BWR OWNERS' GROUP

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Project Number 691

BWROG-09065
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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Responses to Supplemental Request for Additional Information (RAIs) Dated June 16, 2009, Regarding Boiling Water Reactor Owners' Group Licensing Topical Report (LTR) NEDO-33349, Revision 1, "BWR Application to Regulatory Guide 1.97, Revision 4," (TAC No. MD6697)

ENCLOSURE: Responses to SRAI

ATTACHMENTS:
1. BWR Owners' Group Response to NRC Supplemental Requests for Additional Information
2. BWR Owners' Group Proposed Revision to NEDO-33349
3. SRAI Summary and NEDO-33349 R1 Mark-up Changes

REFERENCES:
1. BWR Owners' Group Letter From Randy 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
4. Letter Dated June 16, 2009 from NRC Containing Supplemental Request For Additional Information

Please find the BWROG responses (enclosure) to the NRC Supplemental Request for Additional Information on the subject Licensing Topical Report NEDO-33349, Revision 1. NRC provided the SRAIs for this report by letter dated June 16, 2009, and we now submit responses to the aforementioned SRAIs which address questions raised during continued NRC review of NEDO-33349, Revision 1.

The Enclosure contains three (3) Attachments.

Attachment 1 contains SRAI responses organized into seventeen (17) enumerated question & answer pairs.

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Attachment 2 of the Enclosure is a NEDO-33349 mark-up describing proposed changes to the LTR based on BWROG response to the respective SRAI. Attachment 2 utilizes as a base document, the LTR version submitted to the Commission on October 31, 2008 (also Attachment 2), titled, *BWR Owners' Group RG 1.97 Committee Response to NRC's Request for additional Information Regarding Licensing Topical Report NEDO- 33349, Revision 1 "BWR Application to Regulatory Guide 1.97 Revision 4."* Upon receipt of a Safety Evaluation (SE) issued by the Commission, NEDO-33349 will be revised and released as an approved LTR.

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

Reference 1 is associated with the BWROG request for NRC acceptance of NEDO-33349, Revision 1. References 2 and 3 are associated with the Requests for Additional Information (RAI) related to NEDO-33349, Revision 1. Reference 4 is associated with the Supplemental Request for Additional Information (SRAI) related to NEDO-33349, Revision 1.

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

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ATTACHMENT 1
BWR Owners' Group Response to NRC
Supplemental Requests for Additional Information (SRAI)

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.

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

BWROG RG 1.97 Committee
Response to NRC SRAI dated June 16, 2009 - NEDO-33349 Revision 1

The following is the BWROG response to the Supplemental Request for Additional Information (SRAI) letter from the NRC dated June 18, 2009 concerning review of NEDO-33349 Revision 1 "BWR Application to Regulatory Guide 1.97 Revision 4". The SRAI topics are enumerated followed by the BWROG response.

NEDO-33349 was prepared to identify a methodology that can be used to comply with RG 1.97 Revision 4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," and Institute of Electrical and Electronics Engineers Std 497-2002 (IEEE-497), "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations." The methodology is based on source documents included in IEEE-497 (i.e. generic BWR Safety Analysis methodology consistent with the typical BWR plant "License Bases;" and the application of generic symptom based Emergency Procedure Guidelines - EPG).

SRAI 3 and 17 address NEDO-33349 conclusions related to Type C variables and their role in monitoring fuel cladding integrity. Reactor Pressure Vessel (RPV) Water Level indication, is a key parameter controlled by station operators to ensure cladding integrity is maintained through core submergence. The BWROG does not claim that RPV water level solely indicates that fuel barrier failure has occurred. However, appropriate actions related to fuel barrier integrity, are taken based upon RPV water level indications.

Other backup variables used by Licensed Operators to assess fuel cladding fission product barrier integrity, include the Containment High-Range Radiation Monitor System (CHARMS). Although backup variables discussed in SRAI 3 are utilized to assess extent of core damage for emergency planning purposes, they are not used as primary information sources that drive Licensed Operator actions associated with Emergency Operating Procedures (EOP).

With respect to Type E variables, the BWROG proposes that CHARMS-related requirements contained in RG 1.97 Revisions 2 and 3, be retained as exceptions to provisions of RG 1.97 Revision 4. NEDO-33349 will require CHARMS to meet all requirements of RG 1.97 Revision 2 or 3, Type E, Category 1; and be consistent with plant specific licensee commitments to requirements established in NUREG 0737 II.F.1.

SRAI 1

RG 1.97 Revision 3 recommends Drywell Pressure as a Type B Category 1 variable to monitor the Maintaining Reactor Coolant System Integrity function. NEDO-33349 uses the Pressure Control function instead of the RG 1.97 Revision 3 Maintaining Reactor Coolant System Integrity function and credits Reactor Pressure as a key variable for the Pressure Control function. However, based on the response to RAI 1-I&C-12 it is not clear if Drywell Pressure is or is not being credited, in NEDO-33349, as a key variable for the Pressure Control function.
The reason for this confusion is that the response indicates that Reactor Pressure is included in the Reactor Control Emergency Procedure Guidelines (EPGs) to support Reactor Pressure Vessel (RPV) integrity. The response also indicates that Drywell Pressure is an entry condition for the RPV Control EPG. Please clarify if NEDO-33349 credits Drywell Pressure as a key variable for providing information about the accomplishment of the Pressure Control function.

<u>SRAI 1 Response</u>

Drywell Pressure is not credited as a "key" variable for providing information about the accomplishment of the Pressure Control function. Potential confusion exists due to differences between BWR Emergency Procedure Guidelines (EPG), and variables lists contained in RG 1.97 Revisions 2 and 3.

Type B variables identified in NEDO-33349 are associated with symptom-based EPG. The EPG are organized into four top-level guidelines used to develop the Critical Safety Functions defined in RG 1.97 Revision 4 for Type B variables (the RPV Control functional category requires continuous entry condition assessment for Reactivity, Pressure, and Level guidelines once the procedure is entered):

- Reactivity Control (RPV Control)
- Pressure Control (RPV Control)
- Level Control (RPV Control)
- Primary Containment Control

RG 1.97 Revision 2 and 3 functions (Reactivity Control, Core Cooling, and Maintaining Reactor Coolant System Integrity), are satisfied by the EPG RPV Control guideline which integrates RG 1.97 functions into procedures for Reactivity Control, Pressure Control and Level Control. The result is a common list of Type B variables in NEDO-33349, which do not strictly align with the prescriptive lists of RG 1.97 Revisions 2 and 3. Thus, the Maintaining Reactor Coolant Integrity function of RG 1.97 Revision 2 and 3 questioned in the SRAI, cannot be strictly aligned with the integrated EPG. The EPG for Reactivity

Control, Pressure Control and Level Control, are utilized to control Reactor Coolant System integrity during accident conditions.
The "Maintaining Containment Integrity" function is addressed by the Primary Containment Control EPG, which is employed to monitor post-accident Drywell leakage. Drywell Pressure is identified in NEDO-33349, as a Type B variable for Primary Containment Control - also listed as an entry condition for RPV Control, to address parameters related to RPV inventory control.

