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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

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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
CHARM	Containment High Range Area Radiation Monitor
CIV	Containment Isolation Valve
CRD	Control Rod Drive
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
RCS	Reactor Coolant System

RG	Regulatory Guide
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RPV&PC	Reactor Pressure Vessel and Primary Containment
RPV&PCIS	Reactor Pressure Vessel and Primary Containment Isolation System
SAGs	Severe Accident Guidelines
SLCS	Standby Liquid Control System
SQUG	Seismic Qualification Utility's Group
SRV	Safety/Relief Valve
TMI	Three Mile Island
TSTF	Technical Specification Task Force
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 positions 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 2 and 3 to RG 1.97 (RG 1.97 – Reference 3 and 4), "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 2 and 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 currently contained in RG 1.97 Revisions 2 and 3. 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 Revisions 2 and 3 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 Revision 4 regulatory positions and use of the methodology results to support plant modifications based on the provisions of Revision 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 Revisions 2 and 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 guidance is provided in Revision 2 and 3 to Regulatory Guide 1.97 (RG 1.97 - Reference 3 and Reference 4). RG 1.97 Revision 2 and 3 provides a specific detailed list of the variables for BWRs that are required to be monitored and includes a comprehensive list of design and equipment qualification requirements. BWR Owners were required to comply with the provisions of either RG 1.97 Revision 2 or 3. The list of variables in Table 1 of RG 1.97 Revision 2 are similar to the list of variables in Table 2 of RG 1.97 Revision 3 for BWRs but there are differences in the design and qualification criteria. All current operating nuclear power plants have implemented a set of accident monitoring instruments that are consistent with the specific variables identified in RG 1.97 2 or 3 with NRC agreements on .exceptions.

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 Revisions 2 and 3. To further standardize the design and qualification requirements, these requirements are established based on the level of importance of the specific variable type.

The Boiling Water Reactor (BWR) Owners' Group (BWROG) 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 BWRs. This methodology and examples of its application provided in this licensing topical report (LTR) 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 Revisions 2 and 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 nine sections and one appendix.

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 LTR may be utilized by operating BWRs 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 (EPGs), 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 comparison between RG 1.97 Revision 3 and the results provided in this report, identification of previously approved deviations from earlier versions of RG 1.97, discussion of other changes caused by use of EPGs, potential changes to BWR Standard Technical Specifications for post accident monitoring (PAM) and additional changes which would be deviations from RG 1.97 Revisions 2 or 3 provisions for BWR Owners.

Section 8 provides the conclusions of the report.

Section 9 contains the references for this report.

Appendix A provides a comparison of the accident monitoring variables contained in RG 1.97 Revisions 2 and 3 with the accident monitoring variable identified using the BWROG methodology described in this LTR. For comparison, the accident monitoring variables identified for a typical BWR/4 are provided.

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.

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

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.

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

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 current licensing basis, including but not limited to the UFSAR..

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 that could result in potential offsite exposures comparable to the guidelines in 10CFR 50.34(a)(1), 50.67(b)(2) or in 10CFR100.11 as applicable.

For typical 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.

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. While not explicitly stated in RG 1.97 Revision 4 and IEEE 497, Type E instruments design and qualification criteria should be of high-quality commercial grade and should be selected to withstand the specified service environment which is consistent with the provisions of RG 1.97 Revisions 2 and 3 for Category 3 variables.

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 this 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 modifications and 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 anticipated operational occurrences 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 Revisions 2 or 3 which the NRC has reviewed and accepted including agreements on deviations from requirements. Included within RG 1.97 Revisions 2 or 3 are references to other Regulatory Guides which contain codes and standards and other 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 codes and 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 Revisions 2 or 3 for other Regulatory Guides and 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 chooses to use RG 1.97 Revision 4 would be expected to review their RG 1.97 commitments with respect to Regulatory Guides and 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. Consistent with Section 6.9 of IEEE 497-2002 (Reference 2), it is expected that to the extent practical, a direct variable will be used to monitor the related function. A less direct variable may be substituted if justified by analysis. 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.

The BWR Accident Analysis is typically described in Chapter 14 or 15 of the Updated Final Analysis Report (UFSAR). An integral part of the GE accident analysis is the Nuclear Safety Operational Assessment (NSOA) process which has been developed for BWRs. The focal point of the NSOA is the event analysis. In the event analysis, all essential protection sequences are evaluated until all required safety actions are successfully completed. The event analysis identifies all required front line safety systems and their essential auxiliaries. All events are analyzed until a stable condition is obtained.

In the event analysis, all essential systems, operator actions and limits to satisfy the required safety actions are identified. Limits are derived only for those parameters continuously available to the operator. Credit for operator action is taken only when an operator can be reasonably be expected to perform the required actions based on the information available to him.

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 4-1. Table 4-1 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."

Section 6.2 of IEEE Std. 497 addresses a concern that the use of the microprocessor-based instrumentation at the Type A, B and C variable level could result in a common cause failure. A potential for the common cause failure exists in the software that is used in the microprocessor-based instrumentation design.

Guidance for the application of the single failure criteria, that is defined in 10CFR50, Appendix A, is provided in IEEE Std 379, "Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Class 1E Systems" as is endorsed by the NRC in RG 1.53 Revision 2.

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

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 (SRV) for 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).

4.1.4.5 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 EOPs.

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

Endorsed by RG 1.97Revision 4, IEEE Standard 497-2002 provides design criteria for the accident-monitoring instrumentation. The selection criteria in IEEE Standard 497-2002 define instrumentation variable types based on the level of importance to the operators. The Critical Safety Functions (CSF) concept evolved from the implication that the operator need only monitor a relatively few pieces of information to ascertain the safety of the plant. The operator can carry out his duties by focusing on these critical functions without regard to the specific events that have occurred.

IEEE Standard 497-2002, Item 3.7 states that the CSF are those safety functions that are essential to prevent a direct and immediate threat to the health and safety of the public by maintaining Reactivity Control, Reactor Core Cooling, RCS integrity, Primary Reactor Containment Integrity, and Radioactive Effluent Control. Furthermore, IEEE Standard 497-2002, Section 4.2 states that Type B variables provide primary information to the operators to assess the plant CSF for the EOP implementation.

Unlike its predecessor RG1.97 Revision 3 that for Type B variables considered the Radioactive Effluent Control with Maintaining Containment Integrity function, IEEE Standard 497-2002 designated the Radioactive Effluent Control as a new function and separated it from the Maintaining Containment Integrity function.

The subject of concern associated with post-accident Radioactive Effluent Control is the potentially open radioactive release pathways from the primary containment to the secondary containment. In the BWR design, the Radioactive Effluent release pathways are deliberately isolated at the primary containment penetrations upon receipt of a LOCA isolation signal to establish containment integrity.

The Primary Containment contains isolation features that provide a barrier to the release of radioactive material due to the postulated loss of coolant accident from the primary containment to the secondary containment. Therefore, this barrier is assumed to remain intact for the postulated LOCA and limit any leakage of radioactive material to the secondary containment. These features assure that the Radioactive Effluent Control function as described in IEEE Std. 497-2002; Section 3.7 is met in terms of helping the operator to mitigate an accident. The critical safety function of primary containment control thus will also assure Radioactive Effluent Control safety function in a BWR design.

