI 1-I&C-48 response. Suppression pool water level has been shown as a Type A, B, C variable in the BWR/4 column.

***RAI 1-I&C-53***

On Page A-7, Table A-1 should provide greater detail for the justification for Isolation Condenser System Shell Side Water Level no longer being a Type D key variable to monitor the operation of Safety Systems.

***RAI 1-I&C-53 Response***

Table A-1 depicts a BWR/4 design. Isolation Condensers are associated with BWR/2 & /3 plant designs.

Section 6.3 addresses isolation condensers:

Performance of the isolation condenser is indicated by Type B variables – condensate return valve position (Type D), and isolation condenser shell water level indication (Type D).

Specific BWR designs with isolation condensers are likely to include indications for isolation condenser shell water level and condensate return valve position, as Type D variables, or provide justification using alternate indications.

***RAI 1-I&C-54***

On Page A-7, Table A-1 should provide greater detail for the justification for Isolation Condenser System Valve Position no longer being a Type D key variable to monitor status of Containment Related Systems.

***RAI 1-I&C-54 Response***

Table A-1 depicts a BWR/4 design. Isolation Condensers are associated with BWR/2 & /3 plant designs.

Section 6.3 addresses isolation condensers:

Performance of the isolation condenser is indicated by Type B variables – condensate return valve position (Type D), and isolation condenser shell water level indication (Type D).

Specific BWR designs with isolation condensers are likely to include indications for isolation condenser shell water level and condensate return valve position, as Type D variables, or provide justification using alternate indications.

***RAI 1-I&C-55***

On Page A-8, Table A-1 should provide greater detail for the justification for how High Radioactivity Liquid Tank Level being a normal operating system would have an impact on it meeting the criteria for a Type D variable for monitoring the operation of Radwaste Systems.

***RAI 1-I&C-55 Response***

The LTR concludes that the RG 1.97 R4 Type D variable category is not applicable to BWR liquid radwaste system designs. BWR radwaste systems are normal operating systems as described in Section 4.4.3. Liquid radwaste systems are not required for the mitigation of accidents. The High Radioactivity Liquid Tank is not a typical BWR radwaste component.

***RAI 1-I&C-56***

On Page A-8, Table A-1 should include greater detail for the justification for a generic alternate means for providing Emergency Ventilation Damper Position as a Type D key variable to monitor operation of the Ventilation Systems. Otherwise the review of alternate means should be plant specific.

***RAI 1-I&C-56 Response***

The intent of LTR Appendix A is to assist in reconciliation of LTR results as compared to RG 1.97 R2 & R3. A column was added to show how a typical BWR/4 lists RG 1.97 variables, including NRC approved plant specific deviations (for information purposes). The plant depicted in the column, received approval for use of differential pressure related to emergency ventilation damper position. Consistent with the exclusion of plant specific deviations, LTR Table A-1 has been revised to indicate the following:

Table A-1 – Accident Monitoring Variables Comparison

Variable: Emergency ventilation damper position –
Comments: "Differential pressure is an acceptable alternative." will be deleted through LTR revision.

***RAI 1-I&C-57***

On Page A-9, Table A-1 should provide greater detail for the justification for Secondary Containment Release Point Radiation Level becoming a Type E variable.

***RAI 1-I&C-57 Response***

LTR Table A-1, "Comments" section will be revised to indicate the following:

Table A-1 – Accident Monitoring Variables Comparison

Variable: RG 1.97 Rev. 3 – Type E Variables –
Comments: "All potential releases points from the plant will be monitored or assessed for radioactivity concentration and release rate."

LTR Tables 5-1 and 5-2, include classification bases details for variables related to plant releases.

***RAI 1-I&C-58***

On Page A-9, Table A-1 should provide greater detail for the justification for Radiation Exposure Rate no longer a Type E variable for Area Radiation. RG 1.97 Revision 3 includes Radiation Exposure Rate as a Type E variable to provide detection of significant releases, release assessment, and long-term surveillance for Area Radiation. On Page 7-1, Section 7.1 identifies Radiation Exposure Rate as being granted a generic deviation to Category 3. Therefore, Radiation Exposure Rate should be included in Table A-1.

***RAI 1-I&C-58 Response***

Radiation exposure rate has been included in Table A-1. RG 1.97 R4 Type E stipulates the need to monitor radiation levels in accessible plant areas.

***RAI 1-I&C-59***

On Page A-9, Table A-1 should provide greater detail for the justification for RG 1.97 Revision 3 Airborne Radiation variables no longer being Type E for Airborne Radioactive Materials Released from Plant variables for detection of significant releases and release assessment for Airborne Radioactive Materials Released from Plant. RG 1.97 Revision 3 includes (a) Noble Gases and Vent Flow Rate and (b) Particulates and Halogens as Type E variables to provide detection of significant releases, release assessment, and in some locations long term surveillance, for Airborne Radioactive Materials Released from Plant. On Page A-9, Table A-1 lists (a) Noble Gases and Vent Flow Rate and (b) Particulates and Halogens as Type D variables. Provide detailed justifications for these changes.

***RAI 1-I&C-59 Response***

Table A-1 to be revised to show Airborne Radioactive Materials Released from Plant, as a Type E variable. Table 5-1 Page 5-9 contains lists of release points for BWR/4 designs.

***RAI 1-I&C-60***

RG 1.97 Revision 3 includes Estimation of Atmospheric Stability as a Type E variable. On Page A-9, Table A-1 also lists Estimation of Atmospheric Stability as a Type E variable. However, on Pages 5-6 and 5-12, Tables 5-1 and 5-2 list Ambient Air Temperature as a Type E variable. Discuss the differences, if any, between Estimation of Atmospheric Stability and Ambient Air Temperature and provide appropriate detailed justification for any deviation.

***RAI 1-I&C-60 Response***

RG 1.97 R4 states that Type E variables "monitor the environmental conditions used to determine the impact of releases of radioactive materials through identified pathways (e.g., wind speed, wind direction and air temperature)." Table 5-1 is consistent with the tenets of RG 1.97 R4. Monitoring expectations of atmospheric instability and ambient air temperature parameters are consistent between RG 1.97 R3 and IEEE 497-2002.

***RAI 1-I&C-61***

On Page A-9, Table A-1 should provide greater detail for the justification for Primary Coolant and Sump no longer being Type E for release assessment, verification, and analysis of Accident Sampling Capability.

***RAI 1-I&C-61 Response***

LTR Table A-1, "Comments" section will be revised to indicate the following:

Table A-1 – Accident Monitoring Variables Comparison

Variable: Primary coolant and sump –

Comments: "NEDO-32991A (Aug 2001) contains justification for elimination of PASS, including sampling associated with primary coolant, suppression pool, and building sumps."

***RAI 1-I&C-62***

On Page A-9, Table A-1 should provide greater detail for the justification for Containment Air no longer being Type E for release assessment, verification, and analysis of Accident Sampling Capability.

***RAI 1-I&C-62 Response***

BWROG LTR NEDO 32991-A (Aug 2001) "Regulatory Relaxation for Post Accident Sampling System (PASS)," addresses the elimination of permanent plant equipment for Containment Air Sampling. Containment Air sampling will be included in the LTR as a Type E variable, with reference to the PASS LTR.

## Section 2: Instrumentation and Controls Branch RAI

### *RAI 2-I&C-1*

Page 4-7 of LTR NEDO-33349, Section 4.2.2 includes a discussion of critical safety functions. Sub-clause 3.7 of IEEE standard 497-2002 defines five critical safety functions. These five IEEE standard 497-2002 critical safety functions are: reactivity control, reactor core cooling, reactor coolant system integrity, primary reactor containment integrity, and radioactive effluent control. However, Section 4.2.2 only identifies four critical safety functions and these are different from the five critical safety functions that are defined in IEEE standard 497-2002. Section 4.2.2 should be modified to address each of the five IEEE Standard 497-2002 critical safety functions. The Type B key variables that provide information about each of the five IEEE Standard 497-2002 critical safety functions should be clearly identified.

### *RAI 2-I&C-1 Response*

Section 4.2.2 of the LTR will be revised to reconcile how the five critical safety functions (listed in Section 3.7 of IEEE Standard 497-2002) are satisfied, as compared to the four critical safety functions identified in the Topical Report. The Primary Containment Control function also includes the Radioactive Effluent Control safety function.

