BWR OWNERS' GROUP

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Subject: BWR Owners' Group Licensing Topical Report (LTR) NEDO-33349 Revision 1, BWR Application to Regulatory Guide 1.97 Revision 4

Attachment: (1) NEDO-33349 – BWR Owners' Group Licensing Topical Report (LTR), Revision 1: BWR Application to RG 1.97 Revision 4
(2) BWROG Response to NRC Staff Comments Concerning November 13, 2006 Pre-Submittal Meeting (Draft LTR)

Attached for NRC review is the BWR Owners' Group Licensing Topical Report (LTR) NEDO-33349 Revision 1, BWR Application to Regulatory Guide 1.97 Revision 4. A Draft LTR was previously forwarded to the NRC in support of a Pre-Submittal Meeting held on November 13, 2006. NRC provided comments resulting from the Pre-Submittal Meeting in a letter dated March 13, 2007, from Stacey Rosenberg, Chief Special Projects Branch to Michelle Honcharik, Special Projects Branch. All comments have been incorporated into the Attached LTR. To assist in NRC's review, Attachment (2) is a summary of the comments and the changes made to the LTR.

Please review the attached and provide your proposed review schedule of the LTR. If there are any questions, please contact the undersigned.

Sincerely,

RC Bunt

Randy Bunt, Chair
BWR Owners' Group

cc: Michelle C. Honcharik, NRC
BWROG Primary Representatives
BWROG RG 1.97 Committee
Ken McCall, GEH

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NRR

Response to NRC Staff Comments Concerning the BWROG Pre-Submittal Meeting on the BWR Application of IEEE 497-2002 Draft Licensing Topical Report (LTR)

A revision has been made to the draft BWR Owners' Group Licensing Topical Report (LTR) BWR Application of IEEE 497-2002 which was discussed at the Pre-Submittal meeting on November 13, 2006. Enclosed in the cover letter is the Final LTR for NRC review titled BWR Application to RG 1.97 Revision 4. The revision resulting in the Final LTR which incorporates all of the NRC provided comments provide by Stacey Rosenburgh, Chief Special Projects Branch to Michelle Honcharick, Special Projects Branch on March 13, 2007.

To assist in NRC's review of the LTS the following are the comments provided and references to the changes made in the Final LTR'.

NRC Comment

The draft LTR addresses IEEE 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations." Since Regulatory Guide (RG) 1.97, Revision 4, "Instrumentation for Light Water Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following and Accident," endorses, with exceptions, IEEE 497-2002, the LTR should address RG 1.97, Revision 4, and its regulatory positions.

LTR Changes

Report title changed and numerous changes made throughout the Report to refer to the provisions of RG 1.97 Revision 4.

NRC Comment

Section 3.2, "Type B Variables," of the draft LTR, addresses contingency actions. However, the only discussion of contingency actions in IEEE 497-2002 is in regard to Type A variables. This discussion of contingency actions in IEEE 497-2002 has been modified by Regulatory Position 4 of RG 1.97, Revision 4. The LTR should be revised to relocate this discussion and should take into consideration Regulatory Position 4.

LTR Changes

A section has been added to the LTR to address the Regulatory Positions including Regulatory Position 4. Section 3.2 in the draft LTR is now Section 4.2 and it has been revised. Contingency actions have been considered in the development of the Type A variables. As noted, the process used in the LTR is comprehensive in scope to assure that contingency actions are being addressed.

NRC Comment

The topic of compliance with industry standards that are referenced by IEEE 497-2002 is addressed generically in the draft LTR. The LTR should be revised to state that plant-specific applications that would reference this LTR should address any deviations from the referenced standards.

LTR Change

Section 3.6 of the LTR addresses the issue of codes and standards.

NRC Comment

The draft LTR appears to only address complete conversions to IEEE 497-2002 and not modifications as described in Regulatory Position 1 of RG 1.97, Revision 4. The LTR should indicate that it is also applicable for modifications as described in Regulatory Position 1 of RG1.97, Revision 4. The LTR should also include clear descriptions of a complete conversion to RG 1.97, Revision 4, and a modification as allowed by Regulatory Position 1 of RG 1.97, Revision 4. It appears that what the BWROG representatives described during the meeting as a modification is actually a conversion with deviations from newer versions of industry standards.

LTR Change

Section 3.1 addresses the question of the applicability of the LTR for complete conversions and for modifications. As noted, it is expected that currently operating Plants will use the analysis for modifications. Note that a Plant would consider the conversion of the post accident monitoring to a digital system as a modification. The analyses performed can be applied to a complete conversion or to a change in a component.

NRC Comment

Although during the meeting the BWROG representatives indicated that the LTR referenced NEDO-33160-A (relaxation of safety relief valve position indication), the NRC staff was unable to find this reference. A reference and appropriate discussion of NEDO-33160-A should be included in the LTR if appropriate.

LTR Change

Section 7.3.3 has been added to the Report to address NEDO-33160-A. The process used in NEDO-33160-A for a specific variable is similar to what is included in this LTR for all post accident instrumentation.

NRC Comment

Table A-1 in the draft LTR is hard to follow. The LTR should include information that clarifies that the column titled "BWR/4" describes an example of the current type and category classifications of instrumentation currently installed at a typical BWR/4 plant. The LTR should include information to clarify that this table includes all BWR variables listed in RG 1.97, Revision 3. This table should include the purpose of each variable (the bold text in the variable column of RG 1.97, Revision 3) and the equivalent purpose of each variable under RG 1.97, Revision 4. During the meeting, it was stated that the order that the variables are listed in this table was the order that they appear in RG 1.97, Revisions 3. However, it does not appear that this is the case. The order that variables are listed in this table should be the order that they appear in RG 1.97, Revision 3.

The LTR should include (either in Table A-1 or in a separate table) the following for each variable: the variable name, Revision 3 type, Revision 3 category, Revision 3 purpose, Revision 4 type, Revision 4 purpose, and comments including information about change of type, purpose, deletion from the list of variables, or addition to the list of variables.

The LTR should include an additional table that shows variables that under RG 1.97, Revision 4, have different types, purposes, deletion from the list of variables, addition to the list of variables, and why the change is appropriate, from what was provided under RG 1.97, Revision 3.

LTR Comment

We agree that the draft tables were difficult to follow. We tried to show the results in different formats. We have revised the table to be consistent with the format used in RG 1.97 Revision 3 showing the resulting changes by using the process established in Revision 4.

NRC Comment

The LTR should include more details concerning the proposed change in the primary containment radiation monitoring variables going from both Type C, Category 1, and Type E, Category 1, variables (with different ranges) under RG 1.97, Revision 3, to only a Type E variable under RG 1.97, Revision 4. It is not clear to the NRC staff how this variable would not be considered a Type C variable under RG 1.97, Revision 4. The purpose of this variable under RG 1.97, Revision 3, is for detection of a breach of the reactor coolant pressure boundary and the verification of the level of radiation in the containment. The LTR should include a discussion of how Type C variables in the LTR meet all of the criteria of Clause 4.3 of IEEE 497-2002 and if the purpose of the Type C variables has changed from Revision 3 to Revision 4 of RG 1.97.

LTR Change

Section 7.0 has been added to the LTR to provide a summary of the proposed changes. Section 7.3.2 addresses Containment Radiation Monitors and why it does not meet the requirements as a Type C variable for a BWR.

NRC Comment

The LTR should include more details for other variables that would change types from Revision 3 to Revision 4 of RG 1.97. One example is containment isolation valve indication which is a Type B, Category 1, variable under RG 1.97, Revision 3, and would become a Type D variable under RG 1.97, Revision 4.

LTR Change

Section 7.3.1 addresses Containment Isolation Valve Position Indication and our determination that this is a Type D variable.

NRC Comment

Although not included in the LTR, technical specifications (TSs) were discussed at the meeting. The BWROG would like to see changes to the Improved Standard TSs that would reflect the LTR. A discussion of this subject should be conducted with representatives of the TS Branch (ITSB) of the Division of Inspection and Regional Support.

Response

Our plan is to address resulting Technical Specification Changes later in 2007 after we have agreement on the review of the LTR. We will be working with the Technical Specification Task Force (TSTF) to develop a Traveler for proposed changes to the Improved Standard TSs based on the results of the LTR. We agree that a discussion should be held with the NRC TS Branch and will have this coordinated through the TSTF.

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GEH

NEDO-33349
Revision 1
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BWR Owners' Group Licensing Topical Report

BWR Application to RG 1.97 Revision 4

IMPORTANT NOTICE REGARDING THE CONTENTS OF THIS REPORT

Please Read Carefully

Abstract

This report was prepared to establish a methodology for demonstrating compliance of boiling water reactor (BWR) plants to the requirements of Regulatory Guide (RG) 1.97 Revision 4, Institute of Electrical and Electronics Engineers' Standard 497-2002 (IEEE-497) "Criteria for Accident Monitoring Instrumentation for Nuclear Plants," as an acceptable method for providing instrumentation to monitor variables for accident conditions subject to regulatory positions. RG 1.97 Revision 4 in its endorsement of IEEE-497 establishes criteria for accident monitoring instrumentation for nuclear power generating stations intended to provide a more comprehensive approach to accident monitoring than the prescriptive guidance provided in RG 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Included in this report is an identification of the instrument requirements for typical currently operating BWR to comply with RG 1.97 Revision 4 consistent with the regulatory positions. The BWR Owners' Group Regulatory Guide 1.97 Committee directed the work that is documented in this report. Based on this work, it has been determined that current BWR operating plants generally comply with the provisions of RG 1.97 Revision 4. It has also been determined that use of this new Regulatory Guide allows a more appropriate determination of the design and qualification requirements applicable to selected variables. Guidelines for applying this generic work to specific BWR plants are provided.

PARTICIPATING UTILITIES

The utilities listed below contributed to the development of this report. However, while this report has been endorsed by a substantial number of the members of the BWR Owners' Group, it should not be interpreted as a commitment of any individual member to a specific course of action. Each member must endorse any BWROG position in order for that position to become the member's position.

UTILITY	PLANT
AmerGen	Clinton
Constellation	Nine Mile Point
Detroit Edison	Fermi
Energy Northwest	Columbia
Entergy	Pilgrim FitzPatrick Grand Gulf River Bend Vermont Yankee
Exelon	Dresden LaSalle Quad Cities Limerick Peach Bottom Oyster Creek
First Energy	Perry
Nebraska PPD	Cooper
Nuclear Management Corp	Monticello
Florida Power	Duane Arnold
PPL	Susquehanna
Progress Energy	Brunswick
PSEG Nuclear	Hope Creek
SNC	Hatch
TVA	Browns Ferry

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ACRONYMS AND ABBREVIATIONS

AOO	Anticipated Operational Occurrence
ATWS	Anticipated Transients without Scram
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners' Group
CIV	Containment Isolation Valve
DOR	Division of Operating Reactor
EAL	Emergency Action Level
EOC	End of Cycle
EPGs	Emergency Procedure Guidelines
EOPs	Emergency Operating Procedures
ESF	Engineering Safety Feature
ESW	Emergency service water
GDC	General Design Criteria
GE	General Electric
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
IEEE	Institute of Electrical and Electronics Engineers
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LTR	Licensing Topical Report
MSIV	Main Steamline Isolation Valves
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PAM	Post Accident Monitoring
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RHR	Residual Heat Removal
RPT	Recirculation Pump Trip

RPV	Reactor Pressure Vessel
SAGs	Severe Accident Guidelines
SLCS	Standby Liquid Control System
SQUG	Seismic Qualification Utility's Group
TMI	Three Mile Island
UFSAR	Updated Final Safety Analysis Report

EXECUTIVE SUMMARY

Regulatory Guide (RG) 1.97 Revision 4 "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" (Reference 1) endorses the use of the Institute of Electrical and Electronics Engineers (IEEE) Std 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations" (Reference 2) as an acceptable method for providing instrumentation to monitor variables for accident conditions subject to regulatory positions. The regulatory position include requiring a current operating plant to perform a complete analysis of the plant's accident monitoring variables if they wish to voluntarily use RG 1.97 Rev 4 for a complete conversion or for modifications. This report provides the process and basis for a complete analysis of the plant's accident monitoring system.

RG 1.97 Revision 4 provides a more flexible and comprehensive method of determining an appropriate set of accident monitoring variables for nuclear power plants. This is accomplished by providing explicit criteria establishing how the variables are to be determined. In addition, the specific design and qualification requirements are established based on the importance of the specific variable type. It is intended that RG 1.97 Revision 4 be used to satisfy the prescriptive guidance previously provided by the Nuclear Regulatory Commission in Revision 3 to RG 1.97 (RG 1.97 – Reference 3), "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environ Conditions During and Following an Accident."

The IEEE 497 Standard identifies five specific variable types that are similar to RG 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environ Conditions During and Following an Accident." The selection criteria established by the Standard is intended to provide a set of variables that is similar to the prescriptive list contained in RG 1.97. However, when the Standard is applied, the basis for the selection of each variable can be identified in a comprehensive manner that allows appropriate design and qualification requirement to be applied. The Standard differs from RG 1.97 in that a consistent set of design and qualification requirements is applied to each of the five variable types.

This generic Licensing Topical Report (LTR) has been prepared at the direction of the Boiling Water Reactor (BWR) Owners' Group (BWROG) to identify a methodology that can be used to comply with RG 1.97 Revision 4. It includes implementation recommendations on meeting the RG 1.97 Rev 4 regulatory positions and use of the methodology results to support plant modifications based on the provisions of Rev 4. This methodology is intended to be applicable to all operating BWR plants. The methodology has been developed based on generic BWR safety analysis methodology consistent with the typical BWR plant License Basis and the application of generic symptom-based emergency procedure guidelines (EPGs).

To demonstrate the applicability of the methodology, typical BWR plants (a BWR/4 and 6) are evaluated and a typical list of accident monitoring variables are identified. The list of variables includes the variable type, classification basis, and design and qualification requirements. The list of variables identified using the BWROG methodology does, in some cases, differ from those identified in RG 1.97 Revision 3. These differences are due to the application of the specific criteria identified in the Standard. However, the overall objective of providing an acceptable set of accident monitoring instrumentation is met.

1. INTRODUCTION AND SUMMARY

1.1 Background

As a result of the accident at Three Mile Island (TMI) Unit 2, the Nuclear Regulatory Commission (NRC) established rigorous guidance for accident monitoring systems for light water reactors. The current guidance is provided in Revision 3 to Regulatory Guide 1.97 (RG 1.97 - Reference 3). RG 1.97 provides a specific detailed list of the variables that are required to be monitored and includes a comprehensive list of design and equipment qualification requirements. All current operating nuclear power plants have provided a set of accident monitoring instruments that are consistent with the specific variables identified in RG 1.97.

As the nuclear industry has matured, it has become increasingly difficult to maintain or replace existing accident monitoring system equipment. Further, improved and more reliable instrumentation designed to different standards has been developed. To respond to this situation, the Institute Electrical of and Electronics Engineers (IEEE) identified a need for a more flexible standard to allow the increased use of microprocessor based and other instrumentation systems for both the current generation and advanced nuclear power plants. To satisfy this need, IEEE Std 497 - 2002 (IEEE-497 - Reference 2) was developed and issued. The NRC has endorsed use of this Standard in RG 1.97 Revision 4 dated June 2006 for new plants and for current operating plants subject to regulatory positions. One of the objectives of IEEE-497 was to allow a flexible basis for making changes in accident monitoring systems for currently operating plants. This was accomplished by criteria for selecting accident monitoring variables instead of the current prescriptive list contained in RG 1.97. To further standardize the design and qualification requirements, these requirements are established based on the level of importance of the specific variable type.