Radioactive Effluent Control is also inherent in Type C and Type E of RG 1.97 Revision 4 criteria for accident monitoring instrumentation. Type C variables provide information about the potential or the actual breach of the fission product barriers and Type E provides information about the magnitude and impact of the release of radioactive material, respectively. Thus, radioactive effluent control is comprehensively addressed by RG 1.97 Revision 4 and by this LTR methodology.

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 to 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.

The selection of generic BWR Type B variables for RG 1.97 Revision 4 is based on EPGs which have been approved by the NRC (NEDO-31331, March 1987) and have been subsequently incorporated into emergency procedure guidelines and severe accident guidelines (EPGs/SAGs). BWR EPGs contained in NEDO-31331 include contingency procedures one of which is primary containment flooding. As part of severe accident management initiatives, the BWR EPG/SAGs were developed which incorporated the primary containment flooding contingency procedure into SAGs. BWR EOPs are based on NRC approved EPGs.

The BWR EPGs are structured so that procedural steps can be accommodated concurrently and protection is provided for both the RPV and the Primary Containment. The result is that a common list of Type B variables are identified which do not strictly align with the list of Functions listed in RG 1.97 Revisions 2 and 3 but do address the safety functions listed in Revisions 2 and 3 and the critical safety functions in Revision 4.

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.
    - RPV water level.
2. Reactor coolant pressure boundary.
    - RPV water level.
    - RPV pressure.
    - Drywell pressure.
    - Suppression pool water level.
    - Suppression pool temperature.
3. Primary containment.
    - Drywell pressure.
    - Suppression pool water level.
    - Suppression pool temperature.

An engineering evaluation was performed to determine the most direct Type C variable under the provisions of RG 1.97R4. For BWRs, the parameters that provide the most direct indication of the integrity of the fission product barriers is RPV water level indication. Analysis and testing performed for BWR fuel confirms the relationship between RPV water level and cladding integrity. If water level is maintained above specified levels, fuel cladding integrity will be maintained. If water level drops below specified limits or is indeterminate, cladding integrity is assumed to be breached and operator action directed to restore water level and maintain core cooling.

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

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

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

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

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 4-2. 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 4-3.

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 4-3.

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 4-3.

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. Consistent with Section 6.9 of IEEE 497-2002 (Reference 2), it is expected that to the extent practical, a direct variable will be used to monitor the related function. A less direct variable may be substituted for the most direct variable if justified by analysis

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.

<u>RPV Isolation Valve Position Switches</u>

- 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.

The systems that penetrate primary containment vary from plant to plant. Each plant is required to provide a listing of the isolation valves, including their position for normal operating conditions, post accident conditions, and in the event of valve operator power failures. This information when combined with the plant specific Emergency Operating Procedures requirements is necessary to establish the specific system valve positions that satisfy this requirement. As a result, a plant specific engineering evaluation of each plant and system design is required to identify the specific containment isolation valves that meet this requirement.

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~~**Estimation of Atmospheric Stability**.
- Plant environs radiation monitors.
- Control room air inlet radiation monitors.
- Control room area radiation monitors.
- **Containment Air and Primary Coolant Post Accident Sampling (see PASS LTR NEDO-32991-A)**

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 SRV 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
		SRVs (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	SRVs
	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	SRVs
	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	SRVs
	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	SRVs
	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	SRVs
	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	SRVs
	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	SRVs
	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	SRVs
	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	SRVs
	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 SRV 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	SRVs
	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	SRVs
	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	SRVs
	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	SRVs
	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	SRVs
	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	SRVs
	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	SRVs
	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	SRVs
	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	SRVs
	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	SRVs
	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
SRVs
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 5-1 and 5-2 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 4. 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 following a seismic event.

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 4. Further, significant plant modifications to the current plant accident monitoring program may require NRC approval prior to their implementation. It is anticipated that NRC acceptance of this LTR methodology will be used for plant specific reviews of their post accident monitoring requirements and licensing commitments.

Included in plant post accident monitoring requirements and licensing commitments are those that originated from NUREG 0737. We would expect that provisions would be included in the NRC's acceptance of this Report. Licensees electing to adopt NEDO-33349 should perform a plant-specific review to determine the applicability of NEDO-33349. The use of the provisions of RG 1.97 Revision 4 does not relieve licensees from meeting the requirements of NUREG-0737 and 10 CFR 50.34(f) nor of performing reviews of the impact on the plant's emergency plan under 10 CFR 50.54(q).

Table 5-1 – Typical BWR/4 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 drive system.	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 5-1 – 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.
SRV position indication	Safety system performance indication for SRVs.	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	Not required for safety system performance indication.	D	N	N	**Position switches are not associated with any events requiring seismic or environmental qualification**
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	Not required for safety system performance indication. .	D	N	N	**Position switches are not associated with any events requiring seismic or environmental qualification**
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	
Cooling water temperature to ESF system components	Safety system performance indication for cooling system.	D	Y	Y	
Cooling water flow to ESF system Components (RHR service water system flow)	Safety system performance indication for RHR service water system.	D	Y	Y	
Cooling Water Flow to ESF System Components (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	

Table 5-1 – Typical BWR/4 Accident Monitoring Variables					
Variable	**Classification Basis**	**Type**	**EQ**	**SQ**	**Comments**
Containment radiation level	Monitor identified pathway.	E	~~N~~Y	~~N~~Y	**Retained as RG 1.97 Revision 2 and 3 Type E Category 1 variable with requirements contained in NUREG 0737**
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	Required by IEEE 497
Wind speed and direction	Monitor environmental conditions.	E	N	N	
~~Ambient air temperature~~**Estimate of Atmospheric Stability**	Monitor environmental conditions.	E	N	N	Required by IEEE 497
Plant environs radiation monitors	Monitor plant environs.	E	N	N	
Control room area radiation monitors	Monitor control room	E	N	N	Required by IEEE 497
Containment air and primary coolant sampling	**Post accident sampling**	**E**	**N**	**N**	**See LTR NEDO-32991-A**

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.
SRV position indication	Safety system performance indication for SRVs.	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	**Position switches are not associated**

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.				**with any events requiring seismic or environmental qualification**
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	**Position switches are not associated with any events requiring seismic or environmental qualification**
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	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	

Table 5-2 – Typical BWR/6 Accident Monitoring Variables					
Variable	**Classification Basis**	**Type**	**EQ**	**SQ**	**Comments**
Containment radiation level	Monitor identified pathway.	E	~~N~~**Y**	~~N~~**Y**	**Retained as RG 1.97 Revision 2 and 3 Type E Category 1 variable with requirements contained in NUREG 0737**
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	Required by IEEE 497
Wind speed and direction	Monitor environmental conditions.	E	N	N	
~~Ambient air temperature~~**Estimate of Atmospheric Stability**	Monitor environmental conditions.	E	N	N	Required by IEEE 497
Plant environs radiation monitors	Monitor plant environs.	E	N	N	
Control room area radiation monitors	Monitor control room	E	N	N	Required by IEEE 497
Containment air and primary coolant sampling	**Post accident sampling**	**E**	**N**	**N**	**See LTR NEDO-32991-A**

