Duke Energy Company Oconee 1, 2, 3
Entergy Operations, Inc. ANO-1
Florida Power Corporation Crystal River 3

The B&W Owners Group

AmerGen Energy Company, LLC TMI-1
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Working Together to Economically Provide Reliable and Safe Electrical Power

June 25, 2001
OG-1809

Project No. 693

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington D.C. 20555-0001

Attention: Stewart Bailey

Subject: Submittal of Topical Report BAW-2387 – "Justification for the Elimination of the Post Accident Sampling System From the Licensing Bases of Babcock and Wilcox Designed Plants", June 2001

Dear Mr. Bailey:

Enclosed for NRC Staff review are fifteen (15) copies of Topical Report BAW-2387 entitled "Justification for the Elimination of the Post Accident Sampling System from the Licensing Bases of Babcock and Wilcox Designed Plants". The Licensing Working Group of the B&W Owners Group is submitting Topical Report-2387 at this time to support future requests by the utilities of the B&WOG for elimination of requirements for a PASS system. Members of the B&WOG intend to utilize this report upon successful completion of the report evaluation by the NRC

This document provides a basis for recommending the elimination of all Post Accident Sampling System (PASS) regulatory requirements at operating B&W designed plants. This report concludes that all PASS sample requirements may be eliminated. Flexible requirements may be substituted to (1) determine containment hydrogen concentration, (2) control post-accident reactivity, and (3) maintain RCS coolant at satisfactory pH levels. PASS is not needed to meet these requirements.

Elimination of PASS regulatory requirements on B&W designed plants is consistent with the conclusions of the NRC safety evaluations for eliminating PASS from the Westinghouse and Combustion Engineering designed nuclear power plants (and ANO-1, a B&W designed nuclear plant).

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14 copies sent to S. Bailey

Topical report BAW-2387 does not contain proprietary information. Please contact me at (864) 885-3077 or Robert Schomaker at Framatome ANP at (804) 832-2917 if you have any questions regarding this report.

Sincerely,

Noel T. Clarkson, III

Noel T. Clarkson,
Chairman
B&W Owners Group Licensing Working Group

Enclosures

BAW-2387
Topical Report
June 2001

The B&W OWNERS GROUP

Licensing Working Group

Justification for the Elimination of the Post Accident Sampling System from the Licensing Bases Of Babcock and Wilcox-Designed Plants

FRAMATOME ANP

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ABSTRACT

This document is a B&W Owners Group Topical Report prepared to help licensees eliminate their Post Accident Sampling System (PASS) with minimal need for plant specific Nuclear Regulatory Commission (NRC) reviews. The document provides a basis for recommending the elimination of all PASS regulatory requirements at operating B&W designed plants.

The Three Mile Island Unit 2 (TMI-2) accident resulted in the issuance of NUREG-0737, placing additional requirements on plant capabilities, with the intention of better enabling operators to characterize plant conditions during and following severe accident situations. The requirements were satisfied by the creation of a Post Accident Sampling System (PASS) that allowed Reactor Coolant System (RCS) fluids, containment sump, and containment atmosphere to be sampled for analysis to monitor accident progression.

Subsequent operating experience, reanalysis, and consideration of currently available instrumentation have all contributed to a more thorough understanding of the conditions of the core and reactor coolant system during an accident. This better understanding demonstrates that use of PASS does not enhance the plant response to a severe accident primarily because of the delay times between sampling and analysis relative to the time to core damage in certain events and non-conservative measurement of the radionuclides due to transport and deposition. It suggests that PASS use may even have a negative effect on plant response by unnecessarily exposing personnel to radiation, opening potential leakage paths for fission products, providing potentially misleading information, and occupying plant personnel with procedures which may consume precious time while yielding little in the way of additional useful information.

Costs to upgrade and maintain the PASS have the potential to shunt limited financial and manpower resources from more safety beneficial tasks. The costs and risks do not justify the minimal potential for information gain.

A review of the plant specific accident response procedures provides assurance that PASS elimination does not compromise the efficacy of the response. Reviews of Emergency Operating Procedures (EOP), Severe Accident Management Guidelines (SAMG), Emergency Plans (EP) (including Core Damage Assessments methodology) reveal that PASS capabilities are seldom used and tend to be used as supplemental information. Often this is due to the availability of better instrumentation or methods, such as the in-line RCS and containment instrumentation designed for "harsh" environments.

Therefore, this report recommends that the regulatory requirements for PASS be eliminated.

1.0 INTRODUCTION

This Topical Report is intended to enable participating licensees to eliminate their Post Accident Sampling System (PASS) with minimal need for plant specific Nuclear Regulatory Commission (NRC) reviews.

1.1 Overview

The purpose of each section of this report is outlined below:

Section 1 provides, in addition to this overview, the objective of this report and the background of PASS including industry efforts to eliminate the regulatory requirement for PASS.

Section 2 summarizes the report recommendations and states how those recommendations are to be incorporated.

Section 3 provides the results of a review of plant specific accident management documents, which demonstrates the lack of use of PASS and that PASS is not needed. The lack of use is often due to the availability of better instrumentation or real time evaluations. The accident management documentation includes Emergency Operating Procedures (EOP), Severe Accident Management Guidelines (SAMG), Emergency Plans (EP), Abnormal Operating Procedures (AOP), and Core Damage Assessments (CDA).