The BWR Owners' Group has determined that RG 1.97 Revision 4 provides a more comprehensive basis for establishing accident monitoring requirements. To implement RG 1.97 Revision 4 a consistent methodology has been developed that can be applied to currently operating boiling water reactors (BWRs). This methodology and examples of its application provided in this report meet the RG 1.97 Revision 4 Regulatory Position (1) that a complete analysis of accident monitoring variables be performed for use in existing operating plant modifications. It is intended that this methodology provide an acceptable alternate to the current prescriptive list of variable identified in RG 1.97 Revision 3.

1.2 Report Scope

This report provides an evaluation methodology for currently operating boiling water reactor (BWR) plants that can be used to comply with RG 1.97 Revision 4. The methodology is intended to be applicable to all currently operating domestic BWR product lines (BWR/2 to 6). Two examples of the application of this methodology are provided. The examples are for typical BWR/4 and 6 plants. These application examples demonstrate that current plants are essentially in compliance with RG 1.97 Revision 4 and IEEE-497-2002.

The report consists of seven sections.

Section 1 contains the introductory and summary material. Included is a discussion of the background, the report scope, assumption made in the evaluation, limitations of the application of the methodology, and identification of the safety analysis and license basis requirements. The objective of the report is to demonstrate that the application methodology is capable of identifying a consistent and comprehensive set of accident monitoring variables that comply with RG 1.97 Revision 4.

Section 2 contains a discussion of the requirements of RG 1.97 Revision 4 as they pertain to the application methodology contained in IEEE 497. Included are the selection criteria for all five variable types along with the key design and qualification requirements for each of the variable types. Of particular importance are the seismic and environmental qualification requirements applicable to each variable type.

Section 3 contains a discussion of Regulatory Guide 1.97 Revision 4 regulatory positions as they relate to the adoption of IEEE-497. Included is how this Licensing Topical Report may be utilized by operating BWR's to comply with the provisions of Revision 4.

Section 4 provides the evaluation methodology. The evaluation methodology is dependent on the variable type. Inherent in the evaluation methodology is consideration of the safety analysis and license basis requirements, the emergency procedure guidelines, fission product barriers, safety system and shutdown system performance, and radioactive material release pathways.

Section 5 provides a description of typical BWR compliance with two examples of the results of the use of the evaluation methodology. The examples are for a typical BWR/4 and 6. These examples identify the variables associated with each variable type, the basis for the classification of each variable, and the design requirements focused on the environmental qualification requirements.

Section 6 provides application guidelines for plant specific evaluations. Included are guidelines for BWR/2, 3, and 5 plants and key plant differences that can have a significant impact on the variable selection, such as isolation condensers. Also included is a discussion of equivalent parameters that can be used in lieu of the example parameters for the typical plant. This section also contains a discussion of the design and qualification criteria in IEEE-497 relative to each plant's current licensing basis.

Section 7 provides a summary of RG 1.97 Revision 4 changes. Included are a comparision between RG 1.97 Revision 3 and the results provided in this report, identification of previously approved deviation from earlier versions of RG 1.97, changes to BWR Standard Technical Specifications for post accident monitoring (PAM) and additional changes which would be deviations from RG 1.97 Revision 3 for BWR Owners, supporting information for the changes from RG 1.97 Revision 3.

Section 8 provides the conclusions of the report.

Section 9 contains the references for this report.

1.3 Assumptions

To simplify the evaluations in this report, the following assumptions are made:

1. The following performance criteria for accident monitoring systems are in compliance with RG 1.97 Revision 4
    - Monitoring channel range.
    - Instrument accuracy.
    - Instrument response time.
    - Post event instrumentation duration.
    - Reliability goals.
    - Performance assessment documentation.
2. The following design criteria for accident monitoring systems are in compliance with RG 1.97 Revision 4 subject to agreement with Regulatory Position (6) for use of codes and standards or are in accordance with the plant's current licensing basis:
    - Single failure.
    - Common cause failure.
    - Independence and separation.
    - Isolation.
    - Information ambiguity.
    - Power supply.
    - Calibration.
    - Testability.
    - Direct measurement.
    - Control of access.
    - Maintenance and repair.
    - Minimizing measurements.

- Auxiliary support features.
- Portable instruments.
- Documentation of design criteria.

3. The seismic and environmental qualification is in compliance with RG 1.97 Revision 4 or is in accordance with the plant's current licensing basis consistent with the result of the use of the application methodology.

4. The following display criteria for accident monitoring systems are in compliance with RG 1.97 Revision 4 or is in accordance with the plant's current licensing basis:

   - Display characteristics including information characteristics, human factors, anomalous indications, and continuous vs. on-demand display.
   - Trend or rate information.
   - Display identifications.
   - Type of monitoring channel display.
   - Display location.
   - Information ambiguity.
   - Recording.
   - Digital display signal validation.
   - Display criteria documentation.

5. The quality assurance requirements for accident monitoring systems are in compliance with RG 1.97 Revision 4 or are in accordance with the plant's current licensing basis.

6. Current accident monitoring systems that meet the current plant licensing basis requirements are considered to be equivalent to compliance with RG 1.97 Revision 4.

1.4 Limitations

This report is based on the following limitations:

1. A plant specific analysis consistent with the current plant safety analysis and licensing design basis requirements is performed prior to implementation of a new set of accident monitoring variables.

2. Implementation of a new set of accident monitoring variables is to be made consistent with licensing commitments relative to plant modifications.

3. If a plant chooses to implement RG 1.97 Revision 4 using this application methodology, all accident monitoring variables identified must be included.

4. Accident monitoring requirements do not apply to fire protection, station blackout, or shutdown from outside the control room. Each of these programs has specific licensing basis requirements.

5. Compliance with this report is not a requirement of the BWROG. BWR licensees may choose to maintain their current licensing basis with respect to the guidance provided by their commitments to prior revisions to RG 1.97.

1.5 Safety Analysis and Licensing Design Basis Requirements

Implementation of a new accident monitoring program is highly dependent on the current safety analysis and licensing design basis requirements. For the purpose of the application methodology, the safety analysis is defined by the anticipated operational occurrences and accidents or other equivalent nomenclature used in the safety or accident analysis section of the updated final safety analysis report (UFSAR). This definition is consistent with the definition contained in IEEE-497 for "accident analysis licensing basis". Typical events considered a part of the safety analysis are identified in Section 4.1. License design basis requirements include all plant specific commitments with respect to accident monitoring that are documented in the UFSAR or other applicable license amendment documents.

2. STANDARD REQUIREMENTS

This section identifies the requirements of IEEE-497 with respect to the five types of accident monitoring systems variables. The requirements are subject to the assumptions and limitations identified in Section 1. As a result, the requirements for each type of variable are focused on the selection of variables for each variable type. In addition, consideration of the seismic and environmental qualification requirement is considered important because of the economic implications of implementing the application methodology.

2.1 Type A Variables

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.

2.2 Type B Variables

Type B variables are defined in IEEE-497 as those variables that provide primary information to the control room operators to assess the plant critical safety functions.

For BWRs, the critical safety functions are defined by the EPGs. These include:

- Reactor pressure vessel (RPV) control.
  - Reactivity control.
  - Pressure control.
  - Level control.

- Primary containment control.

In the accident monitoring systems application methodology, Type B variables are limited to those required by the operator to assess the critical safety functions and necessary to implement planned manually-controlled actions in the EPGs and plant specific EOPs to respond to anticipated operational occurrences, accidents, or achieve a safe shutdown condition.

2.3 Type C Variables

Type C variables are defined in IEEE-497 as those variables that provide extended range primary information to the control room operators to indicate the potential breach or the actual breach of the three fission product barriers. The fission product barriers are the fuel cladding, reactor coolant pressure boundary, and primary containment pressure boundary. These variables represent the minimum set of plant variables that provide the most direct indication of the integrity of the fission product barriers and provide the capability for monitoring beyond the normal operating range.

The selection of the appropriate variables is included in the accident monitoring systems application methodology.

2.4 Type D Variables

Type D variables are defined in IEEE-497 as those variables that are required in procedures and licensing design basis to:

- Indicate the performance of those safety systems and auxiliary supporting features necessary for the mitigation of design basis events.

- Indicate the performance of other systems necessary to achieve and maintain a safe shutdown condition.

- Verify safety system status.

For BWRs, these variables are associated with the systems assumed in the safety analysis (anticipated operational occurrences and accidents) to achieve a safe shutdown condition. Where applicable, a single failure is assumed in a mitigating system.

By definition, a safety system is a system that is relied upon to remain functional during and following design basis events to assure:

1. The integrity of the reactor coolant pressure boundary

2. The capability to shut down the reactor and maintain it in a safe shutdown condition; or

3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10CFR100.11.

For BWRs, the accidents that can result in significant offsite exposures are the following four design basis accidents:

1. Control rod drop accident.
2. LOCA.
3. Piping breaks outside of containment.
4. Fuel handling accident.

Systems required to mitigate other anticipated operational occurrences or accidents or achieve and maintain a safe shutdown condition are considered to be required systems, but not safety systems.

2.5 Type E 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. The selection of these variables is to include, but not be limited to, the following:

- Monitor the magnitude of releases of radioactive materials through the identified pathways (e.g., secondary safety valves and condenser air ejector).
- Monitor the environmental conditions used to determine the impact of releases or radioactive material through identified pathways (e.g., wind speed, wind direction, and air temperature).
- Monitor radiation levels and radioactivity in the plant environs.
- Monitor radiation levels and radioactivity in the control room and selected plant areas where access may be required for plant recovery.

2.6 Seismic and Environmental Qualification

IEEE-497 contains seismic and environmental qualification requirements for each type of variable. The following are the specific seismic and environmental qualification requirements contained in IEEE-497:

1. Type A – Instrument channels that are required for planned manual operator action, needed directly or indirectly as a result of a seismic event, are required to be seismically qualified. Instrument channels required for a planned operator action to terminate or mitigate an accident are required to be environmentally qualified for that accident's postulated environment at the installed location.
2. Type B – These instrument channels are required to be seismically qualified. These instrument channels are required to be environmentally qualified for that accident's postulated environment at the installed location. Environmental qualification is to

consider performance testing to the maximum process conditions, while subjected to the worst-case postulated accident environment.

3. Type C – These instrument channels are required to be seismically and environmentally qualified. Environment qualification is to consider performance testing to the maximum process conditions, while subjected to the worst-case postulated accident environment at the installed location of the equipment.

4. Type D – Instrument channels that are expected to be operable following a seismic event are to be seismically qualified. Instrument channels are required to be environmentally qualified for the particular accident's postulated environment at the installed location.

5. Type E – Instrument channels that monitor systems are not required to be environmentally or seismically qualified. If an instrument that is used to determine the magnitude of a radiological release meets the selection criteria for another variable type, then that channel is required to meet the qualification criteria for that variable type.

For BWRs, seismic qualification is only associated with the four design basis accidents:

- Loss of coolant accident (LOCA).
- Pipe breaks outside of containment.
- Control rod drop accident.
- Fuel handling accident.

For BWRs, the events that are associated with a harsh environment are:

- LOCA
- Piping breaks outside of containment.

For the evaluation of a LOCA and high energy pipe breaks outside of containment, the event definition is limited to break sizes greater than the capability of the normal makeup systems, For BWRs, the normal makeup system is defined as the reactor core isolation cooling system (RCIC). The flow of the RCIC is substantially greater than the leakage detection capability of the normal leak detection systems that monitor system leakage during normal plant operation. This means that the leak detection system is assumed to automatically isolate all high energy pipe breaks outside of the primary containment.

A harsh environment is an environment that is significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences (e.g., loss of offsite power). The harsh environment (pressure, temperature, humidity, and radiation) for LOCA is generally limited to equipment located inside the primary containment or secondary containment. The harsh environment (pressure, temperature, humidity, and radiation)

for piping breaks outside of containment is limited to equipment located in areas outside primary containment (i.e., equipment inside primary containment is not exposed to a harsh environment from pipe breaks outside of primary containment). In addition, for pipe breaks outside primary containment, the harsh environment is dependent on the location and characteristics of the high energy piping and the systems required to mitigate the consequences of the particular pipe break.

3. REGULATORY POSITIONS

RG 1.97 Revision 4 endorses IEEE-497 as an acceptable method for providing instrumentation to monitor variables subject to eight regulatory positions. The following are the regulatory positions and how this report complies with the position and how operating BWRs adopting this report would propose meeting the positions.

3.1 Regulatory Position (1)

If a current operating reactor licensee voluntarily converts to the criteria in Revision 4 of this guide, the licensee should perform the conversion on the plant's entire accident monitoring program to ensure a complete analysis. If the licensee voluntarily uses the criteria in Revision 4 of this guide to perform modifications that do not involve a conversion, the licensee should first perform an analysis to determine the complete list of accident monitoring variables and their associated types in accordance with the selection criteria in Revision 4.

The evaluation methodology used in the LTR ensures a complete analysis of a operating BWR plant's entire accident monitoring system for the BWR Fleet. It is expected that currently operating reactor licensees will principally use this analysis to perform modifications, but total conversion may be considered as a part of plant control room upgrades including use of digital systems. Section 4.0 contains the evaluation methodology which reviewed BWR accident analysis including anticipated operational occurrences as well as a check on EPGs to assure a generic complete analysis was performed.

3.2 Regulatory Position (2)

Modify the first sentence in the second paragraph of Clause 6.7, as follows: "Means shall be provided for validating instrument calibration during the accident."

The BWROG agrees with the change made to IEEE-497, which recognizes the difficulties with calibration in a post accident environment.

3.3 Regulatory Position (3)

The range criteria for Type C variables (paragraph 2 of Clause 5.1) should include the basis for the expanded ranges as follows: "The range for Type C variables shall encompass those limits that would indicate a breach in a fission product barrier. These variables shall have expanded ranges and a source term that consider a damaged core (see NUREG-0660). For example, ..."

The BWROG agrees with the change.

3.4 Regulatory Position (4)

Modify the last sentence in Clause 4.1 as follows: "Type A variables include those variables that are associated with contingency actions that are within the plant licensing basis and may be identified in written procedures."

Regulatory position # 4 modifies the application of the term "contingency actions" in IEEE 497. The intent is to assure that the process used to select the actual list of variables is comprehensive and does not screen out actions that are within the plant licensing basis. Section 4.0 describes the accidents and AOOs which have been analyzed to determine the Type A variables. It is comprehensive in scope. In addition, the BWR EPGs have been used. The BWR EPGs are symptom not event based. They contain sections which address contingency actions which have been reviewed.

3.5 Regulatory Position (5)

The number of measurement points should be sufficient to adequately indicate the variable value.

The BWROG agrees.

3.6 Regulatory Position (6)

If the NRC's regulations incorporate an industry code or standard referenced in Clause 2 of IEEE Std. 497-2002, licensees and applicants must comply with that code or standard as set forth in the regulations. Similarly, if the NRC staff has endorsed a referenced code or standard in a regulatory guide, that code or standard constitutes an acceptable method for use in meeting the related regulatory requirement as described in the regulatory guide(s). By contrast, if a referenced code or standard has neither been incorporated into the NRC's regulations nor been endorsed in a regulatory guide, licensees and applicants may consider and use the information in the referenced code or standard, if appropriately justified, consistent with current regulatory practice.

All operating BWRs have made commitments to RG 1.97 Revision 2 or 3 which the NRC has reviewed and accepted including agreements on deviations from requirements. Included within RG 1.97 Revision 2 or 3 are codes and standards which the NRC has endorsed as an acceptable method for use. IEEE-497 was approved in May 2002. Consistent with the development of IEEE Standards, it is based on the latest codes and standards.