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. Use of this LTR in adopting RG 1.97 Revision 4 requires the plant to identify design differences were applicable. 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 SRVs 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 SRVs 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.
- SRV 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 SRV 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 SRV 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. Plants need to document their codes and standards as compared to IEEE-497 2002 as part of plant specific applications.
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 dated June 2006 (Reference 1). RG 1.97 was originally developed in December 1975 to provide general guidance on instrumentation to assist operators in assessing plant conditions during and following an accident. As a result of the TMI Unit 2 accident, RG 1.97 was revised including defining "Type" variables (A-E) which are consistent with RG 1.97 Revision 4 definitions (Section 2.1 of this LTR). The revision to RG 1.97 established Categories for design and qualification requirements and provided prescriptive lists of variables to be provided. Revision 2 of RG 1.97 (Reference 2) was released in December 1980 followed by Revision 3 (Reference 3) in May 1983. Included in Revision 2 and 3 are Tables which define the Type variables to be provided for BWRs. Table 1 of RG 1.97 Revision 2 and Table 2 of RG 1.97 Revision 3 is organized by Type of variable with Type A being noted as being plant specific. Type B-E variable are defined in the Tables under Functions which are in bold type. For Type B variables the Functions listed are Reactivity Control, Core Cooling, Maintain Reactor Coolant System Integrity and Maintaining Containment Integrity. Under each Function, specific variables are listed. Type C variable Functions are Fuel Cladding, Reactor Coolant Pressure Boundary and Containment. Type D and Type E also provide a list of Functions.

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. The list of Functions under Type Variables is eliminated. In addition, the specific design and qualification requirements are established based on the importance of the specific variable Type with the eliminations of the Categories included in Revisions 2 and 3.

Section 4. of this Report describes the methodology used to establish the Type variables to comply with the provisions of RG 1.97 Revision 4 for generic operating BWRs. The methodology defines a generic list of Type A variables based on accident analysis supported by emergency procedure guidelines. This generic list needs to be reviewed on a plant specific basis but should reflect the Type A variables for most operating BWRs. BWRs did not provide for NRC review, a generic list of Type A variables to comply with RG 1.97 Revisions 2 and 3. This was left as being plant specific. As a result, there are currently differences in the BWR Fleet on what is defined as a Type A variables.

The selection of generic BWR Type B variables for RG 1.97 R4 is based on BWR EPGs which have been approved by the NRC (NEDO-31331, March 1987) and have been subsequently incorporated into EPGs/SAGs. BWR EOPs are based on NRC approved EPGs.

There is little or no change in the list of variables resulting from use of Revision 4 from what is depicted in Revisions 2 and 3 for Type D and E variables except for the elimination of Categories.

A comparison has been made between RG 1.97 Revisions 2 and 3 and the results are provided in this LTR in Table A-1.
Section 7.1 summarizes NRC approved deviations currently identified in the Standard Review Plan. Section 7.2, reconciles differences between RG 1.97 Revision 3 and the results of using this LTR methodology for determining Revision 4 Type B and C variables. Section 7.3 discusses the expected Technical Specification impact. Section 7.4 provides addition technical information in support of changes from current RG 1.97 requirements.

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

Included within this LTR results are changes to variables which have resulted from NRC approval of plant specific deviations. Justification for NRC approval of the deviations can be found in the recently approved NRC's Standard Review Plan NUREG 0800**, Revision 5 – March 2007 which provides criteria for use of RG 1.97 Rev 4 and includes Branch Technical Position (BTP) 7-10 titled Guidance on Application of Regulatory Guide 1.97. The BTP includes Table 1 for BWRs .** ~~Section 7 Guidance on Acceptance of RG 1.97 which provides criteria for use of RG 1.97 R4 and includes Table 1 to Branch Technical Position HICB-10-5 Revision 5 dated March 2007.~~ This table identifies the following variables with acceptance guidelines for deviations from Rev 2/**3:**

- Neutron Flux - Design criteria can be based on NEDO-31558-A providing a plant specific evaluation of the electric power distribution to the neutron monitoring system is performed. Neutron Flux is listed as a Type B variable in Rev 2/3 and is concluded to be a Type B variable using the methodology of this Report.

- Coolant level in reactor vessel – range requirements can be modified

- Core Temperature – not necessary. Listed as Type B and C in Revisions 2 and 3

- Drywell sump and drywell drain sumps level – listed as a Type B and Type C variable in Rev 2/3. This Report concludes variable is not relied on in safety analysis nor EPGs for small or large leaks. This is consistent with the acceptance guidelines.

- Primary containment isolation valve position – eliminates need for redundancy.

- Safety Relief Valve position – references NEDO-33160. Listed as Type D, Category 2 in Revisions 2 and 3

- Radioactivity concentration or radiation level in circulating primary coolant – not required. Listed as Type C in Revisions 2 and 3

- Containment and drywell hydrogen and oxygen concentration – requirements based on 10CFR 50.44 which indicates variables are not needed for design basis events but are needed for beyond design basis. Revisions 2 and 3 identified as Type C.

- Suppression chamber and drywell spray flows – use of RHR flow, suppression chamber temperature and pressure as alternatives to flow providing analysis supports use. Identified as Type D in Revisions 2 and 3 and as Type D in this LTR

- Standby liquid control system (SLCS) flow – use of pump discharge pressure and tank level as alternative. Identified as Type D in Revisions 2 and 3 and Type D in this LTR

- 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 for Revisions 2 and 3 which identified as Type E. Report concludes Type E

- Radiation exposure rate used for releases – change to Category 3 for Revisions 2 and 3 which lists as Type E. This LTR concludes Type E.

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.

7.2 Type B and Type C Differences from RG 1.97 Revision 2/3

The purpose of this section is to reconcile the results of this LTR methodology using RG 1.97 Revision 4 with the prescriptive list in RG 1.97 Revision 2 Table 1 and Revision 3 Table 2 for BWRs. Note that the Revisions 2 and 3 tables have not been updated since issuance, so the tables do not reflect all the generic deviations and other licensing action which conclude that certain variables are not required for the BWR design. Table A-1 includes Table 1 of R 2 and Table 2 of R 3 by function and shows the result of applying the LTR methodology to define Type variables for a typical BWR/4 design. Similar results would be shown for a typical BWR/6 design. Also included for information in Table A-1 is a BWR plant licensing commitments relative RG 1.97 Revision 2 (the plant has a license commitment to Revision 2). While Revision 4 does not identify the functions included in Revision 2, all the functions are addressed by the Revision 4 process.

<u>Type B Functions</u>

<u>RG 1.97 Revisions 2 and 3 were issued during the development of BWR EPGs.</u> This LTR is based on the provisions contained in RG 1.97 Revision 4 for Type B variables which are based on the EPGs, which have been approved by the NRC (NEDO-31331, March 1987). Reconciliation of RG 1.97 Revision 4 to RG 1.97 Revision 2 Table 1 and RG 1.97 Revision 3 Table 2 for BWRs must address the EPG organization and licensing actions which have resulted in modifications to the earlier list of variables.

Revision 2 Table 1 and Revision 3 Table 2 titled "BWR Variables" identifies safety functions for Type B Variables. The safety functions are defined as (1) reactivity control, (2) core cooling, (3) maintain reactor coolant system integrity and (4) maintain containment integrity (including radioactive effluent control). This LTR uses the process described in Revision 4 to identify the

Type B variables based on the specified BWR EPG entry level conditions that are consistent with the plant critical safety functions.