Selection criteria in IEEE Standard 497-2002, defines instrumentation variable types based on the level of importance to the operators. The Critical Safety Function (CSF) concept evolved from the implication that the operator need only monitor relatively few pieces of information to ascertain the safety of the plant. The operator can carry out duties by focusing on these critical functions without regard to the specific events that have occurred.

IEEE Standard 497-2002, Item 4.2 (associated with Type B variables relationship to EPGs and the CSF status trees) is not directly applicable to the BWR technology.

The BWR Emergency Procedure Guidelines (EPG) are symptom based and do not rely on the PWR-related CSF concept for procedural execution. Included in the LTR, are results of a review of the EPGs that addressed the application of the CSF concept for typical BWR accident monitoring instrumentation, ensuring the integrity of radiological barriers.

IEEE Standard 497-2002, Item 3.7 states: "The CSF are those safety functions that are essential to prevent a direct and immediate threat to the health and safety of the public by maintaining Reactivity control, Reactor core cooling, Reactor coolant system integrity, Primary reactor containment integrity, and Radioactive effluent control;" and Section 4.2 states "Type B variables provide primary information to the operators to assess the plant CSF for the Emergency Operating procedures implementation."

IEEE Standard 497-2002 separated the Radioactive Effluent Control from the Maintaining Containment Integrity function and listed it as a new function. RG 1.97, Revision 3 listed Radioactive Effluent Control as part of Maintaining Containment Integrity function for Type B variables.

Post-accident Radioactive Effluent Control is primarily concerned with the potential for open radioactive release pathways from the primary containment to the secondary containment. In the BWR design, the Radioactive Effluent release pathways are deliberately isolated following receipt of LOCA isolation signals to establish containment integrity.

The LTR, Section 4.2.2 states: "...the Primary Containment contains isolation features that provide a barrier to the release of radioactive material due to the postulated loss of coolant accident from the primary containment to the secondary containment. Therefore, this barrier is assumed to remain intact for the postulated loss of coolant accident and limit any leakage of radioactive material to the secondary containment." These features ensure the Radioactive Effluent Control function (as described in IEEE Std. 497-2002; Section 3.7) is met in terms of accident mitigation. Thus, the critical safety function of Primary Containment Control also ensures Radioactive Effluent Control safety function in the BWR design.

The BWROG position on the compliance with the design and selection criteria of RG 1.97 R4 (IEEE Standard 497-2002) and the assignment of the Type B key variables for monitoring the CSF is discussed in Section 4.2.2 of the LTR. To establish the CSF for BWRs, the LTR assessed design criteria established in the IEEE Standard 497-2002 and provided a list of four CSFs (Reactivity control, Pressure control, Level control, and Primary containment control).

A review of the EPG, including EPG entry conditions, was performed to ensure that all safety related structures, systems and components (SSC) that provide the accident monitoring safety functions were addressed. Type B key variables applicable to the BWR design (Reactor Power/Neutron Flux, Reactor Pressure, Reactor Water Level, and Suppression Pool Temperature, Suppression Pool Water Level and Drywell Pressure) represent a conservative statement of conditions which, if generally met, will provide a high degree of confidence that public safety is protected even in an unlikely event of a loss of coolant accident.

Note that beyond the identification as the Type B CSF, the Radioactive Effluent Control is inherent in Type C and Type E of RG 1.97 R4 criteria for accident monitoring instrumentation. Type C variables provide information about the potential or the actual breach of the fission product barriers and Type E provides information about the magnitude and impact of the release of radioactive material, respectively. Thus, radioactive effluent control is comprehensively addressed by RG 1.97 R4 and by the LTR methodology.

## Section 3: Nuclear Security and Incident Response (NSIR) RAI

### *RAI 3-NSIR-1*

Section 1.3.6 should be expanded to acknowledge, and provide guidance on, situations in which the current licensing basis requirements are more restrictive than the requirements of Revision 4 to RG 1.97. Please add appropriate language or provide justification why the BWROG believes that this NUREG-0737, Section II.F.1, Attachmentchange is not necessary. For example: 1, "Noble Gas Effluent Monitors" (high range), requires that the monitors are capable of performing their intended function in the environment to which they may be exposed during accidents, be powered from vital instrumentation bus or dependable backup power supply, and that their operability be addressed by TSs. NUREG-0737, Section II.F.1,This parameter is designated as a Type E variable. Attachment 2, "Sampling and Analysis of Plant Effluents," requires the preparation NUREG-0737, Section of TSs. This parameter is designated as a Type E variable. II.F.1, Attachment 3, "Containment High Range Radiation Monitor," requires a minimum of two Category 1 containment high-range monitors qualified to function in the accident environment, be powered from Category 1E power sources, and that their operability be addressed by TSs. This monitor is designated as a Type C and a Type E variable.

### *RAI 3-NSIR-1 Response*

Section 1.3.7 added to the LTR to address this request (also, see response to RAI 1-I&C-23 for additional information):

"Plant specific reviews will need to be performed of current plant licensing basis requirements including commitments to NUREG 0737 prior to the application of the results of this Report to plant changes."

NUREG-0737 resulted in changes to plant current licensing basis including the provisions contained in NUREG 0737, Section II.F.1 as referenced in the RAI. RG 1.97 Revision 2 and 3 captured accident monitoring requirements which included NUREG 0737 requirements by establishing Tables which contain lists of variables for BWRs and assigning Categories to address design and qualification requirements. CHARM is listed in RG 1.97 Revision 2 and 3 as Type E Category 1 monitors (highest requirements) and noble gas as Type E Category 2 (requires EQ).

The methodology used in the LTR based on RG 1.97 Revision 4, results in the same conclusion as that in RG 1.97 Revision 2 and 3 – CHARM and noble gas monitors are Type E variables. RG 1.97 Revision 4 establishes consistent design requirements for all Type E variables resulting in the differences from what is in NUREG 0737.

There have been NRC approved changes to requirements originally imposed in NUREG 0737, including what is shown in NUREG 0737, Section II.F.1 above. Attachment 1 on noble gas monitors indicates the inclusion within Technical Specifications. The original Standard Technical Specifications which incorporated RG 1.97 post accident monitoring did not include noble gas monitors. Attachment 2, "Sampling and Analysis of Plant Effluents," was addressed in BWROG LTR NEDO-32991-A on PASS, which also addressed reconciliation with NUREG 0737 imposed requirements.

***RAI 3-NSIR-2***

Although the LTR has references to the current plant licensing basis and the need to perform evaluations against this licensing basis, the NRC staff believes that more specificity is needed, and requests that a clarification such as that suggested below be added to Section 1.4, "Limitations." Please incorporate appropriate language, or provide justification why the BWROG believes that this change is not necessary. "Proposed changes to a plant's accident monitoring variables, their classification under Regulatory Guide 1.97 Revision 4, and the associated treatment requirements (e.g., environmental qualification, technical specifications, etc.) must be evaluated within the context of the specific plant's current licensing basis pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.59. In addition, proposed changes to any instrumentation, relied upon by the plant's emergency plans to meet the planning standards of 10 CFR 50.47(b) and Appendix E to Part 50, must be evaluated pursuant to 10 CFR 50.54(q) to ascertain whether the proposed change would decrease the effectiveness of those plans. In this context, any change that would reduce the performance, reliability, or availability of such instruments, without compensatory measures, during an emergency condition will likely constitute a potential decrease in the effectiveness of the plans, requiring prior NRC approval."

***RAI 3-NSIR-2 Response***

The following will be added to Section 1.4, Limitations of the LTR (also, see response to RAI 3-NSIR-1 above):

"Proposed changes to a plant's accident monitoring variables, their classification under Regulatory Guide 1.97 Revision 4, and the associated treatment requirements (e.g., environmental qualification, technical specifications, etc.) must be evaluated within the context of the specific plant's current licensing basis pursuant to 10 CFR 50.59. In addition, proposed changes to any instrumentation relied upon by the plant's emergency plans to meet the planning standards of 10 CFR 50.47(b) and Appendix E to Part 50, must be evaluated pursuant to 10 CFR 50.54(q) to ascertain whether the proposed change would decrease the effectiveness of those plans."

The last sentence of the suggested wording in the RAI was not included as the plant 10CFR 50.54 (q) evaluations will make that determination.