While no operating plant fully complies with the standards referenced in RG 1.97, Revision 4, the NRC has previously approved each plant's commitments to the design and qualification topics covered by the referenced standards as part of previous license submittals regarding accident monitoring instrumentation. For operating plants to use RG 1.97 Revision 4 consistent with the agreements in Regulatory Position (1) the plant would expect to use their existing commitments to RG 1.97 Revision 2 or 3 for codes and standards which have been accepted.

The following would be the expected design and qualification criteria to be used by operating BWRs as reconciled to codes and standards referenced in IEEE-497

- Independence and Separation (Section 6.3) and Isolation (section 6.4) - Both sections of IEEE-497 reference the requirements of IEEE 384-1992. Current plants meet the electrical separation, independence, and isolation requirements contained in IEEE 279. These plants were licensed before IEEE 384-1992 was issued and while they do not fully comply with the requirements contained in IEEE 384-1992 they do provide NRC approved provisions.

- Power Supply (Section 6.6) - This section of IEEE 497 states that the requirements of IEEE 308-1991 be met for Class 1E power supplies. Current plants meet the requirements for Class 1E power that were applicable when the plants were licensed (i.e., earlier revisions of IEEE 308). These plants were licensed before IEEE 308-1991 was issued and do not fully comply with the requirements contained in IEEE 308-1991.

- Environmental and Seismic Qualification (Sections 7.1 through 7.4) - These sections of IEEE 497 state that the requirements of IEEE 344-1987 and IEEE 323-1983 must be met. Current plants meet the environmental and seismic qualification requirements of IEEE 297 and 10CFR 50.49. Alternates to IEEE 344 and IEEE 323 approved include use of Seismic Qualification Utility's Group (SQUG) methodology for seismic qualification and DOR guidelines for environmental qualification.

- Human Factors (Section 8.1.2) - This section states that the requirements of IEEE 1023-1988, IEEE 1289-1998, and ISO 9241-3-1992 be met. Current plants meet the human factors requirements contained in NUREG 0737, supplement 1, since this was the requirement imposed as a result of the accident at TMI or was the latest requirement at the time of licensing. Plants may not fully comply with IEEE 1023-1988, IEEE 1289-1998, and ISO 9241-3-1992. These standards were issued after licensing commitments to human factors reviews were made, but they do comply with NRC accepted standards.

- Quality Assurance (Section 9) - This section requires use of ASME NQA-1-2001. All current plants meet the quality assurance requirements of Appendix B of 10CFR 50. However, not all current plants have upgraded from previous industry standards (i.e., ANSI N45) for quality assurance to ASME NQA-1-2001.

Each operating plant which choses to use RG 1.97 Revision 4 would be expected to review their RG 1.97 commitments with respect to codes and standard and perform modifications in accordance with such commitments. As part of NRC's review of this LTR we request agreement on the proposed use of codes and standards.

3.7 Regulatory Position (7)

Modify paragraph (c) of Clause 5.4, as follows: "The operating time for Type C variable instrument channels shall be at least 100 days or the duration for which the measured variable is required by the plant's LBD."

The BWROG agrees.

3.8 Regulatory Position (8)

Modify Clause 5.4 to replace the term "post-event operating time" with "operating time."

The BWROG agrees.

4. EVALUATION METHODOLOGY

This section provides a discussion of the application methodology used to determine the accident monitoring variables consistent with the requirements of RG 1.97 Revision 4 and IEEE-497. Because of the differences in the requirements for each of the five variable types, a different methodology is necessary for each variable type. Included is the methodology for determining the seismic and environmental qualification requirements. The application methodology for each variable type is provided in the following.

4.1 Type A Variables

Type A variables provide the operators with the primary information necessary to take the manual actions credited in the safety analysis. In the safety analysis, a number of different events require manual operator action in order to safely shut down the plant and assure continuity of decay heat removal. This section provides the following information:

- Safety analysis considerations that include the required actions necessary to assure that all safety functions are successfully accomplished.
- The events considered a part of the safety analysis and the basis for their selection.
- The treatment of single failures in the safety analysis and the impact of single failures on the information required by the operator.
- Identification of the planned manually controlled actions required to safety shut the plant down and assure continuity of decay heat removal.
- Identification of the specific parameters that provide the primary information used by the operator to take the planned manually controlled actions for which no automatic control is provided and that are required to mitigate the consequences of events analyzed in the safety analysis.

4.1.1 Safety Analysis Considerations

The BWR safety analysis is performed to demonstrate that there is no undue risk to the health and safety of the public and to demonstrate there is defense in depth.

To demonstrate that the risk to the public is acceptably low, the safety analysis is performed to demonstrate conformance to a set of event acceptance limits based on a qualitative assessment of event probability. In this process, a wide spectrum of events are identified and evaluated. Events assessed as having a relatively high probability of occurrence are required to satisfy a very conservative set of event acceptance limits. Lower probability events are required to meet a less restrictive yet conservative set of event acceptance limits.

Defense in depth is accomplished by providing barriers to fission product release to the environment. The fission product barriers identified in RG 1.97 Revision 4 and IEEE-497 include:

1. The fuel matrix and fuel cladding.

2. Reactor coolant pressure boundary.

3. Primary containment.

Consistent with safety analysis objectives, a set of required actions has been identified that enables the required operator manual actions to be identified on a consistent basis. In the safety analysis process for a specific event, it is assumed, that if the necessary required actions for that event are completed in a timely manner, the safety analysis results are acceptable. This assumption is demonstrated as being acceptable by the analysis of the limiting conditions for the event documented in the UFSAR. The specific required actions considered in the safety analysis are:

1. Reactor shutdown.

2. Pressure relief.

3. Core cooling.

4. Reactor vessel isolation.

5. Rod movement block.

6. Establish and maintain primary containment.

7. Establish and maintain secondary containment.

8. Control room habitability.

For the purpose of the safety analysis, these required actions cover the critical safety functions associated with the EOPs.

4.1.2 Events Considered

For the purpose of the safety analysis, safety analysis events can be separated into two general categories that reflect their estimated probability of occurrence of the initiating event based on engineering. Because of the significant differences in probability of occurrence, different event acceptance limits are applied. The two categories of events are:

1. Anticipated operational occurrences.

2. Accidents.

Anticipated operational occurrences are defined as those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

To select the anticipated operational occurrences, eight nuclear system parameter variations are considered as possible initiating causes of threats to the reactor core, fuel, and reactor coolant pressure boundary. These parameter variations were established during the development of the BWR safety analysis process and are consistent with the current safety analysis and reload analysis process for typical BWRs. The parameter variations are as follows:

1. Decrease in reactor coolant temperature.

2. Increase in reactor coolant temperature.

3. Increase in reactor pressure.

4. Decrease in reactor coolant flow rate.

5. Increase in reactor coolant flow rate.

6. Reactivity and power distribution anomalies.

7. Increase in reactor coolant inventory.

8. Decrease in reactor coolant inventory.

Accidents are postulated events that effect one or more of the barriers to the release of radioactive material to the environment. These events are not expected to occur during the life of the plant, but are used to establish the design basis for many systems. Accidents have the potential for releasing radioactive material as follows:

- From the fuel with the nuclear system process barrier, primary containment, and secondary containment initially intact.

- Directly to the primary containment.

- Directly to the secondary containment with the primary containment initially intact.

- Directly to the secondary containment with the primary containment not intact.

- Outside the secondary containment.

This categorization approach and the events within each category are generally applicable to all BWRs. However, there applicability needs be confirmed on a plant specific basis because there are differences in licensing commitments among the various plants.

For the purposes of developing the application methodology, the events considered in each category are identified in Table 3-1. Table 3-1 is based on a representative set of events that are associated with typical BWR safety analyses.

4.1.3 Treatment of Single Failures

A key component of BWR safety analyses is the treatment of single failures. In the safety analysis process, the single failure criterion is applied to anticipated operational occurrences and accidents to assure there is an appropriate level of redundancy. In developing the accident monitoring requirements it is assumed that the event occurs and there is a single failure in the systems necessary to perform the required actions. Based on this assumption, the necessary information for planned manually-controlled actions can be identified.

The NRC single failure definition, as it applies to the safety analysis, is provided in the introduction to the General Design Criteria (GDC – Reference 3) and is specifically applied to multiple GDCs. The systems included in application of the single failure criterion in the GDCs are the onsite electric power supplies, protection systems, RHR systems, ECCS, containment heat removal systems, and cooling water systems.

The NRC defines a single failures as, "... an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions. Single failures of passive components in electric systems should be assumed in designing against a single failure."

The types of single failures considered in typical BWR safety analyses are:

- The opening or closing of any single valve. (A check valve is not assumed to close against normal flow.)

- The starting or stopping of any single component.

- The malfunction or maloperation of any single control device.

- Any single electrical failure.

The single failure requirements for anticipated operational occurrences and accidents in the BWR safety analysis process are typically applied as follows:

- For anticipated operational occurrences and accidents, the protection sequences within mitigation systems are to be single component failure proof. This requirement is in addition to any single-component failure or single operator error that is assumed as the event initiator. The requirement for assuming a single failure in the mitigation system adds a significant level of conservatism to the safety analysis. However, the event limits for anticipated operational occurrences and accidents are not changed by the application of an additional single-failure requirement.

- For anticipated operational occurrences, it is not necessary to assume a single failure in normal operating systems in addition to the failure assumed as the event initiator. The basic logic for this assumption is based upon the probability of occurrence of a double failure in normal operating systems, which is less than once per plant lifetime and exceeds the probability of occurrence definition for anticipated operational occurrences in the GDC.

- For accidents, single failures are considered consistent with plant specific licensing commitments (e.g., valve malfunctions for LOCA).

- Multiple (consequential) failures from a single failure (e.g., the unavailability of ac power to components because of a failure in the standby ac power system) are considered part of the single failure. Single failures are independently postulated in each operating unit or one failure is postulated in the common systems.

- For mitigation systems included in the safety analysis, single failures of active electrical and fluid components are assumed. Single failures in passive fluid components are treated consistent with plant-specific licensing commitments. More specifically, the only single failure in a passive fluid component typically considered in the plant design is long-term leakage in the ECCS suction piping following a LOCA.

- During required Technical Specifications surveillance testing, the single-failure criterion is not applied to the affected components or systems. This is consistent with component or system reliability assumptions that form the bases for the plant Technical Specifications.

- When complying with the limiting conditions for operation in the Technical Specifications, the single failure criteria is not applied to the affected components or systems. This is consistent with component or system reliability assumptions that form the bases for the plant Technical Specifications.

4.1.4 Planned Manually-Controlled Actions

In the safety analysis, planned manually-controlled actions are required for many anticipated operational occurrences and accidents. These planned manually-controlled actions are based on an evaluation of the specific events and includes the assumption that the appropriate information is available to the operator to take the assumed action.

For BWRs, the required planned manually-controlled actions assumed in the safety analysis anticipated operational occurrences and accidents are associated with long-term core cooling (following the initial automatic system initiation) and long-term decay heat removal. These actions are necessary to assure a safe shutdown with continuity of core cooling and long term decay heat removal. The other required actions are performed by systems that are automatically initiated.

To identify the required planned manually-controlled actions, all of the necessary required actions for each event in the safety analysis are determined and all systems required to perform the required actions are identified. Next, a determination is made if the system is automatically initiated or manually initiated by operator action. The manually initiated systems identified through this process are those that are required by the safety analysis to limit the suppression pool parameters at high reactor pressure to prevent excessive containment loads. The required manually initiated systems for a typical BWR/4 and 6 are:

- Safety/relief valves (manual depressurization).
- High Pressure Core Spray (HPCS – BWR/5 and 6 only), low pressure core spray (LPCS), or low pressure coolant injection (LPCI) (restore and maintain level following depressurization).
- Suppression pool cooling or alternate decay heat removal (limit pool temperature increase).

4.1.5 Parameters Required by Operator

The parameters required by the operator necessary for the required planned manually-controlled actions are based on the specific manual system initiation assumed in the safety analysis. The specific parameters used by the operator are dependent on the systems necessary to perform the required actions and the phenomena occurring during the event. These parameters are identified by determining what information is necessary for the operator to take the appropriate action.

For BWRs, the parameters that provide the primary information required for planned manually-controlled actions for which there is no automatic control provided are:

- Reactor water level.
- Reactor pressure.
- Drywell pressure.
- Suppression pool temperature.
- Suppression pool water level.

The specific values of these parameters that are used by the operators to perform the required planned manually-controlled actions are contained in the plant specific EPGs.

4.2 Type B Variables

Type B variables provide primary information to the control room operators to assess the plant critical safety functions. The critical safety functions are established by the plant safety analysis and are consistent with the EPGs.

This section provides the following information:

- The philosophy used in the development of the EPGs which are applicable to the selection criteria for all variables.
- Identification of the critical safety functions.
- The methodology used to determine the critical safety parameters.

4.2.1 Emergency Procedure Guideline Philosophy for BWRs

The BWR Emergency Procedure Guidelines (EPGs) are symptom based guidelines, thus their associated actions will cover both design basis events, as well as beyond design basis events. The Severe Accident Guidelines (SAGs) are transitioned to when adequate core cooling cannot be assured. The EPGs consist of four "top level guidelines" and six "contingencies". For the purposes of developing Type B variables, the six contingency guidelines are excluded. The four top-level guidelines are considered in developing the critical safety functions.

1. Reactor control
   - Reactivity control.
   - Pressure control.
   - Level control
2. Primary containment control
3. Secondary containment control
4. Radioactive release

4.2.2 Critical Safety Functions

To identify the critical safety functions, the BWROG evaluation methodology is limited to those functions required to protect the three primary fission product barriers. These are:

1. The fuel matrix and fuel cladding.

2. Reactor coolant pressure boundary.

3. Primary containment.

The fuel matrix consists of sintered uranium dioxide pellets that retain a very high percentage of the fission product in the fuel matrix. As long as the fuel rods remain cooled, only the small fraction of fission products contained in the gap between the fuel cladding and fuel pellets or in the plenum of the fuel rods is available for release should the fuel cladding fail. The zircaloy fuel cladding provides the first mechanical barrier to the release of fission products. For anticipated operational occurrences, it is required that specified acceptable fuel design limits be satisfied. By satisfying this requirement, no fuel failures are predicted. Therefore, this barrier is assumed to be maintained intact. For certain accidents, this barrier may be predicted to fail due to specific challenges if certain limits are exceeded.

The reactor coolant pressure boundary provides a barrier to the release of primary coolant to the primary containment. Isolation valves on the reactor coolant pressure boundary are provided to isolate the reactor coolant pressure boundary from postulated pipe breaks outside of the primary containment. Therefore, this barrier is assumed to remain intact except for the postulated loss of coolant accident, which can involve the direct release of radioactive material to the primary containment.

The primary containment contains isolation features so that it provides 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.

Consistent with this approach, there are four critical safety functions:

1. Reactivity control.

2. Pressure control.

3. Level control.

4. Primary containment control.

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.

4.2.3 Critical Safety Parameters

The critical safety parameters used by the operator assess the critical safety functions are those that are used to initiate planned manually-controlled actions in the EPGs (EPG entry conditions) in response to anticipated operational occurrences and accidents or to attain a safe shutdown

condition. These specific parameters used by the operator are dependent on the phenomena occurring. The parameters are determined from the EPGs.

For BWRs, the critical safety parameters are:

- Reactor power/neutron flux (reactivity control).
- Reactor water level (level control).
- Reactor pressure (pressure control).
- Suppression pool temperature (containment control).
- Suppression pool water level (containment control).
- Drywell pressure (containment control).

There are other parameters used as entry conditions in the EPGs that are not considered to be critical safety parameters. These are:

1. Primary containment hydrogen concentration.
2. Primary containment oxygen concentration.
3. Drywell temperature.
4. Secondary containment control.
5. Radioactive release control.

For BWRs, these parameters are not considered to be required by the EPGs for plant critical safety functions.