There are differences in the terminology used in RG 1.97 Revisions 2 and 3 for safety function and the organization of the symptom based BWR EPGs that are used to define critical safety functions consistent with RG 1.97 Revision 4. The BWR EPGs have a different but similar definition for safety function which covers: (1) RPV control with an integrated procedure for reactivity control, pressure control and level control and (2) primary containment control which is integrated with RPV control. The BWR EPGs are structured so that procedural steps can be accommodated concurrently and protection is provided for both the RPV and the Primary Containment. The result is that a common list of Type B variables are identified which do not strictly align with the list of Functions listed in RG 1.97 Revisions 2 and 3but do address the safety functions listed in Revisions 2 and 3 and the critical safety function in Revision 4.

The BWR EPGs (Reference 14) approved by the NRC have been subsequently incorporated into emergency procedure guidelines and severe accident guidelines (EPGs/SAGs). BWR EPGs contained in Reference 14 included contingency procedures. One of the contingency procedures is primary containment flooding. As part of severe accident management initiatives, the BWR EPG/SAGs were developed which incorporated the primary containment flooding contingency procedure into SAGs. BWR EOPs are based on NRC approved EPGs.

As described in Section 4.1 of this LTR, the BWR EPGs have four top-level guidelines that were used to develop the critical safety functions as defined in RG 1.97 Rev 4 for Type B variables.

RPV control (further divided into)
  Reactivity control
  Pressure Control
  Level Control

Primary containment control

The functions described in RG 1.97 Revisions 2 and 3 of Reactivity Control, Core Cooling, Maintaining Reactor Coolant System Integrity are met by the EPGs under RPV Control which integrates the RG 1.97 functions into the procedures for reactivity control, pressure control and level control. The result is a common list of Type B and C variables in the LTR, which do not strictly align with the RG 1.97 Rev Revisions 2 and 3 lists.

The function "Maintaining Containment Integrity" is met by the Primary Containment control EPG. The result of using the EPGs for Type B variables is contained in Section 4.2.3 of the LTR, but this should be viewed in the context of a common list of variables. As an example, drywell pressure is listed in Section 4.2.3 of this LTR as containment control but it also is an entry condition for RPV control to assure operator scram initiation if limits are exceeded.

The following addresses reconciliation of Revisions 2 and 3 Type B safety functions with the Revision 4 results based on BWR EPG integrated procedures and NRC approved deviations. For Reactivity Control, Revisions 2 and 3 includes neutron flux which is also identified in this LTR for R4. The other two Reactivity Control variables listed in Revisions 2 and 3 as Type B are control rod position which this LTR concludes is a Type D variable and RCS soluble boron concentration which was determined to not be a BWR required parameter. Some BWRs do list standby liquid control system (SLCS) tank boron concentration as their means of meeting Revisions 2 and 3 rather than RCS boron concentration. This LTR concludes SLCS tank boron concentration indication is not used post accident, unless there is a plant specific licensing commitment..

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

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

The Revisions 2 and 3 Maintaining Containment Integrity Function is met by the Primary Containment Control EPG for Revision 4. Primary containment pressure is listed in Revisions 2 and 3. This LTR lists drywell pressure as the appropriate variable which is viewed as the same as primary containment pressure. Drywell pressure and Primary Containment Pressure are essentially the same instruments for BWR Mark I and II containments and both terms are used interchangeably. The only pressure difference is due the small water leg in the downcomers and the opening pressure of the vacuum breaker valves between the suppression chamber atmosphere and the drywell. Therefore, for the purposes of the BWR EPGs, the pressure instrumentation in the drywell is effectively the same as the pressure suppression chamber. The Mark III containment design has a drywell completely surrounded by the containment that is separated from the containment atmosphere by the water in the weir wall and the containment to drywell vacuum breakers. The Mark III drywell and containment pressure instrumentation have only minor differences in their readings. The drywell pressure indication is used in EPGs.

Primary Containment Isolation Valve (CIV) position indication is listed in Revisions 2 and 3 as Type B and as Type D using Revision 4. CIV position indication is further discussed in Section 7.4.1. Not included in Revisions 2 and 3 but included using Revision 4 as Type B are suppression pool temperature and suppression pool water level both of which are monitored to assure the BWR containment safety function is being maintained consistent with the EPGs. Suppression pool temperature is controlled within specified limits based on prescribed heat capacity temperature limits, which are established to limit containment loads. Suppression pool level is controlled within specified limits based on prescribed SRV discharge limits, which are established to limit containment loads. Suppression pool temperature and water level are also controlled to assure that the ECCS pump net positive suction head requirements are satisfied. Revisions 2 and 3 identifies suppression pool temperature as Type D and suppression pool water level as Type C and Type D variables.

Type C Functions

RG 1.97 Revi~~d~~**s**ion 4 requires determination of Type C variables based on safety analysis, design basis for fission product barriers and EPGs. **RG 1.97** Revisions 2 and 3 **under**~~The~~ Type C Fuel Cladding function identifies three variables which were determined based on specific licensing actions to not be required for BWRs (radioactivity concentration, analysis of primary coolant, and BWR core thermocouples). An engineering evaluation was performed for this LTR to determine the most direct Type C variable consistent with the requirements of Revision 4. Based on the plant accident analysis licensing basis, design basis documentation for the fission product barriers, and the BWR EPGs, RPV water level was determined to be the best indicator of fuel cladding integrity for the BWR. Analysis and testing performed for BWR fuel confirms the relationship between RPV water level and cladding integrity. If water level is maintained above specified levels, fuel cladding integrity will be maintained. If water level drops below specified limits or is indeterminate, cladding integrity is assumed to be breached and operator action directed to restore water level and maintain core cooling.

Consistent with the BWR EPGs, the status of the fuel cladding barrier can be determined by the status of core cooling. The fuel cladding barrier is protected when the core remains adequately cooled. The fuel cladding barrier is no longer protected when adequate core cooling cannot be restored and maintained. RPV water level instrumentation is the means of determining if adequate core cooling was provided.

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

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

There is other instrumentation and RG 1.97 Rev ision 4 Type E radiation detection variables, which will be available to the operator to determine if core damage has occurred and the magnitude of the damage. These would include off-gas monitors, main steamline radiation monitors, hydrogen monitors, containment radiation monitors, and sampling of RPV radioactivity concentration. These additional variables are used for confirmation and to assist in emergency planning but are not related to the fuel cladding fission product barrier for EPGs.

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

Revisions 2 and 3 Type C Containment Function identifies several variables which are included as NRC approved deviations (e.g., hydrogen and oxygen monitors), and variables which have been determined to be Type E (containment effluent radioactivity and radiation exposure rate effluent radioactivity) as well as RCS Pressure and Primary Containment Pressure. This LTR has concluded that the RCS pressure is not a direct or indirect indicator of a breach in the Primary Containment Integrity barrier. It has been concluded that the Type C variables that comply with Revision 4 are drywell/containment pressure, suppression pool level, and suppression pool temperature. Suppression pool level and temperature have been included as required variables, as these must be maintained within established limits to support containment integrity design requirements, including containment hydrodynamic load assumptions.

7.3 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 this LTR conclusions with a few exceptions. The exceptions are drywell sump and drain level , primary containment radiation and containment isolation valve position indication. .

The LTR provides a generic list of Type A variables that includes RPV water level, RPV pressure, drywell pressure, suppression pool temperature, and suppression pool level instrumentation. In addition, this LTR concludes that neutron flux would be a Type B variable. Prior NRC agreement in a separate BWROG effort contained in NEDO-31558 provided conditions for plant exclusion of neutron monitoring from Technical Specifications.