### *RAI 3-NSIR-3*

Section 1.5 appears to limit the applicability to plant-specific commitments with respect to accident monitoring that are documented in "the UFSAR [update final safety analysis report] or other applicable license amendment documents." The NRC staff believes that the last sentence of this section should read: "...include all plant-specific commitments with respect to accident monitoring that are documented in the current licensing basis, including but not limited to the UFSAR." Please incorporate appropriate language, or provide justification why the BWROG believes that this clarification is not required.

### *RAI 3-NSIR-3 Response*

The last sentence of Section 1.5 will be changed to "Licensing design basis requirements include all plant specific commitments with respect to accident monitoring that are documented in the current licensing basis, including but not limited to the UFSAR."

### *RAI 3-NSIR-4*

The last paragraph of Section 2.1 appears to inappropriately link the safety analyses with emergency procedure guidelines (EPGs) and emergency operation procedures (EOPs). Section 1.5, provides that "safety analysis" is defined by "anticipated operational occurrences [(AOOs)] and accidents or other equivalent nomenclature used in the safety or accident analysis section of the updated final safety analysis report (UFSAR)." Yet, the safety analyses generally do not credit actions taken in accordance with EPGs or EOPs. Also, EPGs and EOPs can be generally characterized as taking credit for all available plant equipment and plant services (e.g., AC power) with "response not obtained" steps to address unavailability of that resource. However, the safety analyses only credit safety-related equipment, and offsite power is generally assumed to be lost at the accident onset. Is "safety analysis" as used here defined differently than in Section 1.5? If so, clarification is needed. Does the dichotomy in treatment impact the LTR methodology? Please reconsider this language and make necessary changes for clarity, or provide justification why the existing language is appropriate.

***RAI 3-NSIR-4 Response***

Section 1.5 addresses safety analysis and licensing design basis requirements and is consistent with what is described in Section 2.1. Section 2.1 states:

"Type A variables are defined in IEEE-497 as those variables that provide the primary information required to permit the control room operating staff to:

- Take specific planned manually-controlled actions for which no automatic control is provided and that are required for safety systems to perform their safety-related functions as assumed in the plant accident analysis.

- Take specific planned manually-controlled actions for which no automatic control is provided and that are required to mitigate the consequences of an anticipated operational occurrence.

Type A variables provide information essential for the direct accomplishment of specific safety-related functions that require manual action.

From a BWR safety analysis perspective, Type A variables are associated with providing the operator with required information for the direct accomplishment of manual actions that are assumed in the safety analysis to obtain a safe shutdown condition. For BWRs, these variables in the accident monitoring systems application methodology are a subset of those necessary to implement the EPGs and plant specific EOPs."

This definition of a Type A variable is consistent with RG 1.97 R4 (IEEE 497-2002).

The BWR safety analysis and EPG/EOPs are linked by operator actions required to achieve safe shutdown, but as noted, the manual actions defined in the safety analysis, for which no automatic control are provided, are a subset of the symptom based EPGs. All manual actions defined in the safety analysis are addressed in the BWR EPGs, but only those safety related operator actions that are credited in the safety analysis are Type A variables. As noted in the RAI, the BWR EPGs go beyond the safety analysis and safety related equipment to utilize all potentially available plant equipment. Not all EPG actions are addressed in the safety analysis, which does not rely on all the actions defined in the EPGs. Only those manual actions required in the safety analysis for which no automatic control is provided to achieve safe shutdown are defined as Type A variables.

***RAI 3-NSIR-5***

In Section 2.4 the third portion of the definition of "Safety System" is not fully consistent with the regulatory definition used in 10 CFR Part 50. The definition should read: "The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in Section 50.34(a)(1), Section 50.67(b)(2), or Section 100.11 of this chapter, as applicable." Please revise the LTR definition accordingly.

***RAI 3-NSIR-5 Response***

Section 2.4 will be revised to incorporate the suggested language.

***RAI 3-NSIR-6***

The discussion in Section 2.4, Pages 2 and 3 should be changed to acknowledge that a plant's current licensing basis safety analyses may identify other BWR accidents for which there is a significant radioactivity release. Please incorporate appropriate changes, or provide justification why the BWROG believes that these changes are not required.

***RAI 3-NSIR-6 Response***

LTR, Sections 1.4 and 1.5 addresses the need for plant specific reviews consistent with the current licensing basis to be performed. The following will be added to Section 2.4 "For typical BWRs, the accidents...."

***RAI 3-NSIR-7***

The last paragraph of Section 2.4 appears to conflict with the definition of "safety-related" established earlier in Section 2.4. Whether or not a particular system, structure, or component is safety-related is established by the three-part definition of "safety-related" without regard to any particular AOOs or accidents in which the equipment may be credited. Please incorporate appropriate changes, or provide justification why the BWROG believes that these changes are not required.

***RAI 3-NSIR-7 Response***

The last sentence of Section 2.4 will be deleted.

***RAI 3-NSIR-8***

The parenthetical phrase in the first bullet of Section 2.5 should be expanded to include standby gas treat system and reactor building ventilation. The major portion of an accident release will be via these two pathways and should not be overlooked. Please incorporate appropriate changes, or provide justification why the BWROG believes that these changes are not required.

***RAI 3-NSIR-8 Response***

LTR Section 2.5 is consistent with the definitions contained in RG 1.97 R4 and IEEE 497. LTR Section 4.5.1 addresses Standby Gas Treatment and Reactor Building Ventilation as pathways requiring monitoring.

***RAI 3-NSIR-9***

In Section 2.5 there needs to be a bullet to reflect the need for monitoring the fission product inventory in the containment atmosphere as a means of assessing potential releases to the environment that (1) have not yet started, or (2) are via unmonitored pathways. Such releases would not be indicated by effluent monitors addressed by the first bullet. Risk studies have shown that the most severe releases may be via unmonitored pathways. Since these releases will generally involve the release of fission products from the containment, monitoring the containment inventory meets an assessment need.

***RAI 3-NSIR-9 Response***

LTR Section 2.5 is consistent with RG 1.97 R4 definitions for Type E variables (also, see RAI 3-NSIR-8). BWROG LTR NEDO 32991-A dated August 2001, "Regulatory Relaxation for Post Accident Sampling System (PASS)" addresses the justification for elimination of accident sampling capability, including the containment air in-plant sampling. The NRC approved PASS LTR concludes that plants should have the capability to sample containment air but no permanent equipment was needed. We will include containment air as Type E with reference to the PASS LTR as to sampling capabilities.

***RAI 3-NSIR-10***

In Section 4.2.2 on Page 4-8 in the last sentence in the paragraph addressing the RCS fission product barrier is potentially misleading. A typical General Electric (GE) design basis analysis for the control rod drop accident postulates that the main steam isolation valves (MSIVs) remain open throughout the accident (In

NEDO-31400, the main steam line radiation monitor activation of reactor trips and MSIV isolation was eliminated); creating a bypass of both the RCS and containment barriers. Please incorporate appropriate changes, or provide justification why the BWROG believes that these changes are not required.

***RAI 3-NSIR-10 Response***

LTR Section 4.2.2 addresses critical safety functions. See response to RAI 2-I&C-1 for discussion on BWR critical safety function. Page 4-8 of Section 4.2.2 last sentence states: "Based on these critical safety functions, the applicable critical safety parameters that provide the primary information to the control room operators to assess the plant critical safety functions can be identified". The control rod drop accident, as analyzed, does not result in the need for primary information in the control room and no operator actions are expected for the generic control rod drop accident.

NEDO-31400A referenced in the RAI is not the original GE design basis analysis for Control Rod Drop Accidents (CRDA). NEDO-31400A has been used by several BWR's to support elimination of the Main Steam Line Radiation Monitor (MSLRM) High Radiation Trip signal from the plant Technical Specifications. Plant adoption of NEDO-31400A is conditioned on a review of Control Rod Drop Accident radiological consequences with the MSLRM High Radiation Trip signal defeated before the NEDO-31400A was adopted. NEDO-31400A documented the acceptable results of such CRDA.

NEDO-10527 documents the results of the original CRDA analysis that is also referenced in plants' FSARs, as applicable. The CRDA is the result of a postulated event in which a high worth control rod is inserted out-of-sequence into the core, becomes decoupled from its drive mechanism, rod is assumed to be stuck in place, and suddenly falls free and drops out of the core. This design basis CRDA results in the release of radioactivity from the RPV to the environment. The analysis assumes that all of the fission products exit the RPV prior to isolation by the MSLRM high radiation signal. Therefore, the availability of the MSLRM isolation trip does not change the analyzed consequences of the CRDA.