Control of Drywell Pressure is not significant in the operator actions associated with Pressure Control. The only action taken with respect to drywell pressure in the Pressure Control Guideline is as a precursor for the operator to prevent injection for those low pressure emergency core cooling system pumps not required to assure adequate core cooling prior to depressurizing below their maximum injection pressures if a high drywell pressure signal exists. This action is to prevent an unnecessary system injection. Therefore, drywell pressure is not considered necessary for the Pressure Control function.

In accordance with NEDO-33349 – based upon RG 1.97 Revision 4 – Drywell Pressure is categorized as a Type A, B, and C variable.

SRAI 2

Regulatory Guide (RG) 1.97 Revision 3 recommends Drywell Sump Level as a Type B Category 1 variable to monitor the Maintaining Reactor Coolant System Integrity function and Drywell Drain Sump Level as a Type C Category 1 variable to monitor the Reactor Coolant Pressure Boundary fission product barrier. NEDO-33349 uses the Pressure Control function instead of the RG 1.97 Revision 3 Maintaining Reactor Coolant System Integrity function and credits Reactor Pressure as a key variable for the Pressure Control function and Reactor Water Level, Reactor Pressure, Suppression Pool Temperature, and Suppression Pool Water Level as key variables for the Reactor Coolant Pressure Boundary fission product barrier.
Previous generic acceptance of the use of RG 1.97 Revision 3 alternate Category 3 instrumentation, in lieu of Category 1 Drywell Sump Level instrumentation, was based on (a) for small leaks the alternate instrumentation will not experience a harsh environment; and (b) for large leaks, the sumps fill promptly and the sump drain lines isolate due to the increase in drywell pressure, negating the need for the measurement; and (c) drywell pressure and temperature indication can be used to detect leakage into the drywell; and (d) the instrumentation neither automatically initiates nor alerts the operator to initiate operation of a safety system in a post-accident situation.
Please discuss the applicability of the above acceptance conditions in light of the fact that under NEDO-33349 neither Drywell Pressure nor Drywell

Temperature is credited as a key variable for the Pressure Control function and Drywell Temperature does not meet the design and qualification criteria for a RG 1.97 Revision 4 Type B or Type C variable.
Please provide additional information on the NEDO-33349 claim that Drywell Sump Level is not relied on in safety analysis or EPGs for small or large leaks.

SRAI 2 Response

Drywell Sump Level and Drywell Drain Sump Level are not relied upon to address small or large leaks, in either the BWR Safety Analysis or EPG. They are used during normal plant operation, in contribution to the overall leak detection system requirements, and are not considered to be either Type B or Type C variables with respect to post-accident monitoring.

Drywell Temperature is a Type D variable for post-accident monitoring, and is available during normal plant operation when the Drywell Sump System instrumentation is operable.

As stated in SRAI 2 and as part of the resolution of RG 1.97 Revision 2 and 3 licensing commitments, BWR Owners' pursued NRC acceptance of alternate Category 3 in lieu of Category 1 Drywell Sump Instrumentation. The NRC granted the use of Category 3 based on the listed acceptance conditions including the last condition (d) which states the instrumentation neither automatically initiates nor alerts the operator to initiate operation of a safety system in a post-accident situation. Although the level of the drain sumps can be an indication of a breach of the Reactor Coolant Pressure Boundary (RCPB), the indication is not unambiguous because there can be water in those sumps during normal plant operation. This is the reason that the Drywell Sump System is considered part of the normally operating leakage detection system controlled by plant Technical Specifications.

RG 1.97 Revision 4, addresses instrumentation to function during and after an accident. By design, Drywell Sump system penetrations are isolated during accident conditions, via the Primary Containment Isolation System (PCIS), thereby contributing to the establishment of Primary Containment integrity. With the system tripped / isolated, post-isolation Drywell Sump level indication cannot be obtained.

NEDO-33349 concludes for generic BWRs, that Drywell Sump Level is not relied upon in either Safety Analysis or EPGs, for the assessment of post-accident conditions.

Concerning previous acceptance condition (c), Drywell Pressure and Drywell Temperature are identified as instruments required by RG 1.97 Revision 4.

Designated as safety related instrumentation, they function to assist Licensed Operators in determining RCPB leakage (as measured by the Drywell Sump systems) during normal operation. A plant shutdown may be required by plant Technical Specifications for unidentified RCPB leakage, based on adverse Drywell Pressure / Drywell Temperature trends.

SRAI 1 discusses the differences between the BWR EPG and RG 1.97 Revision 3, for variables that monitor the Maintaining Reactor Coolant System Integrity Function. The RPV Control EPGs (Reactivity Control, Pressure Control, and Level Control), are needed to maintain Reactor Coolant System Integrity during accidents conditions. Thus, the list of required Type B variables would include: Neutron Monitoring, Reactor Water Level, Reactor Pressure, Suppression Pool Water Level and Suppression Pool Temperature. These variables (with exception to Neutron Monitoring) are also identified as Type C for the Reactor Coolant Pressure Boundary.

Drywell Pressure is categorized as a Type A, B, and C variable based on BWR EPG. It is not considered a key variable for Pressure Control, but is considered a key variable for Primary Containment Control.

The "Maintaining Containment Integrity" function is addressed by the Primary Containment Control EPG, which is employed to monitor post-accident Drywell leakage. Drywell Pressure is identified in NEDO-33349, as a Type B variable for Primary Containment Control - also listed as an entry condition for RPV Control, to address parameters related to RPV inventory control.

Drywell Temperature instrumentation is used during operation and for post-accident monitoring. NEDO-33349 concludes Drywell Temperature is also a post-accident Type D variable, used to monitor post-accident Primary Containment performance, but is not relied upon as a Type B or C variable.

SRAI 3

NEDO-33349 credits Reactor Water Level as the key RG 1.97 Revision 4 Type C variable to provide information on the Fuel Cladding fission product barrier. The justification provided in NEDO-33349 is that Reactor Water Level is the parameter that provides the most direct indication of the fuel cladding fission product barrier integrity; the integrity of the fuel cladding barrier is maintained intact when the core remains adequately cooled by water in the reactor; and a breach of the fuel cladding barrier is assumed when adequate core cooling cannot be restored or has not been maintained. Responses to RAI 1-I&C-13, RAI 3-NSIR-11, and RAI 3-NSIR-14 discuss the use of Reactor Water Level as the key variable to provide information on the Fuel Cladding fission product barrier.

The NRC staff questions Reactor Water Level as the key variable for the Fuel Cladding fission product barrier. Reactor Water Level is a precursor for fuel damage but is not the most direct variable to provide information on the potential for breach or the actual breach of the Fuel Cladding fission product barrier. The arguments in the RAI responses have not addressed the technical issues raised by the RAIs.

The result of a Fuel Cladding fission product barrier failure would be an increase in containment radiation. The NRC staff has previously suggested that existing RG 1.97 Revision 3 Type E Category 1 Containment Area Radiation instrumentation could be used as the key variable to provide information on the Fuel Cladding fission product barrier.