The methodology used in this LTR for establishing on a generic basis Type A variables was based on RG 1.97 Revision 4 (IEEE Standard 497-2002) selection criteria. RG 1.97 Revision 4 eliminates instrument qualification categories, and thus, non-Type A Category 1 variables. The

BWROG is expected tol be pursuing Technical Specification changes, after this LTR has been accepted, based on the inclusion of all RG 1.97 Revision 4, Type A, B and C variables into the improved Standard Technical Specification for PAM.

This Technical Specification change is expected to address the guidance provided in the NRC letter dated May 9, 1988 from Thomas Murley to NSSS Owners Groups, including BWROG concerning the original inclusion of RG 1.97 into PAM Standard Technical Specifications and what is required for changes including risk information.

A preliminary review concludes that RG 1.97 Revision 4, Type A, B and C variables will meet the criteria for inclusion in NUREG 1433 and NUREG 1434 Standard Technical Specifications. Several RG 1.97 Revision 3, and non-Type A, Category 1 variables currently included in Technical Specifications may not meet the criteria for inclusion. A separate proposed PAM Technical Specification change is expected to be processed through the Technical Specification Task Force (TSTF) process for changes to BWROG Standard Technical Specifications (NUREG 1433/1434).

7.4 Technical Basis for 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. BWR 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 LTR.

Using this LTR, 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;

7.4.1 Primary Containment Isolation Valve Position Indication

Primary containment isolation valve (CIV) position indication is included in RG 1.97 Revisions 2 and 3 as a Type B, Category 1 variable under the function Maintaining Containment Integrity. RG 1.97 Revision 4 defines Type B variables as those variables which provide primary information to the control room operators to assess plant critical safety features which includes Primary Containment Integrity. This LTR based on RG 1.97 Revision 4 uses BWR EPGs to determine the Type B variables including the Primary Containment Control EPG, which addresses post accident Containment Integrity. Section 4.2.2 contains the results of this LTR evaluation, which identifies drywell/containment pressure, suppression pool level and suppression pool temperature as the Type B variables for Containment Integrity. Section 4.4.5 discusses isolation valve position indication and requirements for RPV and containment isolation valves, which have different requirements and has been evaluated to be a Type D variable. Valve position indication is used to verify system safety status by confirmation that the safety systems have functioned as designed. The required system isolation requirements are fulfilled by the redundant RPV and primary containment (RPV&PC) isolation valves, which provide isolation of containment as required. The RPV&PC isolation system provides safety related isolation signals to each of the RPV&PC isolation valves. The RPV&PC isolation system (RPV&PCIS) is designed to provide automatic isolation when required.

The RPV&PC isolation valves are required to assure Containment Integrity both prior to and post accident. The primary information the control room operator relies upon post accident is drywell/containment pressure with suppression pool temperature and suppression pool level needed to assure containment integrity is maintained throughout the accident. The RPV&PC isolation valves are safety systems designed to meet single failure criteria and to align properly to support containment integrity post accident. The RPV&PC isolation valves and the isolation signals are included in BWR improved Standard Technical Specifications for containment (Section 3.3.6.1 and 3.6.1.3 of the improved Standard Technical Specifications). RPV&PC isolation valve position indication is included in post accident monitoring (PAM) Standard Technical Specifications because it is listed in RG 1.97 Revision 3 as a non-Type A Category 1 variable.

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

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

Valve position indication is used as verification of the containment system status and to indicate that the RPV&PC isolation valves have performed their safety system function of containment isolation. This is the definition of a Type D variable in RG 1.97 Revision 4. There is a similar definitions in Revisions 2 and 3. A typical BWR will have approximately 40 containment penetrations with automatic isolation valves and thus up to 80 RPV&PCIV position indication systems currently listed as RG 1.97 Revision 3 Type B variables. In the unlikely event that a containment penetration would not meet its design function to be isolated post accident, the operator will have the redundant RPV&PC isolation valve position indication and other plant indications to indicate that the containment has not been isolated.

7.4.2 Containment Radiation Monitors and Noble Gas Monitors

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

~~The containment high- range radiation monitor (CHARM)is included in RG 1.97 Revisions 2 and 3 as a Type E Category 1 variable and is also included in post accident monitoring (PAM) Technical Specifications as it is identified as a non-Type A Category 1 variable. This LTR concludes CHARM is a Type E variable and should have design and qualification requirements based on its intended purpose. The purpose of CHARM is as described in Table 1 of RG 1.97 Revisions 2 and 3 which is "detection of significant release: release assessment: long term surveillance; emergency plan actuation". It provides information used for BWR core damage assessments used in emergency action level (EAL) classifications in addition to other RG 1.97 variables.~~

~~CHARM were required to be installed in operating plants and plants under construction as a result of the TMI Unit 2 accident lessons learned. NUREG 0737 item II.F.1 (Reference 12) established the requirements for such monitors and RG 1.97 R2/3 subsequently incorporated the requirements as a Category 1, Type E variable. Category 1 is the highest level of requirements with requirements imposed for CHARM including redundancy, extended radiation detection ranges for beyond design basis events and environmental and seismic qualification. The separate containment radiation monitors, which existed prior to NUREG 0737 were included in RG 1.97 Revisions 2 and 3 as a Type C Category 3. .To satisfy NUREG 0737 requirements for Category 1, most plants replaced their commercial grade containment radiation monitors with CHARM.~~

~~This LTR concludes CHARM use in BWRs does not meet the criteria for being a Type A, Type B or Type C variable. It is not relied on in plant safety analysis, is not relied upon in EPGs for critical safety functions, and does not provide primary information to the control room operator to indicate a potential or actual breech of a barrier. This LTR concludes CHARM is a Type E variable used to assist in determining the extent of core damage and in emergency planning activities.~~

~~There are specified design requirements in RG 1.97 Revision 4 that apply to Type E variables that would need to be met for CHARM and for noble gas monitors. In addition to what is included in RG 1.97 Revision 4 for Type E, the Design and Qualification criteria contained in RG 1.97 Revision 3 Section 1.3.3.a would apply which is "The instrumentation should be of high-quality commercial grade and should be selected to withstand the specified service environment."~~

~~Type E variables are used for monitoring the magnitude of releases, environmental conditions to determine the impact of releases, monitor radiation level in plant environs and monitoring of radiation levels in plant environs. In each case, it is expected that the conditions the monitor will~~

~~function under will be established as well as the expected range so that information will be available for operator and emergency planning decisions.~~

~~The above does not address NUREG 0737 Section II.F.1 requirements especially with respect to CHARM. CHARM was required to be installed in BWR operating plants and plants under construction as a result of the TMI accident lessons learned, not because of reliance in BWR safety analysis or EOPs. NUREG 0737 Item II.F.1 established the requirements for such monitors and RG 1.97 Revision 3 subsequently incorporated the monitor as a Category 1, Type E variable. Requirements include classification of RG 1.97 Revision 3 Category 1, which is the highest level of requirements consistent with the design and quality requirements for a Basic Component as defined in 10CFR50.2, including requirements for redundancy, provisions for essential power, having extended radiation detection ranges for beyond design basis events and being environmentally and seismically qualified to such extended ranges. This LTR concludes that CHARM in a BWR does not meet the definition of a Basic Component based on the function that it provides which is for core damage assessments and emergency planning activities.~~