The original CRDA analysis credits MSLRM with providing a reactor scram signal in addition to closing the Main Steam Line Isolation Valves (MSIV). For this accident, a scram will be initiated first by the neutron monitoring system, which senses the increase in neutron flux.

The plants that eliminated the MSLRM high radiation trip, considered the radiation transport pathway consisting of radioactive material carryover with steam to the turbine condenser with leakage from the condenser to the environment. The affect of eliminating the MSLRM on the radiological consequences of a CRDA by isolating the MSIV was re-analyzed by General

Electric in NEDO-31400A which resulted in the NRC acceptance that the MSIV closure on high radiation could be eliminated. Plant specific licensing applications including the CRDA radiological consequences are required for implementation of this change. Not all BWR's have adopted NEDO-31400A, but those who have, were required to perform a plant specific evaluation, meet imposed conditions for acceptance, and obtain separate NRC approval prior to implementation. All BWR Owners have analysis and NRC approval on their current licensed CRDA analysis.

***RAI 3-NSIR-11***

Section 4 discusses the application methodology to be used to determine the accident monitoring variables consistent with the requirements of RG 1.97 Revision 4 and IEEE-497. The subsections for Section 4 then develop the accident monitoring variables associated with each of the variable types. Section 4.3 and Section 4.3.2 state that "Type C variables are selected to represent the minimum set of parameters that provide the *most direct indication* of the integrity of the fission product barriers and provide a *capability for monitoring* beyond the normal operating range [emphasis added]." The NRC staff questions the implementation of this stipulation in arriving at the parameter listings for the fuel clad fission product barrier. In particular, Section 4.3.4.1 identifies the following two parameters as meeting the characterization for the fuel clad barrier:

- Reactor water level
- Off gas activity (monitoring performed by normal operating systems)

The NRC staff is of the opinion that this listing inappropriately omits the containment high range radiation monitor -- most direct indication of the integrity of the fuel clad fission product barriers and the instrument most capable of monitoring beyond normal operating ranges. It is important to note that an upscale reading on this monitor is indicative of a breach of both the RCS barrier and the fuel clad barrier, and is therefore an indication for both barriers, differing only in the higher magnitude indication associated with fuel barrier failure as opposed to normal RCS activity associated with an RCS barrier failure.

A. Although the NRC staff agrees that a decrease in reactor water level is a required precursor to fuel damage and is an indicator of potential fuel damage, it is not a direct indicator of that damage. As such, this parameter does not appear to meet the Section 4.3.2 basis as a direct indicator. Consider an accident sequence in which the emergency core coolant system (ECCS) is not initially successful, allowing water level to decrease below the minimum steam cooling reactor water level (MSCRWL), with the ECCS then restored and water level restored (1) prior to significant fuel clad damage, or (2) after fuel clad damage has occurred. In either of these sequences, the reactor water level is not likely a reliable indicator of fuel damage. Monitoring this Reactor water level is not a means for monitoring the integrity of the fuel clad once the water level has been restored.

B. The NRC staff notes that the MSCRWL is a calculated parameter based on conservative analysis assumptions and that, depending on the actual transient conditions, may be uncertain. Since the containment high range monitors directly monitor the increase in radiation levels in the containment atmosphere, a direct consequence of the release of fission products from the fuel, the uncertainties are expected to be less.

C. The off gas radiation monitors are typically designed to detect increases in main steam activity comparable to TS limiting conditions for operation. Although these monitors can be most the sensitive and timely indicators of increased RCS and main steam activity (e.g., clad defects, etc.) during normal operating conditions, they would likely be off scale for an incident involving substantial fuel damage. As such, this parameter does not appear to meet the Section 4.3.2 basis as being capable of monitoring beyond normal operating ranges. Also, since these monitors are located downstream of the main condenser, isolation of the MSIVs following a design basis LOCA, or an accident in conjunction with a loss of circulating water to the main condenser (e.g., loss of offsite power), will effectively isolate the off gas monitors rendering them unusable as a direct indicator or monitor of fuel damage. There is a high degree of uncertainty associated with the transport and deposition of radioactive materials through the main steam piping, in the main condenser, and in the various components and filter media of the off gas system, making the process of equating the reading on the off gas radiation monitor to fuel clad status, uncertain. For the reasons stated above, the NRC staff does not agree with the omission of the containment high range monitor from this list of Type C variables and from Tables 5-1 and 5-2 and requests that the BWROG reconsider its omission, or provide additional justification supporting the use of the reactor water level and off gas activity as meeting the parameter characteristics in Section 4.3.2. Such justification must address the NRC staff's concerns identified above and show that alternative variables meet the direct and beyond normal range characterization provided in Section 4.3.

***RAI 3-NSIR-11 Response***

The RAI raises several issues which are addressed. They are

- The basis for determining that RPV water level is the appropriate Type C variable for fuel cladding barrier (11)
- Postulated scenarios which challenge reliance on RPV water level (11.a)
- Uncertainties with use of RPV water level (11.b)
- Off Gas Radiation monitors (11.c)
- The need to include containment high range radiation monitors (CHARM) as a Type C variable for fuel cladding (11)

<u>RPV Water Level as Type C Variable for BWRs</u>

The LTR conclusion that RPV water level is the appropriate Type C variable for fuel cladding is based on its use in BWR safety analysis, EPGs, and supporting engineering evaluations, consistent with the provisions of RG 1.97 R4. The response to RAI 1-I&C-11 provides the basis for the LTR conclusion which is repeated below.

Revisions 2 and 3 Type C Fuel Cladding function, lists three variables which were later determined not to be required for BWRs (radioactivity concentration, analysis of primary coolant, BWR core thermocouples). An engineering evaluation was performed to determine what should be the most direct Type C variable under the provisions of Revision 4. Based on the plant accident analysis licensing basis, design basis documentation for the fission product barriers, and the BWR EPGs, RPV water level is the best indicator of fuel cladding integrity for the BWR. Analysis and testing performed for BWR fuel confirms the relationship between RPV water level and cladding integrity. If water level is maintained above specified levels, fuel cladding integrity will be maintained. If water level drops below specified limits or is indeterminate, cladding integrity is assumed to be breached and operator action directed to restore water level and maintain core cooling.

In accordance with the BWR EPGs, the integrity of the fuel cladding barrier is determined by the status of core cooling. The fuel cladding barrier is maintained intact when the core remains adequately cooled. The fuel cladding barrier is no longer intact when adequate core cooling cannot be restored and maintained. RPV water level instrumentation is the means of determining if adequate core cooling exists.

RPV inventory decreases (whether due to a break in the RCS, SRV operation, loss of RPV injection capability, or any combination of these events), results in RPV water level decreases. When prescribed level limits are exceeded and the level is not restored in a timely manner, fuel temperatures increase causing overheating with resultant core damage. The amount of core damage is dependent on many factors such as the shutdown state of the reactor, previous power history, duration and depth of core uncovery, etc.

The magnitude of core damage (i.e., percent core damage); however, is irrelevant with respect to accident management strategies and the integrity of the fuel cladding barrier. Fuel cladding integrity either exists or it does not. This is a go-no-go decision in BWR accident management strategies and simply can be distilled to the EPG decision whether Primary Containment Flooding is required, which is the entry condition to the SAG portion of the EPGs/SAGs
There are other instrumentation and RG 1.97 R4 Type E radiation detection variables, which will be available to the operator to determine if core damage has occurred and the magnitude of the damage. This would include off-gas monitors, hydrogen monitors, containment radiation monitors, and sampling of RPV radioactivity concentration. These additional variables are used for confirmation and to assist in emergency planning but are not used to direct protection of the fission product barrier for EPGs.

Uncertainties with RPV Water Level

The RAI notes that the minimum steam cooling reactor water level (MSCRWL) is a calculated parameter based on conservative analysis assumptions and that, depending on the actual transient conditions, may be uncertain. There are uncertainties with use of RPV water level for severe accidents, but these are considered manageable and within the conservatisms assumed in the BWR EPGs. The RAI postulates that CHARM would have less uncertainties stating, "Since the containment high range monitors directly monitor the increase in radiation levels in the containment atmosphere, a direct consequence of the release of fission products from the fuel, the uncertainties are expected to be less."