Primary containment hydrogen levels are not considered risk significant for design basis events consistent with the NRC's revision to the Combustible Gas Control rule (10 CFR 50.44). Hydrogen and oxygen monitors are required to be maintained for "significant beyond design basis" events, but this equipment is not required to be safety related or environmentally qualified.

For design basis events, drywell temperature is not a limiting parameter. The limiting event for drywell temperature is a small break loss of coolant accident (LOCA) inside containment. For this event, reactor scram and core cooling are automatically provided. Drywell pressure and temperature increase, but they do not approach design limits. The only operator actions are based on drywell pressure and suppression pool temperature and level. No operator action is required based on drywell temperature.

Secondary containment control identifies entry conditions associated with reactor coolant leakage into the secondary containment. These entry conditions supplement the automatic

system isolation that is provided in the plant design. Automatic isolation will occur if the reactor coolant system leakage is excessive.

Radioactive release control identifies entry conditions associated with radioactive releases to areas outside the primary and secondary containments. These entry conditions supplement the automatic system isolation and standby gas treatment system initiation that is provided in the plant design. Automatic actions will occur if required.

4.3 Type C Variables

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. This section provides the following information:

- Identification of the fission product barriers.
- Basis for selection of the variables.
- The methodology used to determine the critical safety parameters.

These variables represent the minimum set of plant variables that provide the most direct indication of the integrity of the fission product barriers and provide the capability for monitoring beyond the normal operating range.

4.3.1 BWR Fission Product Barriers

Consistent with IEEE-497 and defined in Section 3.2.2, there are three fission product barriers provided for BWRs. The fission product barriers are:

1. The fuel matrix and fuel cladding.
2. Reactor coolant pressure boundary.
3. Primary containment.

4.3.2 Basis for Selection of Parameters

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 the capability for monitoring beyond the normal operating range. These parameters are selected based on an engineering evaluation of the design of the fission product barriers and the phenomena that would most likely be encountered due to a loss of barrier integrity during an accident.

4.3.3 Treatment of Normal Operating Leak Detection

The normal operating leak detection systems are not considered to provide accident indication of the integrity of any fission product barrier. These systems are provided to detect degradation of

piping systems so that action can be taken prior to the occurrence of an accident. The Technical Specification limits on leakage during normal operating condition provide assurance that appropriate actions can be taken before unacceptable degradation occurs.

4.3.4 Identification of Parameters

For BWRs, the parameters that provide the most direct indication of the integrity of the fission product barriers are typically:

1. Fuel cladding.
   - Reactor water level.
   - Off gas activity (monitoring performed by normal operating systems).
2. Reactor coolant pressure boundary.
   - Reactor water level.
   - Reactor pressure.
   - Drywell pressure.
   - Suppression pool water level.
   - Suppression pool temperature.
3. Primary containment.
   - Drywell pressure.
   - Suppression pool water level.
   - Suppression pool temperature.

4.4 Type D Variables

Type D variables provide information to the control room operators to:

- Indicate the performance of those required systems and auxiliary supporting features necessary for the mitigation of anticipated operational occurrences and accidents.
- Indicate the performance of other systems necessary to achieve and maintain a safe shutdown condition.
- Verify system status.

This section provides the following information:

- Process for the identification of required systems, safe shutdown systems, and auxiliary support functions.
- Basis for the selection of the parameters.
- Treatment of normal operating systems.
- Process for environmental determination.
- Treatment of isolation valve position switches.

4.4.1 Identification of Required Systems, Shutdown Systems, and Auxiliary Support Features

This section describes a process that can be used to identify the required systems, safe shutdown systems, and auxiliary support features for BWRs consistent with the requirements of IEEE-497. This process is generally applicable to all BWR product lines; however, it must be implemented on a plant specific basis because of the differences between individual plant designs.

4.4.1.1 Required Systems

Required systems are those systems relied upon to remain functional during and following anticipated operational occurrences and accidents to demonstrate that the applicable event limits are satisfied. To identify required systems, all events in the safety analysis are evaluated in a systematic and comprehensive manner. In this process, the entire duration of the event is evaluated from the spectrum of possible initial conditions until planned operation is resumed or a stable operating state is attained. Planned operation is considered as being resumed when normal operating procedures are being followed and the plant parameters and equipment being used are identical to those used in any defined planned operating state consistent with the allowable operating modes and operating envelope. A stable operating condition is defined as the completion of all required actions consistent with the EOPs and a stabilization of the plant parameters such that there is no need for further operator action based on the EOPs.

Required systems are identified as being required only if there is a unique requirement for them as being necessary to satisfy the required actions. If a normal operating system that was operating prior to the event (during planned operation) is to be employed in the same manner during the event and if the event did not affect the operation of the system, then the system is not considered a unique system requirement. A unique requirement arises only when the analysis of the event demonstrates that a system in addition to the normal operating systems is required for conformance to the event limits.

For typical BWRs, the required actions and systems assumed in the analysis of anticipated operational occurrences and accidents are identified in Table 3-2. It should be noted that for the

core cooling required action, the core cooling sequence is sometimes separated into an initial and long-term set of requirements. For these events, this situation exists because it is necessary to reach a stable shutdown condition. To reach this condition it may be necessary to depressurize the reactor and remove decay heat from the suppression pool to avoid exceeding limits on the suppression pool.

Based on this evaluation, a typical set of required systems has been determined. These systems are identified in Table 3-3.

4.4.1.2 Shutdown Systems

Shutdown systems are those systems in the primary success paths for the EOPs that are in addition to the required systems assumed in the mitigation of the anticipated operational occurrences and accidents. These systems are identified through a review of the EOPs. Based on a review of typical EOPs, a typical set of shutdown systems are identified in Table 3-3.

4.4.1.3 Auxiliary Support Functions

Auxiliary support functions are those systems or functions necessary to assure the proper functioning of the required systems and safety shutdown systems. Auxiliary support functions are determined from a review of the plant design. Based on a review of typical BWR plant designs, a typical set of auxiliary support functions is provided in Table 3-3.

4.4.2 Basis Selection of Parameters

Type D variables are selected to indicate the acceptable performance of the system or function and assess system status. These parameters are in addition to the Type A, B, or C parameters that provide the information to the operators necessary to assess the accomplishment of critical safety functions and perform any required planned manually-controlled actions.

The selection of Type D variables is highly dependent on the purpose and the design of the particular system or function. Typical parameters that indicate the successful functioning of a system are:

- System flow.
- System discharge temperature for systems that involve heat exchangers.
- Valve position for RPV or containment isolation valves.
- Water supply level and temperature.
- Damper position.
- Power supply status.

A list of Type D variables for a typical BWR/4 and 6 is provided in Section 4.

4.4.3 Treatment of Normal Operating Systems

Normal operating systems provide a substantial capability for mitigating the consequences of anticipated operational occurrences and accidents. However, they are not required for the mitigation of any accident that can challenge the off-site radiological exposure guidelines or that is associated with a harsh environment. Normal operating systems have numerous indications of acceptable system performance located in the main control room, and the plant operators can easily determine their availability. Therefore, they are not considered Type D variables subject to the requirements of IEEE-497.

4.4.4 Environment Determination

Type D variables are required to be environmentally qualified for the particular accident's postulated environment at the installed location of the monitoring equipment. There are only two events that are associated with a harsh environment. These are:

1. LOCA (pipe breaks inside containment).

2. High energy pipe breaks outside (including steam system pipe breaks outside of primary containment and feedwater line breaks).

Further, the pipe break that creates the harsh environment cannot directly fail the mitigating system. For example, if the only failure that can create a harsh environment for the high pressure coolant injection (HPCI) flow indication is the failure of the HPCI steamline, then the flow indication does not require environmental qualification for a harsh environment.

4.4.5 Treatment of Isolation Valve Position Switches

The environmental qualification requirements for RPV and containment isolation valve position switches is particularly complex. This situation occurs because of the different functions or the valves for different accidents.

The RPV isolation valves are valves on lines connected to the reactor coolant pressure boundary (including main steamline isolation valves – MSIVs) are required to isolate pipe breaks outside of containment and provide isolation of the containment for LOCA. This result in the requirement to consider both events relative to environmental qualification of RPV isolation valve position switches.

Containment isolation valves are connected to the primary containment, but not the reactor coolant pressure boundary, and are only required to provide isolation of the containment for a LOCA. Containment isolation valves are not required for pipe breaks outside of containment because the containment is isolated before fuel uncovery. As a result, there is no calculated fuel failure or requirement for containment isolation. This results in only the LOCA being considered for environmental qualification of containment isolation valve position switches.

Further, a number of RPV and containment isolation valves are normally closed and remain closed for the postulated LOCA and pipe breaks outside of containment. The operator has position indication on these valves during normal operation.

This situation leads to the following environmental qualification requirements for RPV and containment isolation valve position switches.

RPV Isolation Valve Position Switches

- Normally open RPV isolation valves inside containment – Position switches require environmental qualification for LOCA conditions. These position switches do not require environmental qualification for pipe breaks outside of primary containment because they are not exposed to that environment. Position switches on valves that are required to close only do not require qualification of accident radiation because their function is completed prior to any significant exposure.

- Normally open RPV isolation valves outside containment – Position switches require environmental qualification for the break of its system piping outside of containment. These position switches do not require environmental qualification for a LOCA because they close and remain closed and are not exposed to that environment. Position switches do not require qualification of accident radiation because their function is completed prior to any significant exposure.

- Normally closed RPV isolation valves that require opening for a LOCA or pipe breaks outside of containment – Position switches require environmental qualification for LOCA or pipe break outside of containment conditions at their installed location.

- Normally closed RPV isolation valves that do not require opening for either a LOCA or pipe break outside of containment – Position switches do not require environmental qualification.

Containment Isolation Valves

- Normally open containment isolation valves inside containment – Position switches require environmental qualification for LOCA conditions. Position switches on valves that are required to close only do not require qualification of accident radiation because there function is completed prior to any significant exposure.

- Normally open containment isolation valves outside of containment that are required to close and remain closed – Position switches do not require environmental qualification. These position switches do not require environmental qualification for a LOCA because they are not exposed to that environment.

- Normally closed containment isolation valves inside containment or outside containment that require opening for a LOCA – Position switches require environmental qualification for LOCA conditions at their installed location.

- Normally closed containment isolation valves inside containment that do not require opening for a LOCA – Position switches do not require environmental qualification.

4.5 Type E Variables

Type E variables provide information to be used in determining the magnitude of the release of radioactive material and continually assessing such releases. The selection of these variables is to include, but is not limited to, the following:

- Monitor releases of radioactive materials through the identified pathways.
- Monitor the environmental conditions used to determine the impact of releases.
- Monitor radiation levels and radioactivity in the plant environs.
- Monitor radiation levels and radioactivity in the control room and selected plant areas where access may be required for plant recovery.

This section provides the following information:

- Identification of release pathways
- Selection of parameters.

4.5.1 Release Pathways

For BWR accidents involving a potential significant amount of radioactive material release, there are basically four release pathways. These are:

1. Directly to the primary containment with the primary containment and secondary containment intact (i.e., LOCA). The pathway is leakage through the primary containment to the secondary containment with a release from the standby gas treatment system to the reactor building vent or offgas stack.

2. Directly to the secondary containment with the primary containment open or intact (i.e., fuel handling accident). The pathway is through the standby gas treatment system to the reactor building vent or offgas stack.

3. Directly through the environment (i.e., high energy pipe breaks outside primary containment. There is no specific pathway.

4. To the condenser or offgas system (i.e., control rod drop accident). The pathway may be to the environment from turbine building through the main condenser or through the offgas system to the reactor building vent or offgas stack.

4.5.2 Selection of Parameters

Based on typical BWR plant designs and the definition of Type E variables in IEEE-497, the following parameters are generally considered Type E variables:

- Containment radiation level.
- Reactor building area radiation level.
- Secondary containment release point radiation level.
- Offgas system release point radiation level.
- Wind direction.
- Wind speed.
- Ambient air temperature.
- Plant environs radiation monitors.
- Control room air inlet radiation monitors.
- Control room area radiation monitors.

Table 4-1 – Safety Analysis Events

ANTICIPATED OPERATIONAL OCCURRENCES

Decrease in Reactor Coolant Temperature

Loss of Feedwater Heating

Inadvertent RHR Shutdown Cooling Operation

Inadvertent HPCI or HPCS Startup

Inadvertent RCIC Startup

Increase in Reactor Coolant Temperature

Failure of RHR Shutdown Cooling

Increase in Reactor Pressure

Pressure Regulator Failure Closed

Generator Load Rejection

Turbine Trip

MSIV Closures

Loss of Condenser Vacuum

Decrease in Reactor Coolant Flow Rate

Recirculation Pump Trip

Recirculation Flow Controller Failure – Decreasing Flow

Increase in Reactor Coolant Flow Rate

Abnormal Startup of Idle Recirculation Pump

Recirculation Flow Controller Failure with Increasing Flow

Reactivity and Power Distribution Anomalies

Rod Withdrawal Error

Control Rod Maloperation

Decrease in Reactor Coolant Inventory

Inadvertent Safety/Relief Valve Opening

Pressure Regulator Failure – Open

Loss of AC Power

Increase in Reactor Coolant Inventory

Feedwater Controller Failure – Maximum Demand

Table 4-1 – Safety Analysis Events

ACCIDENTS

Control Rod Drop Accident

Loss of Coolant Accident

Steam System Piping Break Outside Containment

Fuel Handling Accident

Misplaced Bundle Accident

Pressure Regulator Failure – Downscale (BWR6)

Recirculation Pump Seizure

Recirculation Pump Shaft Break

Feedwater Line Break – Outside Containment

Failure of Small Lines Carrying Primary Coolant Outside Containment (Instrument Line Break)

Radioactive Waste System Leak or Failure

Liquid Radioactive System Failure

Postulated Radioactive Releases Due to Liquid Radwaste Tank Failure

Spent Fuel Cask Drop Accidents

Table 4-2 – Systems Assumed in the Safety Analysis

Event	Required Action	Systems Assumed
Anticipated Operational Occurrence		
Decrease in Reactor Coolant Temperature		
Loss of Feedwater Heating	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	RPV Isolation	Not Required
Inadvertent RHR Shutdown Cooling Operation	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	RPV Isolation	Not Required
Inadvertent HPCI/HPCS or RCIC Startup	Reactivity Control	Neutron Monitoring System (Inadvertent HPCI Start (BWR/3-4))
		Reactor Protection System
		Control Rod Drive System
	Pressure Control	Normal Operating Systems
		Safety/Relief Valves (Inadvertent HPCI Start (BWR/3-4))
	Initial Core Cooling	Normal Operating Systems
		RCIC (Inadvertent HPCI start (BWR/3-4))
		HPCI (Inadvertent HPCI start (BWR/3-4))
	Long Term Core Cooling	Automatic Depressurization System
		LPCS (Inadvertent HPCI start (BWR/3-4))
		LPCI (Inadvertent HPCI start (BWR/3-4))
		Suppression Pool Cooling (Inadvertent HPCI start (BWR/3-4))
	RPV Isolation	Not Required

Table 4-2 – Systems Assumed in the Safety Analysis

Event	Required Action	Systems Assumed
Increase in Reactor Coolant Temperature		
Failure of RHR Shutdown Cooling	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
	Pressure Control	Safety/Relief Valves
	Initial Core Cooling	RCIC
		HPCS (BWR/5-6)
		HPCI (BWR/3-4)
	Long Term Core Cooling	Automatic Depressurization System
		LPCS
		LPCI
		Suppression Pool Cooling
	RPV Isolation	MSIVs
		Shutdown Cooling Isolation Valve Closure
Increase in Reactor Pressure		
Pressure Regulator Failure – Closed	Reactivity Control	Normal Operating Systems
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	RPV Isolation	Not Required
Generator Load Rejection with Bypass	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
		End of Cycle – Recirculation Pump Trip
	Pressure Control	Safety/Relief Valves
	Core Cooling	Normal Operating Systems
	RPV Isolation	Not Required