~~As noted, a review of the CHARM use in a BWR concludes that it provides information used in post accident core damage assessments and in emergency planning. Similarities exist for appropriate design requirements for CHARM with the requirements contained in the amended combustible gas control in containment Rule (10CFR50.44) for hydrogen monitors (Reference 17). While this was an amended Rule concerning combustible gas control, the impacts of revised requirements resulting from NUREG 0737 for equipment later determined to be for severe accidents and emergency planning were addressed and revised requirements provided.~~

~~Hydrogen monitors requirements were modified as a result of NUREG 0737 Item II.F.1 and prior amendments to the Combustible Gas Rule resulting in a determination that the monitors were needed for design basis accidents as a Basic Component (10CFR50.2) and requiring design and qualification requirements consistent with Basic components. The hydrogen monitors were subsequently incorporated into RG 1.97 Revisions 2 and 3 and into PAM Technical Specifications. The amended combustible gas rule resulted in a revision to the requirements for hydrogen monitors to non-safety related commercial grade but imposed a requirement that they be functional. Section (b)(4)(ii) of the amended combustible gas rule states: "Equipment must be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design basis accident for accident management, including emergency planning." Similar requirements would be expected for CHARM as they also provide information used in BWR core damage assessments for emergency planning.~~

Noble gas monitors requirements were defined in NUREG 0737 Item II.F.1 and were incorporated into RG 1.97 Revision 3 as Type C, Category 3 for the containment barrier and as Type E, Category 2. Category 2 imposed environmental qualification requirements in accordance with RG 1.89. It is concluded that Type E is the appropriate classification for noble gas monitors under RG 1.97 Revision 4. This classification will result in high quality commercial grade with the monitors designed to meet the specified service environment for radiation releases and use in emergency planning activities but not to be required to meet the requirements of RG 1.89.

7.4.3 Safety/Relief Valve Position Indication System

The SRV 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 SRV 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 SRV 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 Revisions 2 and 3 Category 2 criteria. Therefore, the SRV 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 Revision 4 parameters. SRV 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 SRV 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. Regulatory Guide 1.97 Revision 4 "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," June 2006

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

3. Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," December 1980

4. 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

5. 10CFR50 Appendix A, "General Design Criteria."

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

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

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

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

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

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

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

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

14. NEDO–31331, BWR Owners' Group Emergency Procedures Guidelines Revision 4, March 1987

15. NEDO-32991-A BWR Owners' Group Licensing Topical Report "Regulatory Relaxation for BWR Post-Accident Sampling Stations (PASS)" dated August 2001.

!6 BWR Owners' Group Emergency and Severe Accident Guidelines Revision 2, March 2001.

17 10CFR 50.44 Combustible Gas Control in Containment amended Rule, October 2003

18 NEDO-24011-A-14 "General Electric Standard Application for Reactor Fuel", July 2000

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 2/3 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 2/3, the evaluation methodology which implements RG 1.97 Revision 4 and IEEE-497, and the actual application to a current plant. The current plant information is based on a licensing commitment to RG 1.97 Revision 2 and includes plant specific deviations which were approved by the NRC and incorporated into their RG 1.97 program

Table A-1 contains 6 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 Rev 2 – This set of information is provided in two columns, consistent with RG 1.97 Revision 2:

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

   - Category (Cat.) – Category in RG 1.97 Revision 2 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. RG 1.97 Rev 3 – same information as Rev 2

4. 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 same 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 their function as required by 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.

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

    a. Type – Same as RG 1.97 Rev 2.

    b. Category (Cat.) – Same as RG 1.97 Rev 2.

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

Table A-1 – Accident Monitoring Variables Comparison										
	RG 1.97 Rev 2 Table 1		RG 1.97 Rev 3 Table 2		IEEE-497 Consistent with RG 1.97 Rev 4			BWR/4 Typical		
Variable	**Type**	**Cat.**	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
Type A Variables										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. The typical Plant listed their Type A as also meeting other Rev 2 Type variables as required
Reactor Water Level	A	1	A	1	A, B, C	Y	Y	A,B	1	
Reactor Pressure	A	1	A	1	A, B, C	Y	Y	A,B,C	1	
Drywell Pressure	A	1	A	1	A, B, C	Y	Y	A,B,C,D	1	
Suppression Pool Temperature	A	1	A	1	A, B, C	Y	Y	A,D	1	
Suppression Pool Water Level	A	1	A	1	A, B, C	Y	Y	A,C,D	1	
Type B Variables										
Reactivity Control										
Neutron Flux	B	1	B	1	B, D	N	N	B	2	Classification based on NRC Safety Evaluation Report for BWROG LTR NEDO-31558.
Control Rod Position	B	3	B	3	D	N	N	B	3	Type D because function is to demonstrate safety system performance.
RCS Soluble Boron Concentration (Grab Sample)	B	3	B	3	N/A			B	3	Not a BWR required parameter to measure RCS boron. Some plants refer to SBLC boron tank soluble boron

Table A-1 – Accident Monitoring Variables Comparison										
	RG 1.97 Rev 2 Table 1		RG 1.97 Rev 3 Table 2		IEEE-497 Consistent with RG 1.97 Rev 4			BWR/4 Typical		
Variable	Type	Cat.	Type	Cat.	Type	EQ	SQ	Type	Cat.	Comments
Core Cooling										
Coolant Level in Reactor	B	1	B	1	A, B, C	Y	Y	B	1	Deviation approved for range.
BWR Core Thermocouple	B	1			N/A			N/A		NRC approved deviation
BWR Core Temperature			B							NRC approved deviation.
Maintaining Reactor Coolant System Integrity										
RCS Pressure	B	1	B	1	A, B, C	Y	Y	B	1	RPV pressure.
Drywell Pressure	B	1	B	1	A, B, C	Y	Y	B	1	
Drywell Sump Level	B	1	B	1	N/A			B	3	NRC approved deviation.
Maintaining Containment Integrity										
Primary Containment Pressure	B	1	B	1	A, B, C	Y	Y	B	1	Drywell pressure provides containment pressure indication.
Primary Containment Isolation Valve Position (excluding check valves)	B	1	B	1				B	1	
Drywell Pressure					**A, B, C**	**Y**	**Y**			**Not included in RG 1.97 Rev 2/3**
Suppression Pool Temperature					**A, B, C**	**Y**	**Y**			**Not included in RG 1.97 Rev 2/3**
Suppression Pool Water Level					**A, B, C**	**Y**	**Y**			**Not included in RG 1.97 Rev 2/3**
MSIV position switches					D	Y	Y			Type D because function is to demonstrate safety system performance.
Cleanup system isolation valve position					D	Y	Y			Type D because function is to

Table A-1 – Accident Monitoring Variables Comparison										
	RG 1.97 Rev 2 Table 1		RG 1.97 Rev 3 Table 2		IEEE-497 Consistent with RG 1.97 Rev 4			BWR/4 Typical		
Variable	**Type**	**Cat.**	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
switches										demonstrate safety system performance.