Please discuss what variables would be credited as key variables for the Fuel Cladding fission product barrier, since the NRC staff has not been convinced that Reactor Water Level can fulfill this purpose.

SRAI 3 Response

Consistent with RG 1.97 Revision 4 and IEEE-497, the BWROG has determined the variable that provides the most direct indication of the potential for breach of the fuel cladding fission product barrier, is Reactor Pressure Vessel (RPV) Water Level; and the Containment High-Range Radiation Monitor System (CHARMS) does not meet the criteria for a Type C variable.

As there are differences with respect to the classification of CHARMS, the BWROG proposes the existing requirements contained in RG 1.97 Revisions 2 and 3, be retained as an exception to the provisions of RG 1.97 Revision 4 for Type E variables. NEDO-33349 will include narrative that CHARMS meet all requirements of RG 1.97 Revision 2 or 3 Type E Category 1, and be consistent with plant specific licensee commitments to requirements established in NUREG 0737 II.F.1. This issue will be further discussed in the response to SRAI 17.

RG 1.97 Revision 4 endorses the use of IEEE Std 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," as an acceptable method for providing instrumentation to monitor variables for accident conditions subject to regulatory positions. IEEE-497 establishes flexible, performance- based criteria for the selection, performance, design, qualification, display, and quality assurance of accident monitoring variables. These variables are the Licensed Operators' primary source of accident monitoring information. In some instances, additional variables providing backup or diagnostic information may exist. Additional variables – not considered to be primary sources of information – need not be classified or aligned with tenets of IEEE-497 or RG 1.97 Revision 4.

NEDO-33349 includes implementation recommendations for existing plant equipment and prospective plant modifications, based on the provisions of RG 1.97 Revision 4. This methodology is intended to be applicable to all operating BWR plants. The methodology is based on source documents included in IEEE-497 (i.e. generic BWR Safety Analysis methodology consistent with the typical BWR plant "License Bases;" and the application of generic symptom based Emergency Procedure Guidelines - EPG).

It is requested that NRC approval of NEDO-33349 be an approval of the methodology described. However, for the implementation of the methodology, a plant specific license amendment application is required, along with the applicable proposed Technical Specification changes. The license amendment must identify the plant specific accident monitoring variables consistent with the application of the NEDO-33349 methodology and address all of the related plant License Bases, consistent with the requirements of 10CFR50.59. In addition, NEDO-33349 will be changed to require that the plant specific implementation include consideration of the post-accident monitoring and License Bases requirements, including (but not limited to) items included in NUREG-0737, "Clarification of TMI Action Plan Requirements," and RG 1.97 Revisions 2 or 3.

NEDO-33349 demonstrates application of its underlying methodology, by providing a typical list of accident monitoring variables complete with supporting data (variable type, classification basis, design requirements, and qualification requirements). With some exceptions, resultant NEDO-33349 variables differ from those identified in RG 1.97 Revisions 2 and 3 - these differences are due to the application of specific criteria identified in IEEE-497. Although the overall objective of providing an acceptable set of accident monitor instrumentation is achieved, it is not intended that this list of variables be implemented without plant specific evaluations consistent with the NEDO-33349 methodology.

Consistent with RG 1.97 Revision 4 and IEEE-497, the BWROG has determined that the variable that provides the most direct indication of the potential for breach of the fuel cladding fission product barrier is Reactor Pressure Vessel (RPV) Water Level indication. Analysis and testing performed for BWR fuel confirms the relationship between RPV water level and availability of the fuel cladding barrier. If water level is maintained above specified levels, the fuel cladding barrier will be maintained. If water level drops below specified limits, the cladding barrier is assumed to be breached and operator action directed to restore water level and maintain core cooling. These actions are consistent with the EPG/SAG and are based solely on RPV water level indication.

In accordance with the BWR EPG, operator actions to address Fuel Barrier integrity are based on RPV Water Level until after entry into the RPV and Containment Flooding SAG is required. The fuel cladding barrier is maintained intact when the core remains adequately cooled as indicated by RPV water level instrumentation. The fuel cladding barrier is assumed to be no longer intact when the RPV water level cannot be restored and maintained. Therefore, RPV water level is used in the EPG/SAG to provide all of the necessary information to the operator to take mitigating actions associated with the fuel cladding barrier.

If RPV water level decreases (whether due to a break in the reactor coolant system (RCS), safety/relief valve (SRV) operation, loss of RPV injection capability, or any combination of these events), and cannot be restored, entry into RPV and Containment Flooding SAG is required. This condition will occur a significant period of time prior to the occurrence of severe core damage. The timing of severe core damage is highly dependent on the event sequence, operating conditions, and fuel operating history. Furthermore, the magnitude of core damage (i.e. percent core damage), is irrelevant with respect to operator actions and accident management strategies relative to the integrity of the fuel cladding barrier. BWR accident management strategies relative to operator actions are dependent on whether Primary Containment Flooding is required.

With respect to the Commission's suggestion that existing CHARMS instrumentation can be used as a key variable, no operator action is taken relative to protection of the fuel cladding barrier based on CHARMS indications. The only operator action specifically based on CHARMS indication, is the SAG requirement that Drywell and Suppression Pool Spray is initiated before the Drywell/Suppression Chamber radiation indication reaches the level requiring a General Emergency notification.

It should be noted that CHARMS is not credited as providing information relative to the fuel cladding barrier in RG 1.97 Revisions 2 and 3. These Regulatory Guide revisions classify primary containment area radiation monitors as providing information relative to reactor coolant pressure boundary integrity, not fuel cladding barrier. The primary containment area radiation monitors are not required to have the same range as CHARMS. Consistent with the methodology provided in NEDO-33349, there are other parameters that provide a more direct indication of reactor coolant pressure boundary integrity.

In addition to CHARMS, there are other backup instrumentation and RG 1.97 Revision 4 Type E radiation detection variables, which will be available to the operator to determine if fuel cladding fission product barrier integrity has been breeched, core damage has occurred and the magnitude of the damage. These would include off-gas monitors, hydrogen monitors, and sampling of RPV radioactivity concentration. These additional variables are used for confirmation and to assist in emergency planning but are not used by operators for actions relative to the fission cladding barrier for Emergency Procedure Guidelines.

In summary, based on the methodology in NEDO-33349, RPV water level is considered the variable that provides the operators' with the primary source of accident monitoring information relative to the fuel cladding barrier. Additional variables which provide backup or diagnostic information do exist. However, these backup and diagnostic variables, which are not considered primary sources of information, should not be classified as Type C in accordance with IEEE-497 and RG 1.97 Revision 4.