Table 4-2 – Systems Assumed in the Safety Analysis

Event	Required Action	Systems Assumed
Generator Load Rejection without Bypass	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
		End of Cycle – Recirculation Pump Trip
	Pressure Control	Safety/Relief Valves
	Initial Core Cooling	Normal Operating Systems
	Long Term Core Cooling	Automatic Depressurization System
		LPCS
		LPCI
		Suppression Pool Cooling
	RPV Isolation	Not Required
Turbine Trip with Bypass	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
		End of Cycle – Recirculation Pump Trip
	Pressure Control	Safety/Relief Valves
	Core Cooling	Normal Operating Systems
	RPV Isolation	Not Required
Turbine Trip without Bypass	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
		End of Cycle – Recirculation Pump Trip
	Pressure Control	Safety/Relief Valves
	Initial Core Cooling	Normal Operating Systems
	Long Term Core Cooling	Automatic Depressurization System
		LPCS
		LPCI
		Suppression Pool Cooling
	RPV Isolation	Not Required

Table 4-2 – Systems Assumed in the Safety Analysis

Event	Required Action	Systems Assumed
Closure of All MSIVs	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
	Pressure Control	Safety/Relief Valves
	Initial Core Cooling	RCIC (Plants with Turbine Driven Feedwater Pumps)
		HPCS (BWR/5-6 Plants with Turbine Driven Feedwater Pumps)
		HPCI (BWR/3-4 Plants with Turbine Driven Feedwater Pumps)
		Normal Operating Systems (Plants with Motor Driven Feedwater Pumps)
	Long Term Core Cooling	Automatic Depressurization System
		LPCS
		LPCI
		Suppression Pool Cooling
	RPV Isolation	Occurs as Part of the Initiating Event
Closure of One MSIV	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	RPV Isolation	Not Required

Table 4-2 – Systems Assumed in the Safety Analysis

Event	Required Action	Systems Assumed
Loss of Condenser Vacuum	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
		End of Cycle – Recirculation Pump Trip
	Pressure Control	Safety/Relief Valves
	Initial Core Cooling	RCIC (Plants with Turbine Driven Feedwater Pumps)
		HPCS (BWR/5-6 Plants with Turbine Driven Feedwater Pumps)
		HPCI (BWR/3-4 Plants with Turbine Driven Feedwater Pumps)
		Normal Operating Systems (Plants with Motor Driven Feedwater Pumps)
	Long Term Core Cooling	Automatic Depressurization System
		LPCS
		LPCI
		Suppression Pool Cooling
	RPV Isolation	MSIVs
Decrease in Reactor Coolant Flow		
Trip of One Recirculation Pump	Reactivity Control	Normal Operating Systems
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	RPV Isolation	Normal Operating Systems
Trip of Both Recirculation Pumps	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
	Pressure Control	Safety/Relief Valves
	Initial Core Cooling	RCIC
		HPCS
		HPCI
	Long Term Core Cooling	Automatic Depressurization System
		LPCS
		LPCI
		Suppression Pool Cooling
	RPV Isolation	Not Required

Table 4-2 – Systems Assumed in the Safety Analysis

Event	Required Action	Systems Assumed
Recirculation Flow Controller Failure – Decreasing Flow	Reactivity Control	Normal Operating Systems
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	RPV Isolation	Normal Operating Systems
Increase in Reactor Coolant Flow Rate		
Abnormal Startup of Idle Recirculation Pump	Reactivity Control	Normal Operating System
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	RPV Isolation	Not Required
Recirculation Flow Controller Failure with Increasing Flow	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	RPV Isolation	Not Required
Reactivity and Power Distribution Anomalies		
Rod Withdrawal Error – Startup	Reactivity Control	Neutron Monitoring System
		Rod Pattern Controller (BWR/6)
		Reactor Protection System (BWR/3-5)
		Control Rod Drive System
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	RPV Isolation	Not Required
Rod Withdrawal Error – Power Operation	Reactivity Control	Rod Block Monitor (BWR/3-5)
		Rod Withdrawal Limiter (BWR/6)
		Control Rod Drive System
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	RPV Isolation	Not Required
Control Rod Maloperation	Covered by other rod withdrawal error evaluations	

Table 4-2 – Systems Assumed in the Safety Analysis

Event	Required Action	Systems Assumed
Decrease in Reactor Coolant Inventory		
Pressure Regulator Failure – Open	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
		End of Cycle – Recirculation Pump Trip
	Pressure Control	Safety/Relief Valves
	Initial Core Cooling	RCIC
		HPCS (BWR/5-6)
		HPCI (BWR/3-4)
	Long Term Core Cooling	Automatic Depressurization System
		LPCS
		LPCI
		Suppression Pool Cooling
	RPV Isolation	MSIVs
Inadvertent Safety/Relief Valve Opening	Reactivity Control	Reactor Protection System
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	RPV Isolation	Not Required
Loss of AC (Offsite) Power	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
	Pressure Control	Safety/Relief Valves
	Initial Core Cooling	RCIC
		HPCS (BWR/5-6)
		HPCI (BWR/3-4)
	Long Term Core Cooling	Automatic Depressurization System
		LPCS
		LPCI
		Suppression Pool Cooling
	RPV Isolation	MSIVs

Table 4-2 – Systems Assumed in the Safety Analysis

Event	Required Action	Systems Assumed
Loss of Feedwater Flow	Reactivity Control	Reactor Protection System
		Control Rod Drive System
	Pressure Control	Normal Operating Systems
	Initial Core Cooling	RCIC
		HPCS (BWR/5-6)
		HPCI (BWR/3-4)
	Long Term Core Cooling	Automatic Depressurization System
		LPCS
		LPCI
		Suppression Pool Cooling
	RPV Isolation	Not Required
Increase in Reactor Coolant Inventory		
Feedwater Controller Failure – Maximum Demand	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
		End of Cycle – Recirculation Pump Trip
	Pressure Control	Safety/Relief Valves
	Initial Core Cooling	RCIC
		HPCS (BWR/5-6)
		HPCI (BWR/3-4)
	Long Term Core Cooling	Automatic Depressurization System
		LPCS
		LPCI
		Suppression Pool Cooling
	RPV Isolation	Not Required

Table 4-2 – Systems Assumed in the Safety Analysis

Event	Required Action	Systems Assumed
Accidents		
Control Rod Drop Accident	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
	Pressure Control	Safety/Relief Valves
	Core Cooling	HPCS (BWR/5-6)
		HPCI (BWR/3-4)
		Automatic Depressurization System
		LPCS
		LPCI
		Suppression Pool Cooling
	RPV Isolation	Not Required
	Control Room Environmental Control	Main Control Room Environmental Control System
Loss of Coolant Accident	Reactivity Control	Reactor Protection System
		Control Rod Drive System
	Pressure Control	Safety/Relief Valves
	Core Cooling	HPCS (BWR/5-6)
		HPCI (BWR/3-4)
		Automatic Depressurization System
		LPCS
		LPCI
	Primary Containment	Containment Isolation Valves
		RPV Isolation Valves
		Suppression Pool Makeup System (BWR/6)
	Secondary Containment	Reactor Building Isolation and Standby Gas Treatment System
	Control Room Environmental Control	Main Control Room Environmental Control System

Table 4-2 – Systems Assumed in the Safety Analysis

Event	Required Action	Systems Assumed
Steam System Piping Break Outside Containment	Reactivity Control	Reactor Protection System
		Control Rod Drive System
	Pressure Control	Safety/Relief Valves
	Core Cooling	HPCS (BWR/5-6)
		HPCI (BWR/3-4)
		Automatic Depressurization System
		LPCS
		LPCI
	RPV Isolation	RPV Isolation Valves
	Control Room Environmental Control	Main Control Room Environmental Control System
Fuel Handling Accident	Reactivity Control	Normal Operating Systems
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	RPV Isolation	Not Required
	Secondary Containment	Reactor Building Isolation and Standby Gas Treatment System
	Control Room Environmental Control	Main Control Room Environmental Control System
Misplaced Bundle Accident	Reactivity Control	Normal Operating Systems
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	RPV Isolation	Not Required
Pressure Regulator Failure – Downscale (BWR/6)	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
	Pressure Control	Safety/Relief Valves
	Initial Core Cooling	Normal Operating Systems
	Long Term Core Cooling	Automatic Depressurization System
		LPCS
		LPCI
		Suppression Pool Cooling
	RPV Isolation	Not Required

Table 4-2 – Systems Assumed in the Safety Analysis

Event	Required Action	Systems Assumed
Recirculation Pump Seizure	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
		End of Cycle – Recirculation Pump Trip
	Pressure Control	Safety/Relief Valves
	Initial Core Cooling	RCIC
		HPCS (BWR/5-6)
		HPCI (BWR/3-4)
	Long Term Core Cooling	Automatic Depressurization System
		LPCS
		LPCI
		Suppression Pool Cooling
	RPV Isolation	Not Required
Recirculation Pump Shaft Break	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
		End of Cycle – Recirculation Pump Trip
	Pressure Control	Safety/Relief Valves
	Initial Core Cooling	RCIC
		HPCS (BWR/5-6)
		HPCI (BWR/3-4)
	Long Term Core Cooling	Automatic Depressurization System
		LPCS
		LPCI
		Suppression Pool Cooling
	RPV Isolation	Not Required

Table 4-2 – Systems Assumed in the Safety Analysis

Event	Required Action	Systems Assumed
Feedwater Line Break – Outside Containment	Reactivity Control	Reactor Protection System
		Control Rod Drive System
	Pressure Control	Safety/Relief Valves
	Core Cooling	HPCS (BWR/5-6)
		HPCI (BWR/3-4)
		Automatic Depressurization System
		LPCS
		LPCI
	RPV Isolation	RPV Isolation Valves
	Control Room Environmental Control	Main Control Room Environmental Control System
Failure of Small Lines Carrying Primary Coolant Outside Containment (Instrument Line Break)	Reactivity Control	Normal Operating Systems
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	Secondary Containment	Reactor Building Isolation and Standby Gas Treatment System
Radioactive Gas Waste System Leak or Failure	Reactivity Control	Neutron Monitoring System
		Reactor Protection System
		Control Rod Drive System
		End of Cycle – Recirculation Pump Trip
	Pressure Control	Safety/Relief Valves
	Initial Core Cooling	RCIC (Plants with Turbine Driven Feedwater Pumps)
		HPCS (BWR/5-6 Plants with Turbine Driven Feedwater Pumps)
		HPCI (BWR/3-4 Plants with Turbine Driven Feedwater Pumps)
		Normal Operating Systems (Plants with Motor Driven Feedwater Pumps)
	Long Term Core Cooling	Automatic Depressurization System
		LPCS
		LPCI
		Suppression Pool Cooling
	RPV Isolation	MSIVs

Table 4-2 – Systems Assumed in the Safety Analysis

Event	Required Action	Systems Assumed
Postulated Radioactive Releases Due to Liquid Radwaste Tank Failure	Reactivity Control	Normal Operating Systems
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	RPV Isolation	Not Required
Spent Fuel Cask Drop Accidents	Reactivity Control	Normal Operating Systems
	Pressure Control	Normal Operating Systems
	Core Cooling	Normal Operating Systems
	RPV Isolation	Not Required

Table 4-3 – Required Systems, Shutdown Systems, and Auxiliary Support Features

Required Systems

Neutron Monitoring System
Reactor Protection System
Control Rod Drive System
Safety/Relief Valves
RCIC
HPCI or HPCS
Automatic Depressurization System
LPCS
LPCI (a Mode of RHR)
Suppression Pool Cooling (a Mode of RHR)
Primary Containment and RPV Isolation Control System
EOC-RPT
MSIVs
RPV Isolation Valves
Containment Isolation Valves
Rod Block Monitor System (BWR/3-5)
Rod Withdrawal Limiter System (BWR/6)
Suppression Chamber or Containment to Drywell Vacuum Breaker System
Reactor Building to Suppression Chamber or Containment Vacuum Breaker System
Secondary Containment Isolation Dampers
Standby Gas Treatment System
Control Room Environmental Control System

Shutdown Systems

Shutdown Cooling System (a mode of RHR)
Standby Liquid Control System
ATWS-RPT

Auxiliary Support Systems

DC Power System
Auxiliary AC Power System
Standby AC Power System
Off-Site AC Power System
Equipment Area Cooling System
RHR Service Water System
Essential Service Water System
Essential Pneumatic Gas Supply
Suppression Pool
Ultimate Heat Sink

5. TYPICAL BWR COMPLIANCE

This section provides a discussion of a typical BWR/4 and 6 plant compliance with RG 1.97 Revision 4 and IEEE-497. The compliance for these two typical plants is provided in Tables 5-1 for a typical BWR/4 plant and in Table 5-2 for a typical BWR/6 plant. Because of the differences in nuclear steam supply system (NSSS) and containment designs, there are some differences in the application of IEEE-497.

Tables 4-1 and 4-2 contain 6 columns to address selected specific requirements of IEEE-497. These columns are:

1. Variable – This column identifies the specific variables required for accident monitoring.

2. Classification Basis – This column identifies the basis for the variable classification consistent with IEEE-497 and the evaluation methodology provided in Section 3. In some cases, there are multiple entries that reflect the variable may belong to several classification types. It should be noted that a variable that falls into more than one classification may require additional display channels to meet the different requirements for different variable types. For example, Type C variable require extended ranges that are not required for Type A or B variables.

3. Type – This column identifies the variable type consistent with the criteria identified in IEEE-497. Based on the classification basis, some variables can be associated with a number of different variable types. For these variables, the most restrictive variable type is identified. For example, if a variable can be Type A, B, or C, then the column would reflect each variable type.

4. Environmental Qualification (EQ) – Type A, B, C and D parameters are required to be environmentally qualified consistent with IEEE-497. Type E parameters are not required to be environmentally qualified consistent with IEEE-497.

5. Seismic Qualification (SQ) – Type A, B, and C parameters are required to be seismically qualified consistent with IEEE-497. Type E parameters are not required to be seismically qualified consistent with IEEE-497. Type D parameters are to be designed to be operable following a seismic event if the systems they monitor are required.

6. Comments – This column contains specific comments relative the specific variable.

Because these tables are typical, they are only intended for illustration purposes. Implementing changes the current plant accident monitoring system capability, a systematic review of the specific plant needs to be performed consistent with the guidance on evaluation methodology provided in Section 3. Further, significant plant modifications to the current plant accident monitoring program may require NRC approval prior to their implementation. It is anticipated that if the NRC approves this Licensing Topical Report based on the technical information in this report, implementation of the methodology approved by the NRC could be implemented

consistent with the provisions of 10CFR50.59 subject to plant reviews of their licensing commitments.