Table A-1 – Accident Monitoring Variables Comparison										
	RG 1.97 Rev 2 Table 1		RG 1.97 Rev 3 Table 2		IEEE-497 Consistent with RG 1.97 Rev 4			BWR/4 Typical		
Variable	**Type**	**Cat.**	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
Shutdown cooling system isolation valve position switches					D	Y	Y			Type D because function is to demonstrate safety system performance.
Other RPV normally open isolation valve position switches on valves inside containment					D	Y	Y			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					D	Y	Y			Type D because function is to demonstrate safety system performance.
Other RPV normally open isolation valve position switches on valves outside primary containment					D	Y	Y			Type D because function is to demonstrate safety system performance.
Other RPV normally closed isolation valve position switches on valves outside primary containment that require opening for pipe breaks outside primary containment					D	Y	Y			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					D	N	N			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 Rev 2 Table 1		RG 1.97 Rev 3 Table 2		IEEE-497 Consistent with RG 1.97 Rev 4			BWR/4 Typical		
Variable	**Type**	**Cat.**	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
Normally open containment isolation valve position switches on valves inside containment					D	Y	Y			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					D	Y	Y			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					D	Y	Y			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					D	N	N			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 Rev 2 Table 1		RG 1.97 Rev 3 Table 2		IEEE-497 Consistent with RG 1.97 Rev 4			BWR/4 Typical		
Variable	**Type**	**Cat.**	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
Type C Variables										
Fuel Cladding										
Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	C	1	C	1	N/A			C	3	NRC approved deviation.
Analysis of Primary Coolant (Gamma spectrum)	C	3	C	3	N/A			C	3	NRC approved deviation
BWR Core Thermocouples	C	1			N/A			N/A		NRC approved deviation
BWR Core Temperature			C							NRC approved deviation.
Reactor Water Level					**A, B, C**					**Not included in RG 1.97 Rev 2/3**
Reactor Coolant Pressure Boundary										
RCS Pressure	C	1	C	1	A, B, C	Y	Y	A.B,C	1	Reactor pressure.
Primary Containment Area Radiation	C	3	C	3	E	N	N	C,E	1	Not relied on in accident analysis or EPGs for fuel cladding integrity.
Drywell Drain Sump Level (Identified and Unidentified Leakage)	C	1	C	1	N/A			C	3	NRC approved deviation.
Suppression Pool Water Level	C	1	C	1	A, B, C	Y	Y	C	1	
Drywell Pressure	C	1	C	1	A, B, C	Y	Y	C	1	
Reactor Water Level					**A, B, C**	**Y**	**Y**			**Not included in RG 1.97 Rev 2/3**
Containment										
RCS Pressure	C	1	C	1	A, B, C	Y	Y	A.B,C	1	Reactor pressure.

Table A-1 – Accident Monitoring Variables Comparison										
	RG 1.97 Rev 2 Table 1		RG 1.97 Rev 3 Table 2		IEEE-497 Consistent with RG 1.97 Rev 4			BWR/4 Typical		
Variable	**Type**	**Cat.**	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
Primary Containment Pressure	C	1	C	1	A, B, C	Y	Y	A,B,C,D	1	Drywell pressure.
Suppression Pool Water Temperature					**A, B, C**					**Not included in RG 1.97 Rev 2/3**
Suppression Pool Water Level					**A, B, C**					**Not included in RG 1.97 Rev 2/3**

Table A-1 – Accident Monitoring Variables Comparison										
	RG 1.97 Rev 2 Table 1		**RG 1.97 Rev 3 Table 2**		**IEEE-497 Consistent with RG 1.97 Rev 4**			**BWR/4 Typical**		
Variable	**Type**	**Cat.**	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
Containment and Drywell Hydrogen Concentration	C	1	C	1	N/A			A,C	1	Provided for severe accident mitigation. Commercial grade equipment is acceptable. Consistent with 10CFR50.44.
Containment and Drywell Oxygen Concentration (for inerted containment plants)	C	1	C	1	N/A			A,C	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 (from identified release points including Standby Gas Treatment)	C	3	C	3	E	N	N	C	3	Plant Deviation approved Cat 3.
Radiation Exposure Rate (including buildings or areas, e.g. auxiliary building, fuel handling building, secondary containment, which are in direct contact with primary containment where penetrations and hatches are located)	C	2			E	N	N	C,E	3	Plant deviation approved for Cat 3
Radiation Exposure Rate (including buildings or areas, e.g. auxiliary building, fuel handling building, secondary containment, which are in direct contact with primary containment where penetrations and hatches are located)			deleted	deleted	E	N	N			NRC approved deviation.

Table A-1 – Accident Monitoring Variables Comparison										
	RG 1.97 Rev 2 Table 1		RG 1.97 Rev 3 Table 2		IEEE-497 Consistent with RG 1.97 Rev 4			BWR/4 Typical		
Variable	**Type**	**Cat.**	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
Effluent radioactivity – noble gases (from buildings or areas where penetrations and hatches are located , e.g., auxiliary building, fuel handling building, secondary containment, which are in direct contact with primary containment)	C	2	C	2	E	N	N	C,E	3	**Typical** Plant **included as Type C and E consistent with RG 1.97 Rev 2 commitments.** ~~a~~Approved deviation **from Cat 2 to Cat 3**
Type D Variables										
Condensate and Feedwater Systems										
Main Feedwater Flow	D	3	D	3	N/A			D	3	Normal operating system.
Condensate Storage Tank Level	D	3	D	3	D	N	N	D	3	Normal operating system.
Primary Containment Related Systems										
Suppression Chamber Spray Flow	D	2	D	2	D	Y	Y	D	2	Approved deviations allows RHR system flow and valve position
Drywell Pressure	D	2	D	2	A, B, C	Y	Y	A,B,C,D	2	
Suppression Pool Water Level	D	2	D	2	A, B, C	Y	Y	A,B,C	2	
Suppression Pool Water Temperature	D	2	D	2	A, B, C	Y	Y	A,D	2	
Drywell Atmosphere Temperature	D	2	D	2	D	Y	N	D	2	.
Drywell Spray Flow	D	2	D	2	D	Y	Y	D	2	Approved deviation allows RHR system flow and valve position

Table A-1 – Accident Monitoring Variables Comparison										
	RG 1.97 Rev 2 Table 1		RG 1.97 Rev 3 Table 2		IEEE-497 Consistent with RG 1.97 Rev 4			BWR/4 Typical		
Variable	Type	Cat.	Type	Cat.	Type	EQ	SQ	Type	Cat.	Comments
Main Steam System										
Main Steamline Isolation Valves Leakage Control System Pressure	D	2	D	2	N/A			deleted		NRC approved **process for** elimination of MSIV leakage control system **in NEDC-31858P. Plant specific review needed to determine if variable is required.**
Primary System Safety Relief Valve Positions, Including ADS or Flow Through or Pressure in Valve Line	D	2	D	2	D	N	N	D	2	NEDO-33160 A contains NRC acceptance of change in requirements for SRV position indication
Safety Systems										
Neutron Flux					**B, D**	**N**	**N**			**See NEDO- 331558. Not identified as Type D in RG 1.97 Rev 2/3**
Control Rod Position					**D**	**N**	**N**			**Not identified as Type D in RG 1.97 Rev 2/3**
Isolation Condenser System Shell-Side Water Level	D	2	D	2	D	Y	Y	N/A		Applies plants with isolation condenser only.
Isolation Condenser System Valve Position	D	2	D	2	D	Y	Y	N/A		Applies plants with isolation condenser only.
RCIC Flow	D	2	D	2	D	N	N	D	2	RCIC required only for anticipated operational occurrences.
HPCI Flow	D	2	D	2	D	Y	Y	D	2	HPCI or HPCS flow.
Core Spray System Flow	D	2	D	2	D	Y	Y	D	2	LPCS system flow

Table A-1 – Accident Monitoring Variables Comparison										
	RG 1.97 Rev 2 Table 1		RG 1.97 Rev 3 Table 2		IEEE-497 Consistent with RG 1.97 Rev 4			BWR/4 Typical		
Variable	**Type**	**Cat.**	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
LPCI System Flow	D	2	D	2	D	Y	Y	D	2	
SLCS Flow	D	2	D	2	D	N	N	D	3	Standby liquid control system pumps running. System not required for anticipated operational occurrences or accidents.