SRAI 4

NEDO-33349 recommends the use of additional key variables to provide information for several functions under Regulatory Guide (RG) 1.97 Revision 4. These variables were not recommended as key variables for these functions under RG 1.97 Revision 3. The use of these variables is discussed in Section 7, "Summary of Regulatory Guide 1.97 Revision Changes." However, not all of these variables are listed under their corresponding functions in Table A-1, "Accident Monitoring Variable Comparison," of the draft markup of NEDO-33349. One example is NEDO-33349 credits Reactor Water Level as a Type C variable to provide information on the accomplishment of both the Fuel Cladding fission product barrier and the Reactor Coolant Pressure Boundary fission product barrier. This is discussed on Pages 7-7 and 7-8 of the draft markup of NEDO-33349. However, Reactor Water Level is not included on Page A-6 of Table A-1 for either Fuel Cladding or Reactor Coolant Pressure Boundary. Please either update Table A-1 to include all variables under each function (function for Type B, fission product barrier for Type C, system for Type D, or release or environmental condition for Type E) that provide information for that function, or provide a table that includes all variables, including their functions and type, which are being credited as RG 1.97 Revision 4 variables (See Question 15 below).

SRAI 4 Response

NEDO-33349 does identify additional key variables as a result of using the process established in RG 1.97 Revision 4, from what was included in RG 1.97 Revision 3. Table A-1 was intended to reconcile the changes from RG 1.97 Revision 2 and Revision 3, to the results of NEDO-33349. This Table will be expanded to also show the additional variables identified in NEDO-33349 which were not identified in the previous RG 1.97 revisions, including RPV Water Level as a Type C variable for Fuel Cladding Integrity and Suppression Pool Temperature as Type B and C variables.

SRAI 5

RG 1.97 Revision 3 recommends Suppression Chamber Spray Flow as a Type D Category 2 variable to monitor the operation of Primary Containment Related Systems. Although NEDO-33349 discusses the use of

residual heat removal (RHR) System Flow and RHR Valve Position, as an alternate to Suppression Chamber Spray Flow, it is not clear if credit is being taken for Suppression Chamber Spray Flow and/or alternate instrumentation to provide status information for Containment System performance and/or RHR System performance. Please provide information that clarifies if Suppression Chamber Spray Flow is credited, in NEDO-33349, with providing status information for Containment System performance and/or RHR System performance.

SRAI 5 Response

The Residual Heat Removal (RHR) System is capable of operating in a number of different normal operating and accident mitigating modes. Suppression Chamber flow is one of the modes considered in the symptom-based EPG. One of the functions of Type D variables is to provide information to the control room operators to verify system status. Therefore, information relative to Suppression Chamber flow is considered to be associated with RHR system performance.

Containment performance assessment does not rely upon Suppression Chamber spray flow. Containment performance is considered to be adequately characterized by Drywell Pressure, Drywell Temperature, Suppression Pool Temperature and Suppression Pool Level.

SRAI 6

RG 1.97 Revision 3 recommends Drywell Spray Flow as a Type D Category 2 variable to monitor the operation of Primary Containment Related Systems. Although NEDO-33349 discusses the use of RHR System Flow and RHR Valve Position, as an alternate to Drywell Spray Flow, it is not clear if credit is being taken for Drywell Spray Flow and/or alternate instrumentation to provide status information for RHR Safety System performance. It is also not clear if the credit is being taken for alternate instrumentation to provide status information for Containment System performance. Please provide information that clarifies if Drywell Spray Flow, and/or alternate instrumentation, is credited, in NEDO-33349, with providing status information for Containment System performance and/or RHR System performance.

SRAI 6 Response

The Residual Heat Removal (RHR) system is capable of operating in a number of different normal operating and accident mitigating modes. Drywell spray flow is one of the modes considered in the symptom-based EPG. One of the functions of Type D variables is to provide information to the control room operators to verify system status. Therefore, information relative to Drywell Spray flow is considered to be associated with RHR system performance.

Containment performance assessment does not rely upon drywell spray flow indication. Containment performance is considered to be adequately characterized by drywell pressure, drywell temperature, suppression pool temperature and suppression pool level.

SRAI 7

RG 1.97 Revision 3 recommends that RHR System Heat Exchanger Outlet Temperature as a Type D Category 2 variable to monitor operation of RHR Systems. NEDO-33349 recommends that RHR System Heat Exchanger Outlet Temperature be classified as a RG 1.97 Revision 4 Type D variable to provide information on the status of Decay Heat Removal Safety System performance. However, it is not clear if RHR System Heat Exchanger Outlet Temperature is being proposed as part of the RHR System and/or as a separate Decay Heat Removal System. Please clarify which systems RHR System Heat Exchanger Outlet Temperature is being credited as providing information about. Please describe the difference between the RHR System and the Decay Heat Removal System.

SRAI 7 Response

Boiling Water Reactors (BWRs) do not have a standalone system identified as a decay heat removal system. Long-term decay heat removal is typically performed by the combination of Residual Heat Removal (RHR) system and RHR Service Water (RHRSW) system. It is the coordinated function of the RHR and RHRSW systems to provide long-term decay heat removal capability. RHR system heat exchanger outlet temperature indication provides information to Licensed Operators with respect to decay heat removal performance of both systems.

SRAI 8

NEDO-33349 credits Other RPV Normally Closed Isolation Valve Position Inside Containment Require Opening for LOCA, Other RPV Normally Closed Isolation Valve Position Outside Containment Require Opening for Pipe Breaks Outside Containment, and Normally Closed CIV Position Inside Containment Require Opening for LOCA, as Type D variables to provide information on the status of Applicable System performance. Please identify the Applicable System(s) for each valve group.

SRAI 8 Response

NEDO-33349 is intended to document a methodology that satisfies the requirements of RG 1.97 Revision 4. As such, it establishes the criteria to be used in determining the plant specific considerations necessary to demonstrate compliance with RG 1.97 Revision 4.

Systems that penetrate Primary Containment vary from plant to plant. Each plant is required to provide a listing of isolation valves (including Valve Position Indication information for normal operating and post-accident conditions; and fail-safe modes for valve operator power failures). This information, when combined with the plant specific Emergency Operating Procedure (EOP) requirements, is necessary to establish the specific system valve position criteria.

As a result, an engineering evaluation of each plant and system design is required to identify the specific Containment Isolation Valves (CIV) that meet this requirement. NEDO-33349 will be revised to incorporate this requirement. Although it is not possible to provide a comprehensive listing, typical valves in this category would be Emergency Core Cooling System (ECCS) injection valves and systems that support decay heat removal.

SRAI 9

For Type D isolation variables, Other RPV Normally Closed Isolation Valve Position on Valves that Do Not Require Opening for either LCOA or Pipe Breaks Outside Containment and Normally Closed CIV Position Inside or Outside Containment Do Not Require Opening for LOCA, the NEDO-33349 system is not Required for Safety System Performance Indication. Please provide more details of Not Required for Safety System Performance Indication system.

SRAI 9 Response

There are containment isolation valves (CIVs) that are normally closed and do not require opening for either a loss of coolant accident (LOCA) or a pipe break outside of primary containment. Further the position switches on these valves cannot affect the valve operability. Because these CIVs are not required to operate for these events and the position switches are for indication only, the position switches are not considered safety related. Loss of position indication is not significant with respect to accident monitoring. Position indication on these valves only provides confirmation that the valve remained in its normal operating position. Since there in no automatic action relative to these valves, position indication is not considered necessary. As a result, the position switches on these valves should not be required to be environmentally or seismically qualified.