Because of the conservative analysis assumptions, the EPGs permit RPV water level decreases below the MSCRWL for event-specific durations and, thus, help decrease the uncertainty of determining that the fuel cladding barrier is lost and entry to the SAGs is required. However, there are significant uncertainties associated with use of CHARM in a BWR to signal fuel cladding barrier loss, especially when steam flow continues to the main condenser and no discharge into the primary containment occurs.

The concept that primary containment radiation levels will increase because the loss of fuel cladding integrity permits the release of radioactive fission products into the reactor coolant with the consequent transportation of these fission products into the primary containment is not without uncertainties:

- If fission products are not released directly to the drywell atmosphere, which also requires a failure of the reactor coolant pressure boundary or a significant fraction of the fission products would be retained in the suppression pool and not be detected by CHARM.

- The fuel matrix will retain a significant portion of the fission products. The release of the fission products from the fuel matrix is highly dependent on the sequence of the postulated event.

- Predictions of the RPV retention times and release rates vary widely, depending on the analytical models used and the accident scenarios considered.

- The BWR pressure/suppression containment design and the use of containment sprays will result in the CHARM which is located in the drywell not fully representing the amount of radiation released.

- Some CHARMS are challenged to ensure on-scale indication during normal operation and, therefore, may not be capable of detecting low levels of coolant activity such as the dispersal of coolant activity at the Technical Specifications limit into primary containment through a break in the RCS.

- Manufacturers of CHARM quote uncertainties of 36% or more.

- Fission product releases outside the primary containment due to breaks, venting, or loss of primary containment integrity cannot be evaluated.

- While all BWR plants have installed high range containment radiation instruments capable of monitoring radiation levels in the necessary ranges, not all plants monitor both the drywell and suppression chamber volumes.

<u>Off Gas Radiation Monitors</u>

Off gas monitors should not be considered a Type C variable. Off gas monitors are identified as a normal operating system as noted in the LTR, which by definition would mean they are not Type C. The LTR should have indicated that off gas monitors, if available, will provide early indication of potential fuel cladding breech, but as noted in RAI 3-NSIR-11.c, they do not meet the criteria as a Type C variable. The LTR will be revised to remove off gas monitors in Section 4.3.4. Off Gas monitors use as a backup and supporting source of operator information is included in Section 7.2 and 7.4.3 of the LTR.

<u>CHARM as a Type C Variable</u>

Evaluations performed to develop the LTR based on RG 1.97 R4 conclude that CHARM is not a post accident variable required by the plant safety analysis, or the EPGs as a direct measurement of a radiological release. The purpose of CHARM is as described in Table 1 of RG 1.97 Rev 2 and Table 2 of RG 1.97 Rev 3, which is "detection of significant release: release assessment: long term surveillance; emergency plan actuation". It provides information used for core damage assessments and emergency action level (EAL) classifications in addition to other RG 1.97 variables.

We conclude that CHARM does not meet the criteria for being a Type A, Type B or Type C variable. CHARM is not considered a direct indicator of fuel cladding or RPV integrity and has limitations in use in a BWR. CHARM is appropriately a Type E variable.

### *RAI 3-NSIR-12*

In Section 5 on Page 5-2, the text: "...consistent with the provisions of 10 CFR 50.59 subject to plant reviews of their licensing commitments." must be revised to read: "... consistent with the provisions of 10 CFR 50.59, and 10 CFR 50.54(q), as applicable, subject to plant reviews of their current plant licensing basis." The NRC staff also requests that language be added to this section to clearly emphasize that not one size fits all and that, because of current licensing basis differences between facilities, what may be implemented under 10 CFR 50.59 and 10 CFR 50.54(q) at one facility may not be acceptable for another facility; that all such reviews shall be made against the plant's current licensing basis. Please make the requested changes or provide justification why the changes are not warranted.

### *RAI 3-NSIR-12 Response*

Multiple changes will be made to the LTR to address this request. See responses to RAI 1-I&C-23; 1-I&C-24; 3-NSIR-1; and 3-NSIR-2.

### *RAI 3-NSIR-13*

Section 4.3.4.1 identifies off gas activity as a Type C variable; yet Table 5-1 and Table 5-2 do not identify this parameter as a Type C variable. Please revise Tables 5-1 and 5-2, or Section 4.3.4.1, accordingly, or provide justification for this inconsistency.

### *RAI 3-NSIR-13 Response*

Section 4.3.4.1 will be revised to remove the reference to off gas monitor as a Type C variable consistent with the response to RAI 3-NSIR-11.

***RAI 3-NSIR-14***

Consistent with the comments in Item 11 above, Table 5-1 and Table 5-2 need to be revised to include the containment radiation level as a Type C variable.

***RAI 3-NSIR-14 Response***

The response to RAI 3-NSIR-11, provides the information to support the BWROG conclusion that containment radiation monitors are not a Type C variable for a BWR.

***RAI 3-NSIR-15***

Tables 5-1 and 5-2 identify containment radiation level as a Type E variable and indicate that EQ and SQ are not necessary. Although the NRC staff recognizes that under RG 1.97 Revision 4 and IEEE-497, the variables do need not be environmentally qualified to the requirements of RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," the monitors are required by NUREG-0737 Section II.F.1, Attachment 3, "Containment High Range Radiation Monitor," to function in the accident environment. Components of these monitoring system (e.g., detectors, cabling) would need to be qualified for the post-accident environment (a design envelope is provided in NUREG-0737) within the containment since it under these conditions that the monitors will perform their design function. Similar requirements apply to the high range noble gas effluent monitors addressed by NUREG-0737 Section II.F.1, Attachment 1, "Noble Gas Effluent Monitors." The NUREG-0737 requirements were imposed on existing licensees via a generic letter and a confirming order. They are imposed on future license applications via 10 CFR Sections 50.34(f), 52.47(a)(ii), 52.79(b), and 52.83. Please revise the EQ categorization for the subject variables to "Y*" or something similar, and explain the requirement in the text or a footnote to the tables, or provide additional justification for the proposed treatment of these variables.

***RAI 3-NSIR-15 Response***

Section 7.4.2 of the LTR will be revised to provide additional information concerning requirements for CHARM.

CHARM is considered a Type E variable as are noble gas monitors. The requirements of RG 1.97 R4 for Type E variables results in differences from those imposed in NUREG 0737 Section II.F.1, which required the highest level design requirements for CHARM (RG 1.97 R3 Category 1) and additional requirements for noble gas monitors (RG 1.97 R3 Category 2). The response to a question concerning Type E design requirements in general, is in RAI 1-I&C-2

response which is: "The qualification of Type E variables is not explicitly addressed in RG 1.97 R4 as referenced in IEEE 497. Instrument channels that monitor systems are not required to be environmentally or seismically qualified."

There are specified design requirements in RG 1.97 R4 that apply to Type E variables that would need to be met. In addition to what is included in RG 1.97 R4 for Type E, the BWROG proposes using the Design and Qualification criteria contained in RG 1.97 R3 Section 1.3.3.a "The instrumentation should be of high-quality commercial grade and should be selected to withstand the specified service environment."

Type E variables are used for monitoring the magnitude of releases; environmental conditions to determine the impact of releases; and monitoring of radiation levels in plant environs. In each case, it is expected that the conditions the monitor will function under will be established as well as the expected range so that information will be available for operator and emergency planning decisions."

The above passage does not address NUREG 0737 Section II.F.1 requirements especially with respect to CHARM. CHARM was required to be installed in BWR operating plants and plants under construction as a result of the TMI accident lessons learned, not because of reliance in BWR safety analysis or EOPs. NUREG 0737 Item II.F.1 established the requirements for such monitors and RG 1.97 R3 subsequently incorporated the monitor as a Category 1, Type E variable. Requirements include classification of RG 1.97 R3 Category 1 which is the highest level of requirements consistent with the design and quality requirements for a Basic Component as defined in 10CFR50.2, including requirement for redundancy, provisions for essential power, having extended radiation detection ranges for beyond design basis events and being environmentally and seismically qualified to such extended ranges. The LTR concludes that CHARM in a BWR does not meet the definition of a Basic Component based on the function that it provides which is for core damage assessments and emergency planning activities.