Table A-1 – Accident Monitoring Variables Comparison										
	RG 1.97 Rev 2 Table 1		RG 1.97 Rev 3 Table 2		IEEE-497 Consistent with RG 1.97 Rev 4			BWR/4 Typical		
Variable	**Type**	**Cat.**	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
SLCS Storage Tank Level	D	2	D	2	D	N	N	D	3	System not required for anticipated operational occurrences or accidents.
Residual Heat Removal System										
RHR System Flow	D	2	D	2	D	Y	Y	D	2	
RHR Heat Exchanger Outlet Temperature	D	2	D	2	D	Y	Y	D	2	
Cooling Water Systems										
Cooling Water Temperature to ESF System Components	D	2	D	2	D	Y	Y	D	2	NRC approved as a deviation based on providing alternate means.
Cooling Water Flow to ESF System Components	D	2	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	D	3	N/A			D	3	Normal operating system.
Ventilation Systems										
Emergency Ventilation Damper Position	D	2	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 (hydraulic, pneumatic)	D	2	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 Rev 2 Table 1		**RG 1.97 Rev 3 Table 2**		**IEEE-497 Consistent with RG 1.97 Rev 4**			**BWR/4 Typical**		
Variable	**Type**	**Cat.**	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
Type E Variables										
Containment Radiation										
Primary Containment Area Radiation – High Range	E	1	E	1	E	~~N~~Y	~~N~~Y	E	1	~~Type E variable not Category 1~~ **Retained as RG 1.97 Revision 2 and 3 Type E Category 1 variable with requirements contained in NUREG 0737**.
Reactor Building or Secondary Containment Area Radiation										
10^{-1} R/hr to 10^{4} R/hr for Mark I and II containments	E	2	E	2	E	N	N	E	3	Reactor building area radiation.
1 R/hr to 10^{7} R/hr for Mark III containment	E	1	E	1	E	N	N	N/A	N/A	Mark III only
Area Radiation										
Radiation Exposure Rate (inside buildings or areas where access is required to service equipment important to safety)	E	2			E	N	N	E	3	Main control room and areas requiring access
Radiation Exposure Rate (inside buildings or areas where access is required to service equipment important to safety)			E	3						

Table A-1 – Accident Monitoring Variables Comparison										
	RG 1.97 Rev 2 Table 1		RG 1.97 Rev 3 Table 2		IEEE-497 Consistent with RG 1.97 Rev 4			BWR/4 Typical		
Variable	**Type**	**Cat.**	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
Airborne Radioactive Materials Released from Plant										
Noble Gases and Vent Flow Rate • **Drywell purge, Standby Gas Treatment System Purge (For Mark III plants) and Secondary Containment Purge (for Mark III plants)** • **Secondary Containment Purge (for Mark I, II and III plants)** • **Secondary Containment (reactor shield building annulus, if in design)** • **Auxiliary Building (including any building containing primary system gates, e.g. waste gas decay tank)** • **Common Plant Vent or Multi-purpose Vent Discharging Any of Above Releases (if drywell or SGTS purge is included)** • **All Other Identified Release Points**	E	2	E	2	E	N	N	E	2	Plant specific list. Includes all potential release points.
Particulates and Halogens. All Identified Plant Release Points. Sampling with	E	3	E	3	E	N	N	E	3	

Table A-1 – Accident Monitoring Variables Comparison										
	RG 1.97 Rev 2 Table 1		**RG 1.97 Rev 3 Table 2**		**IEEE-497 Consistent with RG 1.97 Rev 4**			**BWR/4 Typical**		
Variable	**Type**	**Cat.**	**Type**	**Cat.**	**Type**	**EQ**	**SQ**	**Type**	**Cat.**	**Comments**
Onsite Sampling Capability										
Environs Radiation and Radioactivity										
Airborne Radiohalogens and Particulates (portable sampling with onsite analysis capability)	E	3			E	N	N	E	3	Portable instrumentation can be used.
Plant and Environs Radiation (portable instrumentation)	E	3			E	N	N	E	3	Portable instrumentation can be used.
Plant and Environs Radioactivity (portable instrumentation)	E	3			E	N	N	E	3	Portable instrumentation can be used.
Meteorology										
Wind Direction	E	3	E	3	E	N	N	E	3	
Wind Speed	E	3	E	3	E	N	N	E	3	
Estimation of Atmospheric Stability	E	3	E	3	E	N	N	E	3	

Table A-1 – Accident Monitoring Variables Comparison										
	RG 1.97 Rev 2 Table 1		RG 1.97 Rev 3 Table 2		IEEE-497 Consistent with RG 1.97 Rev 4			BWR/4 Typical		
Variable	Type	Cat.	Type	Cat.	Type	EQ	SQ	Type	Cat.	Comments
Accident Sampling										.
Primary Coolant and Sump	E	3	E	3	~~N/A~~E	N	N	E	3	Grab samples. See PASS LTR **NED0-32991 A**
Containment Air	E	3	E	3	~~N/A~~E	N	N	E	3	Grab samples. See PASS LTR **NED0-32991 A**

ATTACHMENT 3
SRAI Summary and NEDO-33349 R1 Mark-up Changes

SRAI	ISSUE	RESPONSE	LTR Section
1	Drywell Pressure use for RPV pressure control	Confusion created by differences between EPGs and RG 1.97 Rev 2/3	N/A See 7.2
2	Drywell Sump Level previous acceptance conditions	Drywell sump level used for normal operations. Drywell temperature and pressure used for accidents and normal operation	N/A See 7.2
3	Fuel cladding fission barrier	Confirms RPV level is what operators rely on for fuel cladding integrity. Will retain RG 1.97 Rev2/3 CHARMs requirements	7.4.2
4	Additional key variables identified in LTR	Expand Table A-1 to include additional variables	Table A-1
5	Suppression chamber flow	Part of RHR system	N/A
6	Drywell Spray flow	Part of RHR system	N/A
7	RHR Heat Exchanger outlet temp	Part of RHR system	N/A
8	CIV position indication applicable systems	Plant specific analysis required	4.4
9	CIV position indication: not required for safety systems	No automatic action required.	Tables 5.1 and 5.2
10	MSIV Leakage Control	Reference provided	Table A-1
11	Type E release monitoring	Plant specific. All release points require monitoring. Table expanded	Table A-1
12	Table A-1: containment effluent monitoring	Table shows current plant RG 1.97 Rev 2 commitments	Table A-1
13	Estimation of Atmospheric Stability	Agree. Revise LTR and Tables	Tables 5.1, 5.2 and A-1
14	Containment air and primary coolant sampling	Type E variables referencing PASS LTR. Revise Tables	Tables 5.1, 5.2 and Table A-1
15	Include all variables in Table	Agee. Tables revised	Table A-1
16	SRAI letter includes Table. Review for consistency	Good summary: comments provided	N/A
17	Type C variable for fuel cladding: CHARMs	Confirms RPV level is what operators rely on for fuel cladding integrity. Will retain RG 1.97 Rev 2/3 CHARMs requirements	7.4.2