The position switches are retained as Type D variables because they may be used by the operator for other events not requiring seismic or environment qualification. To clarify this situation, a comment will be added to these position switches in NEDO-33349 Tables 5-1 and 5-2 that states that the position switches are not associated with any events requiring seismic or environmental qualification.

SRAI 10

RG 1.97 Revision 3 recommends that Main Steam Isolation Valves (MSIV) Leakage Control System Pressure be monitored by Type D Category 2 instrumentation to provide indication of pressure boundary maintenance for the Main Steam System. NEDO-33349 recommends that this instrumentation does not need to be considered a RG 1.97 Type D variable. In Table A-1, on Page A-9, the Comments column includes, "NRC approved elimination of MSIV leakage control system." Please identify the NRC document that approved the elimination of the MSIV leakage control system.

SRAI 10 Response

In the 1980s, the BWROG formed a Committee to address concerns with the Main Steam Isolation Valve (MSIV) Leakage Control System. In addition, the NRC pursued the issue of MSIV leakage under Generic Issue C-8. The NRC published NUREG-1169, "Technical Findings Related to Generic Issue C-8: Boiling Water Reactor Main Steam Isolation Valve Leakage and Leakage Treatment Methods" (Generic Letter 86-17).

Subsequent to NUREG-1169, the BWROG developed NEDC-31858P Revision 2 "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems", dated September 1993.

The NRC completed the review of NEDC-31858P Revision 2 and provided a Safety Evaluation in a letter dated March 3, 1999 which found the Topical Report acceptable for use in referencing in future individual plant submittals on the MSIV leakage issue subject to conditions.

Subsequently, BWR Owners' have provided individual submittals resulting in the elimination of the MSIV Leakage Control System, which have been approved. As this is plant specific, a note will be added to Table A-1 which references NEDC-

31858P including that a plant specific review would be needed to determine if a Type D variable is required.

SRAI 11

RG 1.97 Revision 3 recommends Drywell Purge and Standby Gas Treatment System Purge, Secondary Containment Purge, Secondary Containment, Auxiliary Building, Common Plant Vent or Multipurpose Vent, and All Other Identified Release Points, as locations for monitoring Noble Gases and Vent Flow Rate, as Type E Category 3 variables, to provide detection of significant releases and release assessment to monitor Airborne Radioactive Materials Release from Plant - Noble Gases and Vent Flow. NEDO-33349 appears to recommend that these locations should be monitored by Type E instrumentation for Airborne Radioactive Materials Released From Plant – Noble Gases and Vent Flow Rate. Table A-1, on Page A-12, lists Noble Gases and Vent Flow Rate with the comment "Plant-Specific list, includes all potential release points." The response to RAI 1-I&C-59 states, "Table 5-1, Page 5-9 contains lists of release points for BWR/4 designs." However, Table 5-1, on Page 5-6, and Table 5-2, on Page 5-12, only includes Secondary Containment Release Point Flow as a Type E variable to monitor releases. Please identify the list of release points for Airborne Radioactive Materials Released from Plant - Noble Gases and Vent Flow Rate, credited in NEDO-33349, or clarify that the list is plant-specific and provide a general description of the locations that would be on plant-specific lists. Is Secondary Containment Release Point Flow a separate variable from the Noble Gases and Vent Flow Rate variables?

SRAI 11 Response

The determination of Type E variables, which involve releases to the environment, will be plant specific as there are significant differences in the design features for the BWR fleet with respect to radiation release points. Other BWRs have different arrangements and pathways. Thus, the note in Table A-1 was added to indicate all pathways need to be monitored.

NEDO-33349 Section 4.5 defines Type E variables which include the need to monitor releases through identified pathways and Section 4.5.1 lists the pathways. The response to RAI 1-I&C-59 should have indicated for a typical BWR as Tables 5-1 and 5-2 indicate for "Typical" plant. Secondary Containment Release Point Flow and Radiation level is intended to address the need to monitor noble gases and vent flow rate as described in RG 1.97 Revision 3. Typical BWR plant designs use a common vent for releases through the Reactor Building vent, although many earlier designs used a main stack.

SRAI 12

RG 1.97 Revision 3 recommends Containment Effluent Radioactivity - Noble Gases (from Identified Release Points Including Standby Gas Treatment Vent) be monitored by Type C Category 3 instrumentation. NEDO-33349 recommends that this instrumentation does not need to be considered a RG 1.97 Revision 4 Type C variable. In Table A-1, on page A-7, there is an "E" in the IEEE-497 Consistent with RG 1.97 Rev 4 Type column, but in the BWR/4 Typical Type Column there is a "C." Is this supposed to indicate that Containment Effluent Radioactivity - Noble Gases (from Identified Release Points Including Standby Gas Treatment Vent) is being credited, in NEDO-33349, as a RG 1.97 Revision 4 Type E variable? Please indicate any corrections that might be needed to Table A-1 or provide an explanation.

SRAI 12 Response

Table A-1 includes a column titled "BWR 4 Typical". This column is based on a specific Plant's current RG 1.97 commitments. The Plant includes Containment Effluent Radioactivity – Noble Gases as a Type C, Category 3 and a Type E variable consistent with RG 1.97 Revision 2 (Plant is committed to Revision 2). NEDO-33349 concludes Containment Effluent Radioactivity – Noble Gases, meets the definition of a Type E variable based on RG 1.97 Revision 4, and does not meet the criteria as a Type C variable. A note will be added to Table A-1 to indicate the conclusion of NEDO-33349.

SRAI 13

RG 1.97 Revision 3 recommends that Type E Category 3 instrumentation be provided to monitor Estimation of Atmospheric Stability for release assessment for Meteorology assessment of releases. NEDO-33349 credits Ambient Air Temperature as a RG1.97 Revision 4 Type E variable. In Table 5-1, on Page 5-6, and Table 5-2, on Page 5-12, Ambient Air Temperature is listed without a reference to Estimation of Atmospheric Stability. In Table A-1, on Page A-12, Estimation of Atmospheric Stability is listed without a reference to Ambient Air Temperature. Is it the intent of NEDO-33349 that Ambient Air Temperature replace Estimation of Atmospheric Stability or that both Ambient Air Temperature and Estimation of Atmospheric Stability be credited as RG 1.97 Revision 4 Type E variables? Whichever is the case should be clearly stated in NEDO-33349.

SRAI 13 Response

In RG 1.97 Revision 3, Type E variables under "Meteorology", Estimation of Atmospheric Stability is defined as based on vertical temperature differences. RG 1.97 Revision 4 and IEEE-497, indicate that Type E variables include the need to monitor the environmental conditions used to determine the impact of releases, listing wind direction, wind speed, and air temperature, as typical Type E variables.