As noted, a review of CHARM use in a BWR concludes that it provides information used in post accident core damage assessments and in emergency planning. Similarities exist for appropriate design requirements for CHARM with the requirements contained in the amended combustible gas control Rule (10CFR50.44) for hydrogen monitors. While this was an amended Rule concerning combustible gas control, the impacts of revised requirements including EQ were addressed and expectations provided.

Hydrogen monitor requirements were modified as a result of NUREG 0737 Item II.F.1 and prior amendments to the combustible gas Rule resulting in determination that the monitors were needed for design basis accidents as a Basic Component (10CFR50.2) and requiring EQ. The hydrogen monitors were subsequently incorporated into RG 1.97 Revisions 2 and 3 and into post accident monitor (PAM) Technical Specifications. The amended combustible gas rule resulted in a revision to the requirements for hydrogen monitors to non-safety related commercial grade, but imposed a requirement that they be "functional" for severe accidents. Section (b)(4)(ii) of the amended combustible gas rule states:

"Equipment must be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident for accident management, including emergency planning."

Similar requirements would be expected for CHARM as they also provide information used in BWR core damage assessments for emergency planning. Noble gas monitor requirements were defined in NUREG 0737 Item II.F.1 and were incorporated into RG 1.97 R3 as Type C, Category 3 for the Containment barrier and as Type E, Category 2. Category 2 imposed environmental qualification requirements in accordance with RG 1.89. Type E is the appropriate classification for noble gas monitors which under RG 1.97 R4 will result in high quality commercial grade with the monitors designed to meet the specified service environment for radiation releases and use in emergency planning activities.

***RAI 3-NSIR-16***

The NRC staff finds the argument in Section 7.3.2 to be non-persuasive. The discussion notes that the high range containment monitors were included in RG 1.97 because of the requirements established in NUREG-0737, Section II.F.1. The requirement for the high range containment monitor arose out of the lessons-learned at TMI, where the existing monitors over-ranged or were otherwise unreliable causing difficulties for assessment of the plant status. The LTR argument then discusses some incremental changes in nomenclature between versions of RG 1.97. However, the intent of the LTR is to provide a methodology for establishing a technical basis for which parameters are required to be accident monitors; as an alternative to RG 1.97. What the argument does not provide is a basis for determining that the post-accident indications that the monitor would provide are not necessary or that they do not warrant the redundancy, quality, and Tech Specs associated with a Type C variable. The NRC staff notes that the NUREG-0737 requirements were the subject of a generic letter and the licensee responses to that generic letter were accepted by means of confirming orders. The requirements were added to 10 CFR 50.34(f) for all pending Part 50 and future licensing under Part 52 where technically relevant. The NRC staff notes that that the containment radiation monitor is an indicator of the failures of both the RCS fission product barrier AND the fuel clad fission product barrier since the monitor measures the radiation from fission products released to the containment which requires BOTH barriers to fail. The NRC staff does not dispute that parameters such as reactor water level and pressure, drywell pressure, etc., are more direct and less uncertain indicators of a RCS barrier breach. However, the same can not be said for the fuel clad barrier, for which Section 4.3.4.1 of this LTR does not provide a direct indication of breach as was discussed in Item 11above. Please provide additional justification for your position.

***RAI 3-NSIR-16 Response***

Section 7.4.2 (previously 7.3.2) will be revised to provide additional information on CHARM requirements.

The intent of the LTR is to provide a methodology for establishing a technical basis for which parameters meet the requirements to be accident monitors under RG 1.97 R4. The methodology used BWR safety analysis and critical safety function EPGs to determine the accident monitors, which has been concluded to be the appropriate process for determining such variables. Prior RG 1.97 R3 did not have the advantage of use of such processes, which resulted in the need for plant deviation requests that often were based on reviews of plant safety analysis requests and required use in EOPs for individual variables. The LTR methodology provides a comprehensive review of all accident monitoring and has identified additional deviation/changes as a result.

As discussed in RAI 3-NSIR-15, CHARM provides a necessary function as a post accident Type E radiation monitor in a BWR but should have the design and qualification requirements consistent with the function it provides. The requirements imposed by NUREG 0737 Item II.F.1 for CHARM as captured in RG 1.97 R3 are an unnecessary burden based on the conclusion that they are not relied upon in BWR accident analysis, safety function EPGs or relied upon to indicate the breech of a fission product barrier. NUREG 0737 requirements imposed on CHARM are consistent with the requirements for design and quality for Basic Components as defined in 10CFR 50.2 and for RG 1.97 Type A. B, and C variables. CHARM does not meet the definition of a Basic Component nor does it satisfy the definition of a Type A, B, or C variable under RG 1.97. A preliminary review of CHARM functions does not support the need for inclusion in Technical Specifications consistent with the criteria in 10 CFR 50.36.

The RAI references confirmatory orders. Not all currently operating BWRs received confirmatory orders for NUREG 0737. Only those plants that had an operating license at the time of the NUREG received confirmatory orders. The RAI also notes that requirements were added to 10 CFR 50.34(f) for all pending Part 50 and future licensing under Part 52 where technically relevant. The LTR is based on currently operating BWRs, which are not impacted by 50.34 (f) or 10CFR Part 52.

The RAI indicates that CHARM would provide information on the failure of two barriers (fuel cladding and RPV). As noted, in the response to RAI 3-NSIR-11, CHARM has limitations on its use as a direct indicator of a breech of either fuel cladding or the RPV. The response to RAI 3-NSIR-11 also addresses use of RPV water level as the best indicator of breech of fuel cladding in a BWR.

### *RAI 3-NSIR-17*

In Table A-1 on Page A-5, with regard to the "primary containment area radiation" entry under Type C variables, the note specifies that "Not relied on in accident analysis or EPGs for breach of barrier. Only function is for EALs." The staff notes that the emergency action levels (EALs) are used to initiate the site's emergency response plan that provide for the protection of the public in those rare circumstances in which engineered design features and human capacity to take corrective actions have both failed to avert a serious mishap. The view that emergency planning is secondary to engineered design features and safe siting was refuted by the unexpected sequence of events that occurred at TMI. The Commission emphasizes the integration of safety, security, and emergency preparedness as the basis for the NRC's primary mission of protecting public health and safety. The Type C variables, which monitor fission product barriers, are particularly significant in that a General Emergency, the level at which public protective actions are necessary, is defined as the loss of two fission product barriers and a potential loss of the third barrier. As such, the staff does not find an argument based on "only function is for EALs" to be particularly persuasive as a justification for reducing the treatment requirements for the containment radiation instrumentation addressed in EALs. Please provide additional justification for your position.

### *RAI 3-NSIR-17 Response*

The comment in Table A-1 indicates the extent to which CHARM is used at a BWR to provide supporting information for EAL determination and to calculate the extent of fuel damage. See responses to RAIs 3-NSIR-11 and 3-NSIR-16 for additional information concerning CHARM use in BWRs.

BWR Emergency Response Organizations (EROs) use CHARM with other variable indications to perform Emergency Planning functions such as dose assessment, core damage assessment, and evaluation of EAL thresholds:

- Dose assessment is a tool for prediction of the possible exposure to the public and plant personnel if radioactivity is discharged outside the primary containment.

- Like containment hydrogen concentrations, CHARM may be used in core damage assessment methodology to correlate readings to the percentage of core damage assumed to exist. The uncertainties of this correlation not withstanding, once any core damage is detected, fuel cladding integrity does not exist irrespective of the core damage percentage. RPV water level with respect to EPG/SAG limits provides the best indication of prevention and mitigation of fuel cladding barrier failure. If containment hydrogen concentration or elevated CHARM readings first detect loss of fuel cladding integrity, the ERO has decided that RPV water level cannot be determined with the instrumentation or has failed to properly monitor the RPV water level instrumentation.

- All BWRs have implemented EAL schemes based on NRC approved guidance. Elevated containment radiation monitor readings are specified in power operation, startup/hot standby and hot shutdown modes for determining: the loss of the fuel clad barrier, the loss of the RCS barrier, and the potential loss of the primary containment barrier.

    - The fuel clad barrier loss CHARM threshold corresponds to a coolant activity level of approximately 300 μCi/gm dose equivalent I-131 uniformly dispersed into the containment atmosphere.

    - The RCS barrier loss CHARM threshold is the lowest of the three thresholds and corresponds to coolant activity at the Technical Specifications limit uniformly dispersed into the containment atmosphere.