<table>
<tr><th colspan="6">Table 5-1 – Typical BWR/4 Accident Monitoring Variables</th></tr>
<tr><th>Variable</th><th>Classification Basis</th><th>Type</th><th>EQ</th><th>SQ</th><th>Comments</th></tr>
<tr><td>Reactor water level</td><td>Required for design basis events.
Level control.
Monitor fuel cladding integrity.</td><td>A, B, C</td><td>Y</td><td>Y</td><td rowspan="5">Type A parameters are plant specific and Category 1 in RG 1.97. From a BWR safety analysis perspective, these parameters are considered Type A consistent with the criteria identified in RG 1.97.</td></tr>
<tr><td>Reactor pressure</td><td>Required for design basis events.
Pressure control.
Monitor RCPB integrity.</td><td>A, B, C</td><td>Y</td><td>Y</td></tr>
<tr><td>Drywell pressure</td><td>Required for design basis events.
Primary containment control.
Monitor RCPB integrity.</td><td>A, B, C</td><td>Y</td><td>Y</td></tr>
<tr><td>Suppression pool temperature</td><td>Required for design basis events.
Primary containment control.
Monitor RCPB integrity.</td><td>A, B, C</td><td>Y</td><td>Y</td></tr>
<tr><td>Suppression pool water level</td><td>Required for design basis events.
Primary containment control.
Monitor RCPB integrity.</td><td>A, B, C</td><td>Y</td><td>Y</td></tr>
<tr><td>Reactor power/neutron flux</td><td>Reactivity control.
Safety system performance indication for reactor protection system and control rod drive system.</td><td>B. D</td><td>N</td><td>N</td><td>NRC Safety Evaluation Report on NEDO-31558, BWROG Proposed Neutron Monitoring System Post-Accident Monitoring Functional Criteria, February 2, 1993 approves the use of alternate criteria.</td></tr>
<tr><td>Drywell temperature</td><td>System performance indication for containment.</td><td>D</td><td>Y</td><td>N</td><td>Not required for seismic events.</td></tr>
</table>

Table 5-1 – Typical BWR/4 Accident Monitoring Variables

Variable	Classification Basis	Type	EQ	SQ	Comments
Control rod position	Safety system performance indication for reactor protection system and control rod drive system.	D	N	N	The rod position indication is a normal operating system that is not required to be seismically designed. Its function is completed before experiencing a harsh environment. Also, the proper functioning of the RPS and CRDs can be inferred from other parameters.
Safety/relief valve position indication	Safety system performance indication for safety/relief valves.	D	N	N	Backup instrument only. Not required to be seismically or environmentally qualified.
RCIC system flow	Required system performance indication for RCIC system.	D	N	N	RCIC is only required for anticipated operational occurrences. It is not associated with any events requiring environmental or seismic qualification.
HPCI system flow	Safety system performance indication for HPCI system.	D	Y	Y	
Condensate storage tank level	Required system performance indication for HPCI and RCIC.	D	N	N	Condensate storage tank is only required for anticipated operational occurrences. It is not associated with any events requiring environmental or seismic qualification.
RHR system flow	Safety system performance indication for all required RHR system modes.	D	Y	Y	RHR system flow and valve lineup used instead of flow indication for individual RHR operating modes.
RHR system valve position indications	System performance indication for all RHR safety and required system modes.	D	Y	Y	

Table 5-1 – Typical BWR/4 Accident Monitoring Variables					
Variable	**Classification Basis**	**Type**	**EQ**	**SQ**	**Comments**
RHR system heat exchanger outlet temperature.	Safety system performance indication for decay heat removal.	D	Y	Y	
LPCS system flow	Safety system performance indication for LPCS system.	D	Y	Y	
MSIV position switches	Safety system performance indication for RPV isolation.	D	Y	Y	
Cleanup system isolation valve position switches	Safety system performance indication for RPV isolation.	D	Y	Y	
Shutdown cooling system isolation valve position switches	Safety system performance indication for RPV isolation.	D	Y	Y	
Other RPV normally open isolation valve position switches on valves inside containment	Safety system performance indication for RPV isolation.	D	Y	Y	
Other RPV normally closed isolation valve position switches on valves inside containment that require opening for a LOCA	Safety system performance indication for the applicable system.	D	Y	Y	
Other RPV normally open isolation valve position switches on valves outside primary containment	Safety system performance indication for RPV isolation.	D	Y	Y	
Other RPV normally closed isolation valve position switches on valves outside primary containment that require opening for pipe breaks outside primary containment	Safety system performance indication for the applicable system.	D	Y	Y	

Table 5-1 – Typical BWR/4 Accident Monitoring Variables					
Variable	**Classification Basis**	**Type**	**EQ**	**SQ**	**Comments**
Other RPV normally closed isolation valve position switches on valves that do not require opening for either a LOCA or pipe breaks outside of containment	Safety system performance indication for containment isolation.	D	N	N	
Normally open containment isolation valve position switches on valves inside containment	Safety system performance indication for containment isolation.	D	Y	Y	
Normally closed containment isolation valve position switches on valves inside containment that require opening for a LOCA	Safety system performance indication for the applicable system.	D	Y	Y	
Normally closed containment isolation valve position switches on valves inside containment that require opening for a LOCA	Safety system performance indication for containment isolation.	D	Y	Y	
Containment isolation valve position switches on valves outside primary containment that require opening for a LOCA	Safety system performance indication for containment isolation.	D.	Y	Y	
Normally closed containment isolation valve position switches on valves inside or outside containment that do not require opening for a LOCA	Safety system performance indication for containment isolation.	D	N	N	
Secondary containment isolation damper position switches	Safety system performance indication for secondary containment.	D	Y	Y	

Table 5-1 – Typical BWR/4 Accident Monitoring Variables

Variable	Classification Basis	Type	EQ	SQ	Comments
Standby gas treatment system flow	Safety system performance indication for secondary containment.	D	Y	Y	
Control room isolation damper position	Safety system performance indication for control room environmental control system.	D	Y	Y	
Standby liquid control system pumps running	Required system performance indication for standby liquid control system.	D	N	N	Standby liquid control system is not associated with any events requiring environmental or seismic qualification.
Standby liquid control system tank level	Required system performance indication for standby liquid control system.	D	N	N	
DC power status	Safety system performance indication for DC power supply.	D	Y	Y	
AC power status	Safety system performance indication for AC power supply.	D	Y	Y	
Equipment area cooling system cooling water temperature	Safety system performance indication for equipment area cooling system.	D	Y	Y	
RHR service water system flow	Safety system performance indication for RHR service water system.	D	Y	Y	
Essential service water system flow	Safety system performance indication for essential service water system.	D	Y	Y	
Essential pneumatic gas supply pressure	Safety system performance indication for essential pneumatic gas supply system.	D	Y	Y	
Containment radiation level	Monitor identified pathway.	E	N	N	

Table 5-1 – Typical BWR/4 Accident Monitoring Variables					
Variable	**Classification Basis**	**Type**	**EQ**	**SQ**	**Comments**
Reactor building area radiation level in areas requiring access	Monitor identified pathway.	E	N	N	
Secondary containment release point radiation level	Monitor identified pathway.	E	N	N	
Secondary containment release point flow	Monitor identified pathway.	E	N	N	
Offgas system release point radiation level	Monitor identified pathway.	E	N	N	
Wind speed and direction	Monitor environmental conditions.	E	N	N	
Ambient air temperature	Monitor environmental conditions.	E	N	N	
Plant environs radiation monitors	Monitor plant environs.	E	N	N	
Control room area radiation monitors	Monitor control room	E	N	N	

Table 5-2 – Typical BWR/6 Accident Monitoring Variables					
Variable	**Classification Basis**	**Type**	**EQ**	**SQ**	**Comments**
Reactor water level	Required for design basis events. Level control. Monitor fuel cladding integrity.	A, B, C	Y	Y	Type A parameters are plant specific and Category 1 in RG 1.97. From a BWR safety analysis perspective, these parameters are considered Type A consistent with the criteria identified in RG 1.97.
Reactor pressure	Required for design basis events. Pressure control. Monitor RCPB integrity.	A, B, C	Y	Y	
Drywell pressure	Required for design basis events. Primary containment control. Monitor RCPB integrity.	A, B, C	Y	Y	
Suppression pool temperature	Required for design basis events. Primary containment control. Monitor RCPB integrity.	A, B, C	Y	Y	
Suppression pool water level	Required for design basis events. Primary containment control. Monitor RCPB integrity.	A, B, C	Y	Y	
Reactor power/neutron flux	Reactivity control. Safety system performance indication for reactor protection system and control rod	B. D	N	N	NRC Safety Evaluation Report on NEDO-31558, BWROG Proposed Neutron Monitoring System Post-Accident Monitoring Functional Criteria, February 2, 1993 approves the use of alternate criteria.
Drywell temperature	System performance indication for containment.	D	Y	N	Not required for seismic events.

Table 5-2 – Typical BWR/6 Accident Monitoring Variables

Variable	Classification Basis	Type	EQ	SQ	Comments
Control rod position	Safety system performance indication for reactor protection system and control rod drive system.	D	N	N	The rod position indication is a normal operating system that is not required to be seismically designed. Its function is completed before experiencing a harsh environment. Also, the proper functioning of the RPS and CRDs can be inferred from other parameters.
Safety/relief valve position indication	Safety system performance indication for safety/relief valves.	D	N	N	Backup instrument only. Not required to be seismically or environmentally qualified.
RCIC system flow	Required system performance indication for RCIC system.	D	N	N	RCIC is only required for anticipated operational occurrences. It is not associated with any events requiring environmental or seismic qualification.
HPCS system flow	Safety system performance indication for HPCS system.	D	Y	Y	
Condensate storage tank level	Required system performance indication for HPCS and RCIC.	D	N	N	Condensate storage tank is only required for anticipated operational occurrences. It is not associated with any events requiring environmental or seismic qualification.
RHR system flow	System performance indication for all RHR safety and required system modes.	D	Y	Y	RHR system flow and valve lineup used instead of flow indication for individual RHR operating modes.
RHR system valve position indications	Safety system performance indication for all required RHR system modes.	D	Y	Y	

Table 5-2 – Typical BWR/6 Accident Monitoring Variables					
Variable	**Classification Basis**	**Type**	**EQ**	**SQ**	**Comments**
RHR system heat exchanger outlet temperature.	Safety system performance indication for decay heat removal.	D	Y	Y	
LPCS system flow	Safety system performance indication for LPCS system.	D	Y	Y	
MSIV position switches	Safety system performance indication for RPV isolation.	D	Y	Y	
Cleanup system isolation valve position switches	Safety system performance indication for RPV isolation.	D	Y	Y	
Shutdown cooling system isolation valve position switches	Safety system performance indication for RPV isolation.	D	Y	Y	
Other RPV normally open isolation valve position switches on valves inside containment	Safety system performance indication for RPV isolation.	D	Y	Y	
Other RPV normally closed isolation valve position switches on valves inside containment that require opening for a LOCA	Safety system performance indication for the applicable system.	D	Y	Y	
Other RPV normally open isolation valve position switches on valves outside primary containment.	Safety system performance indication for RPV isolation.	D	Y	Y	
Other RPV normally closed isolation valve position switches on valves outside primary containment that require opening for pipe breaks outside primary containment..	Safety system performance indication for the applicable system.	D	Y	Y	
Other RPV normally closed	Not required for safety system	D	N	N	

Table 5-2 – Typical BWR/6 Accident Monitoring Variables

Variable	Classification Basis	Type	EQ	SQ	Comments
isolation valve position switches on valves that do not require opening for either a LOCA or pipe breaks outside of containment	performance indication.				
Normally open containment isolation valve position switches on valves inside containment	Safety system performance indication for containment isolation.	D	Y	Y	
Normally closed containment isolation valve position switches on valves inside containment that require opening for a LOCA	Safety system performance indication for the applicable system.	D	Y	Y	
Normally closed containment isolation valve position switches on valves inside containment that require opening for a LOCA	Safety system performance indication for containment isolation.	D	Y	Y	
Containment isolation valve position switches on valves outside primary containment that require opening for a LOCA	Safety system performance indication for containment isolation.	D	Y	Y	
Normally closed containment isolation valve position switches on valves in or outside containment that do not require opening for a LOCA	Not required for safety system performance indication.	D	N	N	
Secondary containment isolation damper position switches	Safety system performance indication for secondary containment.	D	Y	Y	
Standby gas treatment system flow	Safety system performance indication secondary	D	Y	Y	

Table 5-2 – Typical BWR/6 Accident Monitoring Variables					
Variable	**Classification Basis**	**Type**	**EQ**	**SQ**	**Comments**
	containment.				
Control room isolation damper position	Safety system performance indication for control room environmental control system.	D	Y	Y	
Standby liquid control system pumps running	Required system performance indication for standby liquid control system.	D	N	N	Standby liquid control system is not associated with any events requiring environmental or seismic qualification.
Standby liquid control system tank level	Required system performance indication for standby liquid control system.	D	N	N	
DC power status	Safety system performance indication for DC power supply.	D	N	N	
AC power status	Safety system performance indication for AC power supply.	D	Y	Y	
Equipment area cooling system cooling water temperature	Safety system performance indication for equipment area cooling system.	D	Y	Y	
RHR service water system flow	Safety system performance indication for RHR service water system.	D	Y	Y	
Essential service water system flow	Safety system performance indication for essential service water system.	D	Y	Y	
Essential pneumatic gas supply pressure	Safety system performance indication for essential pneumatic gas supply system.	D	Y	Y	
Containment radiation level	Monitor identified pathway.	E	N	N	
Reactor building area radiation level in areas requiring access	Monitor identified pathway.	E	N	N	

Table 5-2 – Typical BWR/6 Accident Monitoring Variables					
Variable	**Classification Basis**	**Type**	**EQ**	**SQ**	**Comments**
Secondary containment release point radiation level	Monitor identified pathway.	E	N	N	
Secondary containment release point flow	Monitor identified pathway.	E	N	N	
Offgas system release point radiation level	Monitor identified pathway.	E	N	N	
Wind speed and direction	Monitor environmental conditions.	E	N	N	
Ambient air temperature	Monitor environmental conditions.	E	N	N	
Plant environs radiation monitors	Monitor plant environs.	E	N	N	
Control room area radiation monitors	Monitor control room	E	N	N	

6. GUIDELINES FOR APPLICATION TO SPECIFIC PLANTS

This section provides a discussion of the application guidelines to specific plants based on the evaluation methodology in Section 3. Section 4 provides the result of the implementation of the evaluation methodology for typical BWR/4 and 6 plants. To implement the evaluation methodology and develop establish a plant specific set of accident monitoring variables, it is necessary to understand the differences between the BWR product lines.

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. The application to a specific plant is to be consistent with the plant's licensing design basis, including the requirements for environmental and seismic qualification requirements.

6.1 BWR Product Lines

The earliest BWRs (BWR/1s) were developmental in nature. These reactors were intended to demonstrate various design features that were to be incorporated into later designs. There are no operating BWR/1 plants in the US.

BWR/2s were the first large BWRs constructed and operated in the US. These plants incorporated the Mark I pressure suppression primary containment concept and are characterized by having five external recirculation loops, two LPCS systems, an ADS, a separate shutdown cooling system and containment spray/cooling system, and isolation condensers. The pressure relief system consists of spring safety valves and non-Code qualified relief valves.

The BWR/3 product line is characterized by a low power density core design that continued the use of the Mark I containment. It is the first product line to implement jet pumps (reducing the external recirculation loop requirement to two) that provided a floodable volume for ECCS flow. It incorporates an HPCI system as a high pressure makeup system that provided part of the overall small and intermediate break LOCA protection. Later BWR/3s incorporate the RCIC system to essentially replace the isolation condensers and a multifunction residual heat removal (RHR) system with the three primary modes (LPCI with loop selection logic, containment spray/cooling, and shutdown cooling). During The BWR/3 product line, the pressure relief system made a transition to dual function Code qualified self actuated safety/relief valves supplemented by spring safety valves.

The BWR/4 product line was a continuation of the BWR/3 product line except that a higher power core design was adopted. Later BWR/4s adopted the Mark II containment design and many BWR/4s implemented the LPCI modification that eliminated the loop selection logic. Some plants eliminated the use of spring safety valves and some plants initiated the use of dual function safety/relief valves that were incorporated in the design of subsequent product lines. The BWR/4 product line was the first to incorporate the low-low-low (Level 1) reactor water level initiation of the ADS and low pressure ECCS.

The BWR/5 product line is essentially the same power density as BWR/4 with continued use of Mark II containments. The BWR/5 product line ECCS incorporates an ADS, one LPCS system, LPCI (with three pumps injecting inside the core shroud), and HPCS system.