Estimation of Atmospheric Stability, is the appropriate variable of which to reference. No changes to Meteorology variables are expected, resulting from comparison of RG 1.97 Revision 4 provisions to those listed in RG 1.97 Revision 3. NEDO-33349 Table 5.1, Table 5.2, and Table A-1, will be revised to indicate Estimation of Atmospheric Stability for monitoring meteorology conditions.

SRAI 14

RG 1.97 Revision 3 recommends that Type E Category 3 instrumentation be provided to monitor Containment Air Grab Sample for release assessment, verification, and analysis for Accident Sampling of the Containment Air. The responses to RAI 1-I&C-62 and RAI 3-NSIR-9 indicate that Containment Air sampling will be included as a RG 1.97 Revision 4 Type E variable, with reference to the NEDO-32991-A as to sampling capabilities. However, in Table A-1, on Page A-13, under the column titled, "IEEE-497 Consistent with RG 1.97 Rev 4," Containment Air is listed as "N/A," with the comment, "Grab sample. See [post-accident sampling systems] PASS LTR." The NRC staff has been unable to find any discussion of the Containment Air Grab Sample Type E variable in NEDO-33349. Please clarify if NEDO-33349 credits Containment Air Grab Sample as a Type E variable and add the appropriate discussion in NEDO-33349. Similar to Containment Air Grab Sample, Primary Coolant and Sump Grab Sample is listed in Table A-1, on Page A-13, under the column titled, "IEEE-497 Consistent with RG 1.97 Rev 4," Primary Coolant and Sump is listed as "N/A," with the comment, "Grab sample. See PASS LTR." The NRC staff has been unable to find any discussion of the Primary Coolant and Sump Grab Sample Type E variable in NEDO-33349. Please clarify if NEDO-33349 credits Primary Coolant and Sump Grab Sample as a Type E variable and add the appropriate discussion in NEDO-33349.

SRAI 14 Response

NEDO-33349 Table A-1 will be revised to list Containment Air Grab Sample and Primary Coolant and Sump Grab Sample as Type E variables with the reference to the Post Accident Sampling System (PASS) Licensing Topical Report (LTR), which defines expectations. Section 4.5.2 of NEDO-33349 will be revised to include reference to the PASS LTR (NEDO-32991-A) for determination of Type E variables used for post-accident sampling analysis.

SRAI 15

The purpose of Table A-1, as described on Page A-1, is to provide a comparison of the accident monitoring variables developed using the BWROG evaluation methodology to those in RG 1.97 Revision 2 and Revision 3 and a typical BWR/4 plant. However, Table A-1 does not include all RG 1.97 Revision 2 or Revision 3 variables or all NEDO-33349 variables. While Table A-1 is helpful in determining how the variables listed in the table would be applied under RG 1.97 Revision 4, it does not provide a complete picture for all variables that are being credited as RG 1.97 Revision 4 variables. NEDO-33349 should include a table that includes all variables, including their functions and type, which are being credited as RG 1.97 Revision 4 variables.

SRAI 15 Response

The original intent of NEDO-33349 Table A-1 was to assist in reconciling variables included in RG 1.97 Revisions 2 and 3, to the conclusions of NEDO-33349. To simplify the Table, not all of the detail contained in RG 1.97 Revisions 2 and 3 (i.e. Type E variables) was included. Type E variables will differ depending on Plant design features. The list of potential variables included in RG 1.97 Revisions 2 and 3 for Type E, have been used by BWR Owners' to develop specific lists.

Tables 5-1 and 5-2, contained determination results of the process described in NEDO-33349, pertaining to RG 1.97 Revision 4. Table A-1 will be improved through expansion to include all variables (including those identified uniquely in NEDO-33349), and detailed lists in RG 1.97 Revisions 2 and 3. In addition, Tables 5-1 and 5-2 will be expanded to address issues identified in related RAI / SRAI, associated with Type E variables.

SRAI 16

NEDO-33349 has recommended a significant number of changes from the list of variables in RG 1.97 Revision 3. The NRC staff has constructed the attached table to indicate the variables that NEDO-33349 recommends for each function under RG 1.97 Revision 4 and for each RG 1.97 Revision 3 variable that would no longer be monitored. Please review the attached table to ensure that the NRC staff understands the NEDO-33349 position for

each variable. The attached table should not be considered as an approval of any variable credited in NEDO-33349, but rather a summary of the recommendation in NEDO-33349. The NRC staff is aware that the responses to the above questions may have an impact on the attached table. Please provide corrections and/or comments to the attached table.

SRAI 16 Response

The table provided with the SRAI letter provides an effective documentation and comparison between RG 1.97 Revision 3, and the results of NEDO-33349. Tables in NEDO-33349 will be revised, resulting from issues identified in the SRAIs, will yield greater consistency within the SRAI Table (especially with respect to Type E variables).

As noted in the response to SRAI 1 and in Section 7 of NEDO-33349, the Boiling Water Reactor (BWR) Emergency Procedure Guidelines (EPG) do not align with the list of Functions contained in RG 1.97 Revision 3. Based on the BWR EPG, Reactor Pressure Vessel (RPV) Control encompasses Reactivity Control, Level Control and Pressure Control. The BWROG recommends revising the Table (enumerated below) as Type B Reactor Pressure Vessel Control and Reactivity Control as B.1; Level Control as B.2; and Pressure Control as B.3. In spite of the terminology difference, all the RG 1.97 Revision 3 Type B and C Functions are met, as discussed in SRAI 1.

The following are suggested changes/clarifications to the SRAI Table:

1. BWRs currently have differing lists for what is a Type A variable. The Table notes as Plant Specific, but it is not clear that a different list may exist. Suggest adding in Remarks: "...lists may vary for specific plants."
2. Type B Pressure Control, Drywell Pressure – suggest including in remarks: "...entry condition for RPV control, not relied upon for RPV control actions..."
3. RG 1.97 Revision 3, Containment Function lists Reactor Coolant System (RCS) pressure with the Remark being "Not a Revision 4 Variable". RCS Pressure is the same as Reactor Pressure Vessel (RPV) Pressure, which is a Type A, B, and C variable, under Revision 4.
4. Type D, Containment System, Drywell Spray Flow – suggest adding in Remarks: "Alternate instrument"
5. Type D, RCIC System – suggest adding to remarks "...required for AOOs..."
6. Type D, HPCI System, Condensate Storage Tank Water Level – should be Category 3

7. Type D, RHR System – change RHR System Flow to "LPCI/LPCS flow for NEDO-33349"
8. Type D, RHR System – delete RHR System Valve Position for NEDO-33349
9. Type D – delete Decay Heat Removal System and replace with "RHR system for RHR Heat Exchanger"
10. Type D, LPCS - delete. This is the same as Core Spray Flow
11. Type E, Airborne Radioactive Materials Released from Plant-Noble Gases, variable Secondary Containment Release Flow – delete. This is the same as Secondary Containment –Noble Gases and Vent Flow Rate
12. Consistent with our response to SRAI 13, Ambient Air Temperature will be listed as Estimation of Atmospheric Stability for NEDO-33349