    - The primary containment barrier potential loss CHARM threshold is the highest of the three thresholds and corresponds to approximately 20% fuel damage.

Other variables and instruments are also used to detect the loss or potential loss of the three EAL fission product barriers, most notably, RPV water level instrumentation. The latest EAL guidance requires a General Emergency to be declared when primary containment flooding is required. This represents a loss of the fuel cladding barrier because RPV water level cannot be restored and maintained above limits. It is a loss of the RCS barrier because RPV water level cannot be restored and maintained which is a potential loss of the primary containment. The primary indication of fuel cladding integrity (RPV water level instrumentation), therefore, ensures the General Emergency classification is recognized regardless of the status of CHARM readings. For any event leading to core uncovery and fuel damage, the recognition of the need for a General Emergency declaration should always occur through monitoring RPV water level long before the CHARM readings reach the equivalent of 20% fuel damage. An ERO that makes this declaration from CHARM readings has determined that RPV water level cannot be monitored with the instrumentation or has failed to properly monitor the RPV water level instrumentation.

Furthermore, CHARM readings can produce questionable indication of a fission product barrier loss. For example, with the actual RCS barrier intact, it is possible for an elevated CHARM reading to exceed the RCS barrier loss threshold simply because the amount of coolant activity in the RCS is significantly higher than the Technical Specifications coolant activity limit assumed for the RCS barrier loss threshold. EALs has even acknowledged this by cautioning the ERO as follows:

"...it is important to recognize that in the event the radiation monitor is sensitive to shine from the reactor vessel or piping, spurious readings will be present and another indicator of fuel clad damage is necessary or compensated for in the threshold value."

Site Area Emergency and General Emergency EALs are applicable only in the cold shutdown and refueling mode rely on CHARM readings as an indirect indication of loss of inventory in the RPV. Thus, for all EAL classification purposes, CHARM readings serve only as a backup to other instruments and, if RPV water level instrumentation functions and is properly monitored, CHARM readings would not be the source for making EAL classifications.

The most direct indication of the integrity of the fuel cladding barrier, therefore, is the variable that detects the presence of adequate core cooling. In the EPG portion of the EPGs/SAGs, this variable is RPV water level. BWR RPV water level instrumentation is highly redundant, reliable and capable of detecting water level changes in the RPV well above and below the normal operating limits.

## Section 4: Reactor Systems Branch RAI

### *RAI 4-RS-1*

The LTR identifies the requirements of IEEE-497 with respect to the five types of accident monitoring system variables. The five types of accident monitoring system variables are defined as follows: Type A variables provide the operators with the primary information necessary to take the normal actions credited in the safety analysis; Type B variables provide primary information to the control room operators to assess the plant critical safety functions; Type C variables provide extended range primary information to the control room operators to indicate the potential breach or the actual breach of the fission product barriers; Type D variables provide information to the control room operators to indicate both the performance of those required systems and auxiliary supporting features necessary for the mitigation of AOOs and accidents and the performance of other system necessary to achieve and maintain a safe shutdown condition, and to verify system status; and Type E variables provide information to be used in determining the magnitude of the release of radioactive material and continually assessing such releases. Safety analysis events including AOOs and accidents are given in Table 4-1, systems assumed in the safety analysis including events, required action, and system assumed are given in Table 4-2, and required system, shutdown systems, and auxiliary support systems are given in Table 4-3. It requires that approved methodologies should be used to analyze safety analysis events identified in Table 4-1 and to apply the results of the analysis to supporting the required action using system assumed for safety analysis events listed in the Table 4-2. Please provide: (1) approved methodologies used to analyze the safety analysis events listed in Table 4-1; (2) identification of parameters and its acceptable criterion to be used for required action taken by operators using system assumed; (3) description of an updated version of BWR EPGs and the Severe Accident Guidelines (SAGs) used to perform the critical safety functions; and (4) description of an updated BWR EOPs to support shutdown systems.

### *RAI 4-RS-1 Response*

(1). The LTR is based on NRC approved methodologies used in the safety analysis process for the events listed in Table 4-1 and Table 4-4, including but not limited to Licensing Topical Report "General Electric Standard Application for Reactor Fuel ", NEDO-24011-A-14 (GESTAR II) and various prior revisions and plant updated FSAR Chapters 14 and 15, which provide the current safety analysis.

(2) The need for operator manual actions is identified in the LTR methodology used to determine Type A variables based on safety analysis. Acceptance criteria for such actions are beyond the scope and intent of the LTR. Operator actions are further described in the updated FSAR Chapters 14 and 15.

(3) NRC approval for EPGs was provided in the review of NEDO-31331 dated March 1987, which has been used in the LTR. The LTR uses BWR symptom based EPGs to determine critical safety functions for BWRs. The EPGs contained in NEDO-31331 included contingency procedures one of which was primary containment flooding. As part of severe accident management initiatives, severe accident guidelines (SAGs) were developed based on the NEDO-31331 primary containment flooding contingency procedure. NRC has reviewed and provided comments on subsequent revisions to NEDO-31331 including EPG/SAGs.

(4) All BWRs provide plant specific EOPs based on EPGs, including systems required for safe shutdown. The LTR is not based on EOPs but on EPGs. Plant specific use of the LTR would require use of the plant specific EOPs.

## Section 5

### Revision of NEDO-33349, Section 7 for Reconciliation of Type B and Type C Variables with RG 1.97 R3 Design Requirements

The purpose of this section is to reconcile the results of this LTR methodology using RG 1.97 R4 with the prescriptive list in RG 1.97 R3 for BWR variables. Note that RG 1.97 R3 list of variables has not been updated since its release, so the list does not reflect all the generic deviations and other agreements which conclude that the variable is not required for the BWR design. Table A-1 lists Table 1 of Rev. 2 and Table 2 of Rev. 3 by function and shows the result of applying this LTR methodology to define Type variables for a BWR/4 design. Similar results would be shown for a BWR/6 design. Also included for information in Table A-1 is a BWR plant's depiction of their RG 1.97 R2 commitments (the plant has a licensing commitment to Rev. 2). While R4 does not list the functions included in Rev. 2, all the functions are addressed by the Rev. 4 process.

Type B Functions

The Rev. 2 and Rev. 3 Type B functions of Reactivity Control, Core Cooling and Maintain Reactor Coolant System Integrity are met by the BWR EPGs for Reactor Control which includes integrated procedures for reactivity control, reactor pressure control and reactor water level control. For Reactivity Control, Rev. 2 and Rev. 3 include neutron flux which is also identified in the LTR using Rev. 4. The other two Reactivity Control variables listed in Rev. 2 and Rev. 3 as Type B are control rod position which this LTR concludes is a Type D variable and RCS Soluble Boron concentration which is not a BWR required parameter.

The Rev. 2 and Rev. 3 Core Cooling Function, Coolant Level in the Reactor is met by RPV level control using Rev. 4, and the Rev. 2 and Rev. 3 variable of BWR core thermocouple was eliminated for BWRs as a generically approved deviation.

The Rev. 2 and Rev. 3 Maintaining Reactor Coolant System Integrity variables of RCS (RPV) pressure is identified using Rev. 4 as is drywell pressure while drywell sump level in Rev. 2 and Rev. 3 is addressed as a deviation with conditions imposed that most if not all BWRs meet. Drywell sump level is a normal operating system in a BWR with the drywell sump valves being isolated as part of containment isolation. Drywell pressure is identified as a Type B variable under Rev. 2 and Rev. 3. The LTR lists drywell pressure as a Type B variable under containment control but it could also be shown as a Type B under the Rev. 2 and Rev. 3 Reactor Coolant System integrity function.

The Rev. 2 and Rev. 3 Maintaining Containment Integrity Function is met by the Primary Containment Control EPG for Rev. 4. Primary containment pressure is listed in Rev. 2 and Rev. 3 and also using Rev. 4. Primary Containment Isolation

Valve (CIV) position indication is listed in Rev. 2 and Rev. 3, as Type B and as Type D using Rev. 4. CIV position indication is further discussed in Section 7.4.1. Additionally not included in Rev. 2 and Rev. 3 but included using Rev. 4 as Type B are suppression pool temperature and suppression pool water level both of which are monitored to ensure the BWR containment safety function is being maintained throughout design basis accidents. Operator actions are described in the EPGs if suppression pool temperature or level exceeds specified limits based on prescribed heat capacity temperature or SRV discharge limits, which are based on post event containment load and emergency core cooling system pump requirements. Rev. 2 and Rev. 3 lists suppression pool temperature as Type D and suppression pool water level as Type C and Type D variables.