The BWR/6 product line incorporates a higher power density core design in Mark III containments. The ECCS is the same as BWR/5. Most of the systems remained functionally the same as BWR/5. The BWR/6 includes a number of changes that affect the safety analysis. These changes are primarily the implementation of a high reactor water level scram and recirculation pump trip and the rod pattern control system and rod withdrawal limiter in the rod control and information systems.

6.2 Application to BWR/2, 3 and 5

The application of the evaluation methodology to BWR/4 and 6 plants is relatively straight forward. Basically, the accident monitoring parameters identified in Tables 4-1 and 4-2 need to be modified consistent with the plant specific design and terminology. In addition, any unique plant specific licensing design basis requirements need to be recognized.

Application of the evaluation methodology to BWR/2, 3, and 5 plants is more complex. The strategy for these plants is dependent on the similarity between these product lines and the BWR/4 and 6 product lines.

For BWR/2 and 3 plants, the similarity to BWR/4 can be utilized recognizing the plant and product line differences. Most of the accident monitoring variables contained in Table 4-1, modified consistent with the plant specific design and terminology, are applicable. Modification of the Type D variables to reflect the differences between ECCS, containment and shutdown cooling, and isolation condensers, as applicable, is necessary. In addition, any plant specific licensing design basis requirements need to be recognized.

For BWR/5 plants, a hybrid of BWR/4 and 6 product line requirements can be used. The typical plant and containment related variables for BWR/4 contained in Table 4-1 are generally applicable. The typical ECCS related variables for BWR/6 contained in Table 4-2 are generally applicable. These variables need to be modified consistent with the plant specific design and terminology. In addition, any unique plant specific licensing design basis requirements need to be recognized.

6.3 Isolation Condensers

BWR/2 and early BWR3 plants incorporate isolation condensers. The primary function of the isolation condensers is to provide core cooling and remove decay heat for events that involve a loss of feedwater or the main heat sink. The isolation condensers are designed to take steam from the RPV, condense the steam, and return the condensate to the RPV. The initial functioning of the isolation condenser is accomplished by opening the condensate return valves.

An isolation condenser has a sufficient water inventory in the condenser shell to condense the steam produced by decay heat for a specified period of time. Typically a variety of highly reliable makeup water sources are provided to assure continued operation of the isolation

condensers. These makeup water sources can provide adequate makeup without reliance on the availability of offsite power. The isolation condensers are capable of maintaining a hot shutdown condition for an indefinite period of time.

In essence, isolation condensers in the safety analysis replace the functioning of the RCIC for the following set of design basis events, as applicable to the specific plant:

- Inadvertent HPCI Startup (BWR/3)
- Failure of RHR Shutdown Cooling
- Closure of All MSIVs
- Loss of Condenser Vacuum
- Trip of Both Recirculation Pumps
- Pressure Regulator Failure – Open
- Loss of AC (Offsite) Power
- Loss of Feedwater Flow
- Feedwater Controller Failure – Maximum Demand
- Recirculation Pump Seizure
- Recirculation Pump Shaft Break
- Radioactive Gas Waste System Leak or Failure

Because isolation condensers are not required for any of the four design basis accidents, they are considered a required system, not a safety system. The performance of the isolation condenser is indicated by Type B variables and the condensate return valve position and the isolation condenser shell water level. The condensate return valve position and the isolation condenser shell water level are considered Type D variable

6.4 Other Equivalent Variables

In some cases, the specific variables identified in Tables 4-1 and 4-2 may not be available in specific plant designs. In these cases, the identified variables need to be replaced by other equivalent parameters. Two specific examples are:

- System flow measurements.
- Safety/relief valve tailpipe temperature.

- Isolation damper position

If system flow measurement instrumentation is not available for accident monitoring, this variable can be replaced by an indication of pump running or pump discharge pressure along with the appropriate valve position indication. Another variable that may be considered is supply tank level. This type of other equivalent variable is used for the standby liquid control system performance monitoring.

Some plants may incorporate different safety/relief valve position indication monitoring concepts. These may include pressure switches in the tailpipes from the SRV to the suppression pool or acoustic monitors in the drywell. Either of these indications can replace the safety/relief valve tailpipe temperature monitors.

Some plants may use differential pressure measurement instead of isolation damper position. Differential pressure is a direct measure of the performance of heating, ventilating, and air conditioning systems.

6.5 Compliance with IEEE-497 Referenced Standards

No current operating plant fully complies with the standards referenced in IEEE-497 as discussed in Regulatory Position (6). However, the commitment to the set of standard in the current plant licensing basis is considered an acceptable alternative to the referenced standards. Further, the NRC has previously approved each plant's commitments to the design, qualification, and quality topics covered by the referenced standard as a part of previous license submittals regarding accident monitoring instrumentation. The current plant commitments to the following design, qualification, and quality standard are considered acceptable alternates to the standards referenced in IEEE-497:

1. Independence and Separation (IEEE-497 Section 6.3) and Isolation (IEEE-497 Section 6.4) – Both sections of IEEE-497 state that the requirements of IEEE 384-1992 must be met. However, many current operating plants were licensed before IEEE 384-1992 was issued and meet the electrical separation, independence, and isolation requirements contained in IEEE 279. The current license basis for independence, separation, and isolation are acceptable alternatives to IEEE 384-1992.

2. Power Supply (IEEE-497 Section 6.6) – This section of IEEE-497 states that the requirements of IEEE 308-1991 must be met for Class 1E power supplies. Current plants meet the requirements for Class 1E power that were applicable when the plants were licensed (i.e., earlier revisions of IEEE 308). The current license basis for Class 1E power supplies is an acceptable alternative to IEEE 308-1991.

3. Environmental and Seismic Qualification (IEEE-497 Sections 7.1 through 7.4) - These sections of IEEE-497 state that the requirements of IEEE 344-1987 and IEEE 323-1983 must be met. Many current plants meet the environmental and seismic qualification requirements of IEEE 297 and 10CFR 50.49. Alternates to IEEE 344 and IEEE 323 approved include use of Seismic Qualification Utility's Group (SQUG) methodology for seismic qualification and Division of Operating (DOR) guidelines for environmental

qualification. These approved approaches are acceptable alternatives to IEEE 344-1987 and IEEE 323-1983.

4. Human Factors (IEEE-497 Section 8.1.2) - This section of IEEE-497 states that the requirements of IEEE 1023-1988, IEEE 1289-1998, and ISO 9241-3-1992 be met. Current plants meet the human factors requirements contained in NUREG 0737, Supplement 1. Because this was the requirement imposed as a result of the accident at Three Mile Island or was the latest requirement at the time of licensing, the current license plant's license commitment for human factors is considered an acceptable alternative to IEEE 1023-1988, IEEE 1289-1998, and ISO 9241-3-1992.

5. Quality Assurance (IEEE-497 Section 9) - This section requires use of ASME NQA-1-2001. All current plants meet the quality assurance requirements of Appendix B of 10CFR 50. However, not all current plants have upgraded from previous industry standards (i.e., ANSI N45) for quality assurance to ASME NQA-1-2001. The current plant's license basis for quality assurance is an acceptable alternative to ASME NQA-1-2001.

7. SUMMARY OF REGULATORY GUIDE 1.97 REVISION 4 CHANGES

The analysis described in Section 4 of this report has been used to determine the BWR accident monitoring variables based on RG 1.97 Revision 4. A comparison has been made between RG 1.97 Revision 3 and the results provided in this report which is contained in Table A-2. RG 1.97 Revision 3 has not been updated since 1983. BWR Owners have processed both generic and plant specific deviations since Revision 3 was released to obtain acceptance on changes to plant commitments to RG 1.97. The majority of the changes identified by using RG 1.97 Revision 4 have been previously accepted as deviations from RG 1.97 Revision 3.

Additional changes which result from the use of RG 1.97 Revision 4 are the generic determination of Type A variables for the current BWR operating plants, changes to BWR improved Standard Technical Specifications for PAM instrumentation and additional changes which would be deviations from RG 1.97 Revision 3 for BWR Owners.. The following addresses BWR Owner implementation considerations including adoption of NRC approved deviations, changes to Technical Specifications and specific additional changes identified in the report.

7.1 NRC Approved Deviations To Regulatory Guide 1.97 Revisions 2 and 3

Included within the results are variables which have resulted in NRC approval of plant specific deviations. Justification for NRC approval of some of the deviations can be found in NRC's Standard Review Plan Section 7 which includes Table 1 to Branch Technical Position HICB-10-5. This table identifies the following:

- Drywell sump and drywell drain sump level – change to Category 3.
- Primary containment isolation valve position – eliminates need for redundancy.
- Radioactivity concentration or radiation level in circulating primary coolant – not required.
- Containment H_2 and O_2 concentration – range required.
- Suppression chamber and drywell spray flows – use of temperature and pressure as alternatives to flow.
- Standby liquid control system (SLCS) flow – use of pump discharge pressure and tank level as alternative.
- Reactor building or secondary containment area radiation – change to Category 2 for Mark III Containment design and Category 3 for Mark I & II containment design.
- Radiation exposure rate used for releases – change to Category 3.

NRC Standard Review Plan (Reference 5) Section 7 was last updated June 1997 for generically acceptable RG 1.97 deviations. NRC's next revision of the Standard Review Plan Section 7 is

expected to be based on RG 1.97 Revision 4 so additional generically approved deviations will not be reflected. If it was updated, the following would be expected to be included:

- H_2 monitors from Category 1 to Category 3 based on Combustible Gas Control Rule.
- O_2 monitors from Category 1 to Category 2 without EQ based on Combustible Gas Control Rule.
- Safety/relief valve position indication from Cat 2 to Cat 3 based on BWR NEDO-33160 A.
- Core thermocouples based on documentation of a BWR review of RG 1.97, July 1982 (Reference 8).
- Neutron Flux based on NEDO 31558 (Reference 4).

The NRC has approved other plant specific deviations beyond that contained in the Standard Review Plan. Additional reviews for applicability to a specific plant are recommended due to design differences and lack of a Topical Report or other documentation to support applicability to the currently operating BWR plants. The following is a list of plant specific approved deviations:

- Change reactor core isolation cooling (RCIC) flow from Type D Category 2 to Category 2 without EQ or Category 3. Using this report, the basis for the deviation would be that RCIC is only relied upon for Anticipated Operational Occurrences (AOOs) not in mitigating the consequences of an accident.
- Reactor building effluent stack monitor from Type E Category 2 to Category 3. RG 1.97 Revision 4 would support this change consistent with Type E variables requirements.
- RHR heat exchanger outlet temperature from Type D Category 2 to Category 3. Basis provided was that the outlet temperature is not essential for determining the performance of the heat exchanger and is secondary to other indications.
- MSIV pneumatic pressure indication instrumentation from Type D Category 2 to Category 2 without EQ. Basis provided is that MSIVs perform their safety function before there is a harsh environment and that design of MSIV uses both springs and pneumatics for closure.
- Cooling water temperature and flow to ESF system components from Category 2 to Category 3 with the instruments being used being ESF room temperature and ESW pump running instead of cooling water flow/temperature.

7.2 Technical Specifications

All BWR Owners have a section on PAM Technical Specifications, which is based on provisions of RG 1.97. Owners who have converted to improved Standard Technical Specifications (NUREG 1433 and 1434 – References 6 and 7) have a table (3.3.3.1-1) which defines their PAM instrumentation based on a list of instruments, which is to be supplemented by plant specific identification of RG 1.97 Type A instruments and RG 1.97 Category 1 non-Type A instruments specified in the plant's RG 1.97 NRC Safety Evaluation Report. The instruments contained in improved Standard Technical Specifications for Type B and C variables match the conclusions of this report with a few exceptions. The exceptions are drywell sump and drain level and neutron flux which, all Owners may have not included in their Technical Specifications based on NRC approved deviations. Primary containment radiation and containment isolation valve position are the new potential changes identified in this report. A proposed Technical Specification change via the Technical Specification Task Force traveler is anticipated to be developed and submitted separately based on the conclusions of this report.

Using RG 1.97 Revision 4 the following are the generic Type A variables. These variables have also been determined to meet the criteria of a Type B and C variable.

- Reactor water level.
- Reactor pressure.
- Drywell pressure.
- Suppression pool temperature.
- Suppression pool level.

In addition, this report concludes that neutron flux would be a Type B variable. Prior NRC agreements in a separate BWROG effort contained in NEDO-31558 (Reference 4) provided the basis for plant exclusion of neutron monitoring from Technical Specifications.

A review of selected BWR Owner Technical Specifications indicates differences in Type A variables within the currently operating BWR plants including the addition of drywell temperature and emergency diesel voltage as Type A variables. Drywell temperature is considered a Type D variable under RG 1.97 Revision 3, and the same conclusion is reached in this report.

7.3 Basis for Identified Changes

RG 1.97 Revisions 2 and 3, which all BWR Owners have committed to as part of initial licensing or as a result of commitments to NRC Generic Letter 82-33 Supplement 1 (Reference 9), include variables as Type B or C. Owner approved NRC deviations are expected to cover the majority of the differences between what is in RG 1.97 Revisions 2 and 3 and this report.

Using this report, all other variables listed in RG 1.97 Revision 2 and 3 are considered Type D (indicate performance of safety systems), Type E (magnitude of release of radioactive material), or are eliminated. The following are the substantive changes and supporting information for implementation;

7.3.1 Primary Containment Isolation Valve Position Indication

Primary containment isolation valve (CIV) position indications are high maintenance costs for BWR Owners due to the number of CIVs and the inclusion of harsh LOCA environmental qualification requirements which necessitate frequent replacements. A typical plant will have approximately 40 containment penetrations with automatic isolation valves. Thus up to 80 CIV position indication systems as RG 1.97 variables. To satisfy CIV position indication being prescribed as a Type B Category 1 variable, some Owners were required to make extensive modification to comply with RG 1.97 Revision 2 and 3. The conclusions of this report is that CIV position indication should be changed to a Type D variable as it is used to verify system status for the containment. Primary accident monitoring information for the containment would be provided by reactor pressure, reactor water level, drywell pressure, suppression pool level and suppression pool temperature, all of which are considered as Type A, B, and C variables.

Regardless of the change in classification from Type B to D, the CIV position indication is required to meet the postulated accident environment at the installed location and to provide operators with information on isolation and to demonstrate opening post accident if opening is assumed to be required in the plant safety analysis. It is expected that the postulated environment for certain outboard CIV position indication would be mild. Certain containment isolation valves and their position indication also provide isolation for the RPV, such as main steam isolation valves. These valves must also be designed for the consequence of postulated high energy line breaks, including pipe breaks outside containment. High energy line breaks outside containment effects are included in plant safety analysis. High energy line breaks do not result in the release of radiation and do not require closure of CIVs other than the RPV CIVs

7.3.2 Containment Radiation Monitors

The containment radiation monitor is included in RG 1.97 Revision 2 and 3 as a Type C (breech of barrier) Category 1 variable and is also included in PAM Technical Specifications. Evaluations based on RG 1.97 Revision 4 conclude the containment radiation monitor is not a post accident variable required by the plant safety analysis, the EPGs or a direct measurement of a radiological release. The purpose of the containment radiation monitor is to provide information for emergency action level (EAL) classifications. Containment high range radiation monitors were required to be installed in operating plants and plants under construction as a result of the TMI accident lessons learned, not because of reliance in BWR safety analysis. TMI action item II.F.1 (Reference 10) established the requirements for such monitors and RG 1.97 subsequently incorporated the monitor as a Category 1, Type C variable. The containment radiation monitors were included in RG 1.97 Revision 3 as a Type C Category 3 under "Reactor Coolant Pressure Boundary," which noted the purpose as "detection of breech; verification". Primary Containment Area Radiation – High Range is also included in RG 1.97 Revision 3 as a Type E Category 1 variable with the purpose listed as "detection of significant release: long term surveillance: emergency plan actuation". The containment radiation monitor does provide

backup information but is not a direct indicator of RPV integrity, so it not a Type C variable under the provisions of RG 1.97 Revision 4. RPV integrity under Revision 4 would be determined by RPV water level and RPV pressure based on BWR accident analysis and EPG provisions. Containment radiation monitors are considered a Type E variable, but it is not solely used for determining EALs as a reactor coolant system barrier. Cautions are contained in EALs in using the containment radiation barrier directly for decisions due to locations of the monitors and need to consider adjacent pipe radiation shine and other considerations. The EALs refer to drywell pressure, RPV water level, and indications of reactor coolant system leakage as other considerations in determining the EAL.