SRAI 17

a. **RAI NSIR-11 Response / Supplemental RAI. The NRC staff does not find the information provided by the BWROG in response to RAI NSIR-11 persuasive. As identified in Paragraph A of the RAI text, the NRC staff views the RPV level at any given time as only an indicator that precursor conditions to fuel damage exist; the level does not indicate that damage has occurred or its magnitude. The NRC staff does not agree that the magnitude of core damage is irrelevant with respect to accident management strategies, as stated by the RAI response. The NRC staff considers the classification of emergency conditions to be a significant aspect of accident management and emergency preparedness. When plant design and the actions of plant operators are unsuccessful in preventing or mitigating the accident, reliance shifts to emergency preparedness. The NRC staff also notes that the effectiveness of public protective actions is enhanced when the actions are implemented prior to the start of a release (e.g., before such a release would be indicated by the Type E effluent monitors). The containment high range area monitors (CHARMs) provide a significant and critical input to the accident and emergency management decisions, especially given the retirement of the PASS at many facilities.**

b. **Although the NRC staff agrees that there can be significant uncertainties in detection of fuel damage by CHARMs, these uncertainties do not change the fact that, absent a common-mode failure of both CHARMs, a significantly elevated reading on these monitors can have only one meaning: core damage has occurred.**

The monitor reading under these circumstances will be proportional to the amount of core damage. Contrast this with the fact that the RPV level can only mean that conditions that could cause fuel damage are present. Once the RPV level is restored by ECCS, the RPV level reading provides no intelligence with regard to the occurrence or amount of core damage.

c. The NRC staff recognizes that a RCS barrier failure has to occur before the fission products released into the RCS by core damage can be released to the drywell, as stated in the response. The NRC staff also recognizes that such a release could bypass the drywell. However, for the more probable accident sequences, the RCS barrier failure will precede the fuel clad barrier failure and the release will be to the drywell and will be indicated by the CHARMs.

d. The response that the CHARMs are challenged to provide on-scale indication during normal operation and may be unable to detect low levels of coolant activity, as stated in the response, is not a concern in as much as the CHARMs were installed to serve as post-accident accident monitors. The consequences of upsets in normal operations are administratively controlled by technical specifications on RCS specific activity, RCS leakage, and containment integrity, leak detection systems, normal range radiation monitors, etc., were evaluated by FSAR analyses, and shown not to significantly affect public health and safety.

e. With regard to retention of fission products in the suppression pool as addressed in the RAI response, accident progression insights developed since TMI show that, although the blowdown of the RCS to the suppression pool will occur at the time of the RCS failure, the onset of release of gap fission products from the fuel will not occur before 2 minutes and will continue over several hours, with the release associated with core melt starting at about 30 minutes or later (See NUREG-1465). It is the blowdown of the RCS to the drywell that provides driving force for fission products to be driven into the suppression pool. Since this fuel damage occurs long after the blowdown occurs, significant fission product would be released to the drywell after the blowdown was complete and the pressure differential that would drive flow into the suppression pool had abated. Nonetheless, even if the suppression pool were to retain the aerosol fission products, the retention would not mitigate the substantial release of noble gas fission products from the fuel. Although the CHARM monitors could detect and measure iodine, the

monitor indication is largely driven the noble gases released to the drywell. As the accident progresses, analyses indicate that drywell and suppression pool will behave as a single volume as the action of vacuum breakers and recirculation injection redistribute fission products. The NRC staff recognizes that there are uncertainties involved; nonetheless, elevated readings on the CHARMs can only be caused by fuel damage and the magnitude of those readings would be proportional to the magnitude of core damage. This cannot be said for the RPV level instrumentation as an indicator of actual fuel damage.

f. **The NRC staff does not find the proposed reliance on RPV level as the sole indication of fuel clad barrier failure to be prudent or acceptable. The exchange between the RAI, the BWROG response to the RAI, and this supplementary RAI serve to highlight the need for diverse indications. This diversity is entirely in keeping with the defense-in-depth philosophy upon which plants were licensed. The NRC staff notes that another diverse means of assessing fuel damage, namely the post-accident sampling system, was previously retired at many plants, and is no longer available at those plants.**

   Please provide additional information which supports your position that designating the RPV level instrument as the sole Type C instrument for assessing the occurrence and magnitude of fuel damage, is technically appropriate and supportive of the defense-in depth philosophy, or revise the LTR accordingly.

<u>SRAI 17 Response</u>

For convenience in responding, this supplemental request for information has been separated into six separate requests for information. Each separate request is responded to based on the technical work performed by the BWROG to develop an acceptable methodology for determining the appropriate variables which meet the requirements of RG 1.97 Revision 4.

As there are differences with respect to the classification of CHARMS, the BWROG proposes that the existing requirement contained in RG 1.97 Revisions 2 and 3 for CHARMS be retained as an exception to the provisions of RG 1.97 Revision 4 for Type E variables. NEDO-33349 will require that CHARMS meet all Type E Category 1 variable requirements, and be consistent with plant specific licensee commitments established in NUREG 0737 II.F.1 and RG 1.97 Revisions 2 or 3. Please note that the Category 1 requirement effectively makes CHARMS a Type C instrument, however, this does not imply that the BWROG EPG or other source documents listed in IEEE-497 will be revised to reflect CHARMS as a Type C variable. NEDO-33349 will be revised with the following to be included in Section 7.4.2:

The Containment High- Range Radiation Monitor (CHARM) is included in RG 1.97 Revisions 2 and 3 as a Type E Category 1 variable. The existing requirements for CHARMs are to be retained as an exception to the provisions of RG 1.97 Revision 4 for Type E variables. CHARMs is to meet the plant specific licensee requirements established in plant commitments to NUREG 0737 II.F.1 and RG 1.97 Revisions 2 and 3 Category 1.

Response to technical issues identified in SRAI 17 a-f:

a. NEDO-33349 provides a methodology to comply with Regulatory Guide (RG) 1.97 Revision 4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" endorses the use of the Institute of Electrical and Electronics Engineers Std 497-2002 (IEEE-497), "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations" as an acceptable method for providing instrumentation to monitor variables for accident conditions subject to regulatory positions. Based on the NEDO-33349 methodology and consistent with RG 1.97 Revision 4 and IEEE-497, the BWROG has determined that the variable that provides the most direct indication of the potential breach of the fission product barrier is Reactor Pressure Vessel (RPV) Water Level indication. If water level is maintained above specified levels, the fuel cladding barrier will be maintained. If water level drops below specified limits, the fuel cladding barrier is assumed to be breached and operator action is directed to restore water level and maintain core cooling.

   Consistent with the application of generic symptom-based Emergency Procedure Guidelines and Severe Accident Guidelines (EPG/SAG), the accident management operator actions to address the status of the fuel barrier are based solely on RPV water level until after entry into the RPV and Containment Flooding SAG is required. The fuel cladding barrier is maintained intact when the core remains adequately cooled as is evidence by the RPV water level. The fuel cladding barrier is assumed to be no longer intact when the RPV water level cannot be restored and maintained. Therefore, RPV water level is used in EPG/SAG to provide the necessary information to the operator to take mitigating action associated with the fuel cladding barrier.