Type C Functions

Rev. 2 and Rev. 3 Type C Fuel Cladding function lists three variables which were later determined to not be required for BWRs (radioactivity concentration, analysis of primary coolant, BWR core thermocouples). An engineering evaluation was performed to determine what should be the most direct Type C variable under the provisions of Rev. 4. Based on the plant accident analysis licensing basis, design basis documentation for the fission product barriers, and the BWR EPGs, RPV water level is the best indicator of fuel cladding integrity for the BWR. Analysis and testing performed for BWR fuel confirms the relationship between RPV water level and cladding integrity. If water level is maintained above specified levels, fuel cladding integrity will be maintained. If water level drops below specified limits or is indeterminate, cladding integrity is assumed to be breached and operator action directed to restore water level and maintain core cooling.

In accordance with the BWR EPGs, the integrity of the fuel cladding barrier is determined by the status of core cooling. The fuel cladding barrier is protected when the core remains adequately cooled. The fuel cladding barrier is no longer intact when adequate core cooling cannot be restored and maintained. RPV water level instrumentation is the primary means of determining if adequate core cooling exists.

RPV inventory decreases (whether due to a break in the RCS, SRV operation, loss of RPV injection capability, or any combination of these events), results in RPV water level decreases. When prescribed level limits are exceeded and the level is not restored in a timely manner, fuel temperatures increase causing overheating with resultant core damage. The amount of core damage is dependent on many factors such as the shutdown state of the reactor, previous power history, duration and depth of core uncovery, etc.

The magnitude of core damage (i.e., percent core damage); however, is irrelevant with respect to accident management strategies and the integrity of the fuel cladding barrier. Fuel cladding integrity either exists or it does not. This is a go-no-go decision in BWR accident management strategies and simply can be distilled to the EPG decision whether Primary Containment Flooding is required, which is the entry condition to the SAG portion of the EPGs/SAGs

There are other instrumentation and RG 1.97 R4 Type E radiation detection variables, which will be available to the operator to determine if core damage has occurred and the magnitude of the damage. This would include offgas monitors, steam line radiation monitors, hydrogen monitors, containment radiation monitors, and sampling of RPV radioactivity concentration. These additional variables are used for confirmation and to assist in emergency planning but are not used to direct protection of the fission product barrier for EPGs.

Rev. 2 and Rev. 3 Type C function of reactor coolant pressure boundary lists as variables RCS pressure, primary containment area radiation (listed as Category 3), drywell drain sump level, suppression pool water level and drywell pressure. Using the engineering evaluation process described in Section 4.3 for reactor coolant pressure boundary, the Type C variables are RPV pressure, RPV water level, drywell pressure, suppression pool water level and suppression pool temperature. This list of variables is needed to address all potential breaches of reactor coolant pressure boundary, including small and large pipe breaks and open SRVs, which discharge into the suppression pool resulting in increased suppression pool temperature. Primary containment radiation may be an indicator of radiation release from fuel cladding breach and reactor coolant pressure boundary breach, but it is not a direct indicator or meets the IEEE 497 definition of a less direct variable supported by analysis as a substitute for the listed variables. Primary containment radiation is considered a Type E variable. Drywell drain sump is isolated on a loss of coolant accident. The drywell sump level indication is not used for other than normal operation to determine potential degradation in the reactor coolant pressure boundary, so that repair can be made prior to any potential failure. Drywell sump and drywell drain sump level is an NRC approved deviation with conditions imposed, which most BWRs implemented as a plant specific deviations.

Rev. 2 and Rev. 3 Type C Containment function lists several variables which are included as NRC approved deviations (hydrogen and oxygen monitors), and variables which have been determined to be Type E (containment effluent radioactivity and radiation exposure rate effluent radioactivity) as well as RCS Pressure and Primary Containment Pressure. It has been concluded that the Type C variables that comply with Rev. 4 are drywell/containment pressure, suppression pool level, and suppression pool temperature. Suppression pool level and temperature have been include as required variables, as these must be maintained within established limits to support containment integrity design requirements, including containment hydrodynamic load assumptions.

## Section 6

## Revision of NEDO-33349, Section 7.3.1 for Additional Information on Primary Containment Isolation Valve Position Indication as Type D Variable

Primary containment isolation valve (CIV) position indication is included in RG 1.97 Rev. 2 and Rev. 3 as a Type B, Category 1 variable under the function Maintaining Containment Integrity. RG 1.97 R4 defines Type B variables as those variables which provide primary information to the control room operators to assess plant critical safety features which includes Primary Containment Integrity. The LTR based on RG 1.97 R4 uses the BWR EPGs to determine the Type B variables including the Primary Containment Control EPG, which addresses post accident Containment Integrity. Section 4.2.2 contains the results of the LTR evaluation, which identifies drywell/containment pressure, suppression pool level and suppression pool temperature as the Type B variables for Containment Integrity. Section 4.4.5 discusses isolation valve position indication and requirements for RPV and primary containment (RPV&PC) isolation valves, which have different requirements and have been evaluated to be a Type D variable. Valve position indication is used to verify system safety status by confirmation that the safety systems have functioned as designed. The required system isolation requirements are fulfilled by the redundant RPV&PC isolation valves, which provide isolation of the RPV and the primary containment as required. The RPV&PC isolation system provides safety related isolation signals to each of the RPV&PC isolation valves. The RPV&PC isolation system (RPV&PCIS) is designed to provide automatic isolation when required.

The RPV&PC isolation valves are required to ensure Containment Integrity both prior to and post accident. The primary information the control room operator relies upon post accident is drywell/containment pressure with suppression pool temperature and suppression pool level needed to ensure containment integrity is maintained throughout the accident. The RPV and containment isolation valves are safety systems designed to meet single failure criteria and to align properly to support containment integrity post accident. The RPV&PC isolation valves and the isolation signals are included in BWR improved Standard Technical Specifications for containment (Section 3.3.6.1 and 3.6.1.3 of the improved Standard Technical Specifications). RPV&PC isolation valve position indication is included in post accident monitoring (PAM) Standard Technical Specifications because it is listed in RG 1.97 R3 as a non-Type A Category 1 variable.

Additional requirements for containment design were published in NUREG-0737, "Clarification of TMI Action Plan Requirements", Section II.E.4.2, "Containment Isolation Dependability". NUREG-0737 Section II.E.4.2 does not provide additional requirements for containment isolation valve position indication. Additional requirements are that each non-essential penetration (except instrument lines) is required to meet post-accident isolation requirements specified by SRP, Section 6.2.4. Isolation must be performed automatically (i.e., no credit can be given for operator action). Each automatic isolation valve in a nonessential penetration must receive the diverse isolation signals. The General Design Criteria (GDC) establishes requirements for isolation barriers in lines penetrating the primary containment boundary. In general, two isolation barriers in series are required to ensure that the isolation function is satisfied assuming any single active failure in the containment isolation provisions. The operability of the RPV&PC isolation valves ensures that the primary containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the primary containment atmosphere or pressurization of the containment consistent with the assumptions used in the analyses for a postulated loss of coolant accident (LOCA). RPV&PC isolation valves are automatically initiated for a postulated LOCA.

The RPV&PCIS is designed to prevent the inadvertent opening of an isolation valve when closed by an initiating signal. The position indication from each valve is monitored in the control room by status lights. The position of an isolation valve for normal and shutdown plant operating conditions and post-accident conditions depends on the fluid system function. If a fluid system does not have a post-accident function, the isolation valves in the lines will be automatically closed and not reopened.

Valve position indication is used as verification of the containment system status and to indicate that the RPV&PC isolation valves have performed their safety system function of containment isolation. This is the definition of a Type D variable in RG 1.97 R4. There is a similar definition in Rev. 2 and Rev. 3. A typical BWR will have approximately 40 containment penetrations with automatic isolation valves and thus up to 80 CIV position indication systems currently listed as RG 1.97 R3 Type B variables. In the unlikely event that a containment penetration would not meet its design function to be isolated post accident, the operator will have the redundant RPV&PC isolation valve position indication and other plant indications to indicate that the containment has not been isolated.