A change of the containment radiation monitor to a Type E variable is appropriate and, in addition, its removal from PAM Technical Specifications is recommended.

7.3.3 Safety/Relief Valve Position Indication System

The safety/relief valve position indication system is included in RG 1.97 Revision 2 and 3 as a Type D (safety system) Category 2 variable. This categorization is based on the assumption that safety/relief valve position is a key variable for providing detection of an accident and reactor coolant pressure boundary integrity indication of the main steam system. The BWROG has submitted and received NRC approval on an LTR (Reference 11) that provides the basis for relaxation of the accident monitoring requirements related to the safety/relief valve position indication system.

With respect to reactor coolant pressure boundary integrity, RPV pressure and suppression pool temperature in combination with other instruments (e.g., RPV water level, suppression pool level, and containment pressure) satisfy the RG 1.97 accident detection and boundary integrity indication requirements. This alternate instrumentation either meets or exceeds the RG 1.97 Category 2 criteria. Therefore, the safety/relief valve position indication can be reclassified as a Type D Category 3 variable.

Further, operator actions to mitigate the consequences of accidents are based on other RG 1.97 parameters. Safety/relief valve position indication only provides a confirmation of valve opening. This information is of secondary importance to operators following the EPGs or plant specific EOPs. Therefore, the change in safety/relief valve categorization is appropriate and consistent with NRC conclusions in their Safety Evaluation Report on Reference 11.

8. CONCLUSIONS

Based on the information provided in this report, the following conclusions have been reached:

1. IEEE-497 provides a means for identifying a comprehensive set of required accident monitoring variables.

2. A systematic evaluation methodology has been identified for BWRs that allows the systematic identification of the required accident monitoring variables in accordance with IEEE-497.

3. The evaluation methodology has been applied to typical BWRs to demonstrate its effectiveness in identifying an appropriate set of accident monitoring variables.

4. Current operating BWRs may be able to convert their current accident monitoring program to be in compliance with IEEE-497 consistent with their current licensing design basis.

5. The methodology provided in this report can be used as a basis for developing Technical Specification changes for the accident monitoring variables.

9. REFERENCES

1. IEEE Std. 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations."

2. Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," May 1983.

3. 10CFR50 Appendix A, "General Design Criteria."

4. NRC Safety Evaluation Report on NEDO-31558, "BWROG Proposed Neutron Monitoring System Post-Accident Monitoring Functional Criteria," February 2, 1993.

5. NUREG-0800, "Review of Safety Analysis Reports for Nuclear Power Plants."

6. NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4."

7. NUREG-1434, "Standard Technical Specifications General Electric Plants, BWR/6."

8. BWR Owners Group, "Position on Position on Regulatory Guide 1.97 Revision 2," July 1982.

9. Generic Letter 82-33, "Supplement 1 to NUREG-0737 – Emergency Response Capabilities."

10. NUREG-0737, "Clarification of TMI Action Plan Requirements."

11. NEDO-33160-A, "Regulatory Relaxation for the Post Accident SRV Position Indication System," October 2006.

APPENDIX A. COMPARISON TO REGULATORY GUIDE 1.97 VARIABLES

This appendix provides a comparison of the accident monitoring variables developed using the BWROG evaluation methodology to those in RG 1.97 Revision 3 and a typical BWR/4 plant. Table A-1 provides the specific variables to facilitate a comparison. This comparison is provided to allow an assessment of the differences between RG 1.97 Revision 3, the evaluation methodology which implements RG 1.97 Revision 4 and IEEE-497, and the actual application to a current plant.

Table A-1 contains 5 sets of information:

1. Variable – This column identifies the specific variables required for accident monitoring, consistent with RG 1.97 Revision 3.

2. RG 1.97 – This set of information is provided in two columns, consistent with RG 1,97 Revision 3:

    - Type – This column identifies the variable type identified in RG 1.97 for BWRs.

    - Category (Cat.) – Category in RG 1.97 Revision 3 is used to identify the design and qualification requirements for the accident monitoring systems. With respect to equipment qualification, Category 1 and 2 are required to be environmentally and seismically qualified, while there are no specific provisions for Category 3.

3. IEEE-497 – This set of information is contained in three columns and is intended to be consistent with the implementation of RG 1.97 Revision 4:

    - Type – This column identifies the variable type consistent with the criteria identified in IEEE-497. Based on the classification basis, the some variables can be associated with a number of different variable types.

    - Environmental Qualification (EQ) – Type A, B, C and D parameters are required to be environmentally qualified consistent with IEEE-497. Type E parameters are not required to be environmentally qualified consistent with IEEE-497.

    - Seismic Qualification (SQ) – Type A, B, and C parameters are required to be seismically qualified consistent with IEEE-497. Type E parameters are not required to be seismically qualified consistent with IEEE-497. Type D parameters are to be designed to be operable following a seismic event if the systems they monitor are required to be operable following a seismic event.

4. BWR/4– This set of information is provided in two columns:

    a. Type – Same as RG 1.97.

    b. Category (Cat.) – Same as RG 1.97.

5. Comments – This column contains specific comments relative the specific variable or groups of variables.

Table A-1 – Accident Monitoring Variables Comparison

	RG 1.97		IEEE-497			BWR/4		
Variable	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
RG 1.97 Rev. 3 – Type A Variables	A	1						Type A parameters are plant specific and Category 1 in RG 1.97. From a BWR safety analysis perspective, these parameters are considered Type A consistent with the criteria identified in RG 1.97 Rev. 3.
Reactor water level			A, B, C	Y	Y	A	1	
Reactor pressure			A, B, C	Y	Y	A	1	
Drywell pressure			A, B, C	Y	Y	A	1	
Suppression pool temperature			A, B, C	Y	Y	A	1	
Suppression pool water level			A, B, C	Y	Y	A	1	
RG 1.97 Rev. 3 – Type B Variables								
Reactivity Control								
Reactor power/neutron flux	B	1	B, D	N	N	B	3	Classification based on NRC Safety Evaluation Report for BWROG LTR NEDO-31558.
Control rod position	B	3	D	N	N	B	3	Type D because function is to demonstrate safety system performance.
RCS Soluble Boron Concentration	B	3	N/A			N/A		Not a BWR required parameter.
Core Cooling								
Coolant level in reactor	B	1	A, B, C	Y	Y	A	1	Reactor water level.
BWR core thermocouple	B	1	N/A			N/A		NRC approved deviation.
Maintain Reactor Coolant System Integrity	B	1						
RCS pressure	B	1	A, B, C	Y	Y	A	1	Reactor pressure.

Table A-1 – Accident Monitoring Variables Comparison								
	RG 1.97		IEEE-497			BWR/4		
Variable	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
Drywell pressure	B	1	A, B, C	Y	Y	A	1	
Drywell sump level	B	1	N/A			B	3	NRC approved deviation.
Maintaining Containment Integrity								
Primary containment pressure	B	1	A, B, C	Y	Y	A	1	Drywell pressure.
Primary containment isolation valve position								
MSIV position switches	B	1	D	Y	Y	B	1	Type D because function is to demonstrate safety system performance.
Cleanup system isolation valve position switches	B	1	D	Y	Y	B	1	Type D because function is to demonstrate safety system performance.
Shutdown cooling system isolation valve position switches	B	1	D	Y	Y	B	1	Type D because function is to demonstrate safety system performance.
Other RPV normally open isolation valve position switches on valves inside containment	B	1	D	Y	Y	B	1	Type D because function is to demonstrate safety system performance.
Other RPV normally closed isolation valve position switches on valves inside containment that require opening for a LOCA	B	1	D	Y	Y	B	1	Type D because function is to demonstrate safety system performance.
Other RPV normally open isolation valve position switches on valves outside primary containment	B	1	D	Y	Y	B	1	Type D because function is to demonstrate safety system performance.

Table A-1 – Accident Monitoring Variables Comparison

	RG 1.97		IEEE-497			BWR/4		
Variable	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
Other RPV normally closed isolation valve position switches on valves outside primary containment that require opening for pipe breaks outside primary containment	B	1	D	Y	Y	B	1	Type D because function is to demonstrate safety system performance.
Other RPV normally closed isolation valve position switches on valves that do not require opening for either a LOCA or pipe breaks outside of containment	B	1	D	N	N	B	1	Type D because function is to demonstrate safety system performance. Position is known prior to an accident. Both isolation valves not assumed to spuriously operate.
Normally open containment isolation valve position switches on valves inside containment	B	1	D	Y	Y	B	1	Type D because function is to demonstrate safety system performance.
Normally closed containment isolation valve position switches on valves inside containment that require opening for a LOCA	B	1	D	Y	Y	B	1	Type D because function is to demonstrate safety system performance.
Containment isolation valve position switches on valves outside primary containment that require opening for a LOCA	B	1	D	Y	Y	B	1	Type D because function is to demonstrate safety system performance.
Normally closed containment isolation valve position switches on valves inside or outside containment that do not require opening for a LOCA	B	1	D	N	N	B	1	Type D because function is to demonstrate safety system performance. Position is known prior to an accident. Both isolation valves not assumed to spuriously operate.

Table A-1 – Accident Monitoring Variables Comparison

	RG 1.97		IEEE-497			BWR/4		
Variable	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
RG 1.97 Rev. 3 – Type C Variables								
Fuel Cladding								
Radioactivity concentration or radiation level in circulating primary coolant	C	1	N/A			N/A		NRC approved deviation.
BWR core thermocouples	C	1	N/A			N/A		NRC approved deviation.
Reactor Coolant Pressure Boundary								
RCS pressure	C	1	A, B, C	Y	Y	A	1	Reactor pressure.
Primary containment area radiation	C	3	E	N	N	C	1	Not relied on in accident analysis or EPGs for breach of barrier. Only function is for EALs.
Drywell drain sump level	C	1	N/A			B	3	NRC approved deviation.
Suppression pool water level	C	1	A, B, C	Y	Y	A	1	
Drywell pressure	C	1	A, B, C	Y	Y	A	1	
Containment								
RCS pressure	C	1	A, B, C	Y	Y	A	1	Reactor pressure.
Primary containment pressure	C	1	A, B, C	Y	Y	A	1	Drywell pressure.
Containment and drywell hydrogen concentration	C	1	N/A			C	1	Provided for severe accident mitigation. Commercial grade equipment is acceptable. Consistent with 10CFR50.44.

Table A-1 – Accident Monitoring Variables Comparison								
	RG 1.97		**IEEE-497**			**BWR/4**		
Variable	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
Containment and drywell oxygen concentration	C	1	N/A			A	1	Provided for severe accident mitigation. Commercial grade equipment is acceptable. Consistent with 10CFR50.44. Oxygen monitored during normal operation.
Containment effluent radioactivity – noble gases	C	3	N/A			N/A		Not an identified pathway.
Radiation exposure rate	C	2	N/A			N/A		NRC approved deviation.
Effluent radioactivity – noble gases (buildings)	C	2	E	N	N	D	2	
Condensate and Feedwater Systems								
Main feedwater flow	D	3	N/A			D	3	Normal operating system.
Condensate storage tank level	D	3	N/A			D	3	Normal operating system.
Primary Containment Related Systems								
Suppression chamber spray flow	D	3	D	Y	Y	D	2	RHR system flow and valve position
Drywell pressure	D	2	A, B, C	Y	Y	A	1	
Suppression pool water level	D	2	A, B, C	Y	Y	A	1	
Suppression pool water temperature	D	2	A, B, C	Y	Y	A	1	
Drywell atmosphere temperature	D	2	D	Y	N	D	2	Not required for accidents.
Drywell spray flow	D	2	D	Y	Y	D	2	RHR system flow and valve position

Table A-1 – Accident Monitoring Variables Comparison

	RG 1.97		IEEE-497			BWR/4		
Variable	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
Main Steam System								
Main Steamline Isolation Valves Leakage Control System Pressure	D	2	N/A			N/A		NRC approved elimination of MSIV leakage control system.
Primary System Safety Relief Valve Positions, Including ADS or Flow Through or Pressure in Valve Line	D	2	N/A			N/A		NEDO-33160 A contains NRC acceptance of change in requirements for SRV position indication
Safety Systems								
Isolation Condenser System Shell-Side Water Level	D	2	N/A			N/A		Applies plants with isolation condenser only.
Isolation Condenser System Valve Position	D	2	N/A			N/A		Applies plants with isolation condenser only.
Safety/relief valve position indication	D	2	D	N	N	D	2	Backup instrument only. Not required to be seismically or environmentally qualified.
RCIC system flow	D	2	D	N	N	D	2	RCIC required only for anticipated operational occurrences.
HPCI system flow	D	2	D	Y	Y	D	2	HPCI or HPCS flow.
Core spray system flow	D	2	D	Y	Y	D	2	LPCS system flow
SLCS flow	D	2	D	N	N	D	2	Standby liquid control system pumps running. System not required for anticipated operational occurrences or accidents.
SLCS storage tank level	D	2	D	N	N	D	2	System not required for anticipated operational occurrences or accidents.
Residual Heat Removal System								

Table A-1 – Accident Monitoring Variables Comparison								
	RG 1.97		**IEEE-497**			**BWR/4**		
Variable	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
RHR system flow	D	2	D	Y	Y	D	2	
RHR heat exchanger outlet temperature	D	2	D	Y	Y	D	2	
Cooling Water Systems								
Cooling water temperature to ESF system components	D	2	N/A			N/A		NRC approved as a deviation based on providing alternate means.
Cooling water flow to ESF system components	D	2	D	Y	Y	D	2	RHR service water flow and essential service water flow.
Radwaste Systems								
High radioactivity liquid tank level	D	3	N/A			D	3	Normal operating system.
Ventilation Systems								
Emergency ventilation damper position	D	2	D	Y	Y	D	2	Differential pressure is an acceptable alternative.
Power Supplies								
Status of standby power and other energy sources important to safety	D	2	D	Y	Y	D	2	AC and DC power and pneumatic system pressure.

Table A-1 – Accident Monitoring Variables Comparison								
	RG 1.97		IEEE-497			BWR/4		
Variable	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
RG 1.97 Rev. 3 – Type E Variables								
Containment Radiation								
Containment area radiation level – high range	C	1	E	N	N	C	1	Type E variable not Category 1.
Reactor building or secondary containment area radiation	E	2	E	N	N	E	3	Reactor building area radiation.
Area Radiation								
Noble gases and vent rate flow	E	2	E	N	N	D	2	Secondary containment release point flow.
Particulates and halogens	E	3	E	N	N	D	2	Secondary containment release point radiation level.
Environs radiation and radioactivity	E	3	E	N	N	E	3	Portable instrumentation can be used.
Meteorology								
Wind speed and direction	E	3	E	N	N	E	3	
Estimate of atmospheric stability	E	3	E	N	N	E	3	
Accident Sampling	E	3						
Primary coolant and sump	E	3	N/A			N/A		Grab samples
Containment air	D	2	N/A			N/A		Grab samples