   Operators are directed by the EPG to initiate the RPV and Containment Flooding Guideline if RPV water level cannot be restored. This action represents transitional action from the EPG, which is within the scope of RG 1.97 Revision 4, to the SAG, and is part of BWR accident management strategy to minimize radioactive releases from the plant. The extent of core damage does not influence this key step in providing RPV water level restoration and assessment of extent of core damage.

It is at this point that the focus on actions related to severe core damage begins and the focus transitions to the Containment and Radioactive Release Control SAG to limit the radioactive release from the plant. It is this SAG that has the operator actions associate with the Type E variables, including the Containment High Area Radiation Monitors (CHARMS). The only operator action that is required specifically based on CHARMS indications is the SAG requirement that Drywell and Suppression Pool Spray is required before the Drywell/Suppression Chamber radiation indication reaches the level requiring a General Emergency notification.

It should be noted that NEDO-33349 is consistent with RG 1.97 Revisions 2 and 3 in that CHARMS is not credited as providing information relative to the fuel cladding barrier in RG 1.97 Revisions 2 and 3 and are considered Type E variables. RG 1.97 Revisions 2 and 3, classify primary containment area radiation monitors as providing information relative to Reactor Coolant Pressure Boundary integrity, not fuel cladding barrier. The Primary Containment Area Radiation Monitors are not required to have the same range as CHARM. Consistent with the methodology provided in NEDO-33349, there are other parameters that provide a more direct indication of Reactor Coolant Pressure Boundary integrity.

The BWROG does not disagree that actual core damage is important to emergency response but has concluded that it is not needed for accident management strategies involving control room operator actions to mitigate the consequences of an accident. The magnitude of core damage is used for Emergency Preparedness and is not contained within the requirements applied in the identification of RG 1.97 for Type C variables. Type E variables are defined in IEEE-497 as those variables required for use in determining the magnitude of the release of radioactive material and continually assessing such releases. This is the function provided by CHARMS in Emergency Management along with the other Type E variables.

The BWROG acknowledges individual plant designs and licensee commitments with respect to CHARMS may differ. Before the CHARMS design at any current operating plant can be modified, all aspects of the design are to be considered consistent with the requirements of 10CFR50.59. The design considerations include, but are not limited to, commitments made relative to NUREG-0737 - the quality requirements; the relationship of CHARMS to the requirements relative to primary containment radiation monitors; and the sampling capability and other requirements committed to relative to RG 1.97 Revisions 2 or 3. NEDO-33349 will be revised to reflect these requirements.

b. The BWROG concurs that there can be significant uncertainties in the detection of fuel damage by CHARMS. These uncertainties can be associated with the characteristics of the postulated event leading to core damage; the release pathway from the RPV; effectiveness of the operator actions relative to event mitigation; and the accuracy of the CHARMS instrumentation. The BWROG also concurs that a significantly elevated reading from CHARMS, if verified, is indicative of significant core damage.

   However, the CHARMS readings are not necessarily proportional to the amount of core damage. CHARMS readings are highly dependent on the release pathway and the availability of the core cooling systems. Based on typical BWR probabilistic safety analyses, the more likely event scenarios leading to significant core damage are not associated with a Loss of Coolant Accident (LOCA).

   For these events, the Reactor Coolant Pressure Boundary will be intact and the primary fission product release pathway will be through the Safety/Relief Valves (SRVs) to the Suppression Pool. As a result, the majority of the fission products released from the RPV will be retained in the suppression pool, and the CHARMS readings will be substantially smaller than for a direct release to the Drywell. As a result, other means will be required to determine the extent of core damage (i.e. grab samples of reactor coolant or suppression pool water).

   For the Loss of Coolant Accident, the release of fission products to the drywell is dependent on the timing of the core damage and the effectiveness of operator action to restore core cooling. Depending on the timing and effectiveness of core cooling, fission products released from the fuel may be trapped in the core cooling water, and transported to the suppression pool through the break. As a result, CHARMS will likely indicate significant core damage has occurred, but the magnitude of the core damage may be underestimated.

   As a result for BWR Pressure Suppression Containments, CHARMS is likely to provide relatively little quantitative information with respect to core damage. Furthermore, all of the operator required accident management actions relative to the fuel cladding barrier, will be taken based on RPV Water Level indication prior to any indication of severe core damage by as indicated by CHARMS. However, the BWROG does recognize CHARMS' role in Emergency Planning actions taken to minimize releases similar to other Type E variables.

c. The BWROG concurs that a failure in the Reactor Coolant Pressure Boundary is a condition required for a release to the Drywell, which can provide an indication of fuel cladding barrier failure during a core damage scenario. However, based on typical BWR probabilistic safety analyses, the more likely event scenarios leading to significant core damage are not associated with a LOCA. Further, actions taken by the operators based on RPV level to mitigate core damage can have the effect of reducing the CHARMS readings. Therefore, dependence on CHARMS without other verification (i.e. grab samples of reactor coolant or suppression pool water) is not recommended.

d. The BWROG concurs that CHARMS is a Type E accident monitoring variable. It is noted that this classification is consistent with RG 1.97 Revisions 2 and 3. The BWROG also notes that the Primary Containment Area Radiation Monitors (not CHARMS) were typically a part of the original plant design. However, some plants may have combined both functions. For plants that have combined both functions, the requirements of 10CFR50.59 must be addressed before any plant modification is made.

e. This scenario is associated with the very low probability large break LOCA with failure of the Emergency Core Cooling System. The BWROG concurs that for this scenario, it is likely that CHARMS will have elevated readings that are indicative of severe core damage. However, for this scenario, it should be noted that the operators would already have taken all possible actions to restore Adequate Core Cooling on RPV level indications. Furthermore, entry into the RPV and Containment Flooding SAG would already have been made based on RPV level. As a result, the only operator action that is required specifically based on CHARMS indications is the SAG requirement that Drywell and Suppression Pool Spray is required before the Drywell/Suppression Chamber radiation level reaches that required for a General Emergency notification. This action would be taken to reduce airborne fission products in order to minimize releases from the plant as part of the emergency response.

f. Based on the NEDO-33349 methodology and consistent with RG 1.97 Revision 4 and IEEE-497, the BWROG has determined that the variable that provides the most direct indication of the potential breach of the fission product barrier is RPV Water Level indication. If water level is maintained above specified levels, the fuel cladding barrier will be maintained. If water level drops below specified limits, cladding barrier is assumed to be breached. Furthermore, all required operator accident management actions in the EPG/SAG relative to the fuel cladding barrier, are taken based on RPV water level.

   The BWROG does not claim that RPV Water Level indicates that fuel barrier failure has occurred. However, the appropriate actions relative to the fuel barrier have been taken based on RPV Water Level indications.

   The BWROG notes that diversity in accident monitoring variable indications is not required in RG 1.97 Revision 4. However, the plants do have other means for determining the extent of fuel damage including hydrogen monitoring and post-accident sampling capability, consistent with the NRC's acceptance of the removal of the Post Accident Sampling System (PASS).