Section 4 provides an evaluation of the effect of PASS elimination to demonstrate that parameter measurement is unnecessary or that the purpose of the measurement can be better satisfied in another way.

Section 5 lists the documents referred to in this report.

Appendix A contains a review of plant-specific SAMGs.

Appendix B contains a review of plant-specific EOPs.

Appendix C contains a review of plant-specific EPs.

Appendix D contains a review of plant-specific Core Damage Assessment procedures.

1.2 Objective

This document supports the elimination of regulatory requirements for the PASS used to sample the Reactor Coolant System (RCS) fluids, containment sump and containment atmosphere. Justification is provided for the position that accident classification, response, and mitigation will not be adversely affected by the elimination of PASS sampling, nor will there be any degradation of emergency preparedness procedures or actions.

1.2.1 Background

Following the Three Mile Island Unit 2 (TMI-2) accident, root cause analysis led to the conclusion that core and reactor coolant system conditions could not be determined from the indications available to plant operators. Subsequent issuance of NUREG-0737 placed additional requirements on plant capabilities, with the intention of better enabling operators to characterize plant conditions during and following severe accident situations. A severe accident is one involving catastrophic fuel rod failure, core degradation, and fission product release into the reactor vessel/containment/environment. The requirements were satisfied by the creation of a PASS that allowed the RCS fluids, containment sump, and containment atmosphere to be sampled and analyzed for radionuclide concentrations which could be used to monitor accident progression.

Additional years of operating experience, reanalysis, and consideration of currently available instrumentation have all contributed to a more thorough and better understanding of the conditions of the core and reactor coolant system during an accident and the behavior of radionuclides. This better understanding demonstrates that use of PASS does not enhance the plant response to a severe accident. It suggests that PASS may even have a negative effect on plant response by unnecessarily exposing personnel to radiation, opening potential leakage paths for fission products, and occupying staff with procedures which may consume precious time while yielding little in the way of additional useful information and may lead to a non-conservative assessment of core damage.

In addition, plant specific accident response procedures (i.e., Emergency Operating Procedures (EOP), Severe Accident Management Guidelines (SAMG), Emergency Plans (EP)) seldom use PASS, and when used, it is used primarily as supplemental information. Often this is due to the availability of better instrumentation or methods, such as the in-line RCS and containment instrumentation designed for "harsh" environments.

Because PASS is not needed to manage accidents, because PASS provides little benefit and because of the resources required to maintain it, an industry initiative has been undertaken to eliminate the regulatory requirement for PASS. The initiative has resulted in the NRC concluding that there is reasonable assurance that the health and safety of the public will not be endangered by operation of nuclear power plants without PASS and that it is acceptable for licensees to eliminate PASS from the plant licensing basis. This NRC conclusion is provided in the Safety Evaluation for the Combustion Engineering designed plants as discussed in the Combustion Engineering Owners Group report CE NPSD-1157-A and for Westinghouse designed plants as discussed in Westinghouse Owners Group report WCAP-14986-A, Rev. 2 and WCAP-14696-A, Rev.1

2.0 SUMMARY AND CONCLUSIONS

This report provides a basis for recommending the elimination of all PASS regulatory requirements at B&W designed plants.

Considering the limitations and risks inherent in the sampling process that restrict the potential positive contribution PASS can make, the process of assessing core condition during a severe core accident is better performed by existing plant instrumentation. Plant instrumentation within the RCS and containment is part of the complement of inadequate core cooling instrumentation, designed for "harsh" environments, and expected to survive core damage accidents. Compared to plant monitors, sampling requires more time, often results in a measurement that is not representative and non-conservative, risks sample line plugging and iodine plate-out or release, and exposes personnel to greater radiological dose.

Periodic sampling cannot continuously monitor parameters. In fact, the time needed to acquire and analyze samples effectively limits the amount of useful information that can be gathered during the rapidly changing conditions that exist during core damage accidents. With little information of immediate usefulness to be gained, the sampling process could actually distract staff from more beneficial tasks. The small information gain does not outweigh the risks associated with its use or the cost to maintain PASS. The Westinghouse SER concludes " .... that the use of fixed plant instruments in the manner described in the CDAG provides an acceptable alternative to radiochemistry analysis of a radionuclide sample to obtain an approximate estimate of the extent of core damage during the transient phase of an accident."

Therefore, this report concludes that all PASS sample requirements may be eliminated. Flexible requirements may be substituted to (1) determine containment hydrogen concentration, (2) control post-accident reactivity, and (3) maintain RCS coolant at satisfactory pH levels. PASS is not needed to meet these requirements.

Consistent with the conclusions of the NRC safety evaluation for eliminating PASS from the Westinghouse and Combustion Engineering designed nuclear power plants (and ANO-1, a B&W designed nuclear power plant), this report recommends that regulatory requirements for PASS also be eliminated from all operating B&W designed nuclear power plants.

Acceptance of this recommendation will remove unnecessary regulatory burdens. These burdens include (1) the requirements for time period following an accident when a sample must be available, (2) requirements for sample accuracy, (3) requirements for the sample location, (4) requirements for demonstration of sample capability following a core damage accident, and (5) requirements for in-place procedures and demonstrable sampling methods.

Implementation by member utilities will involve a thorough review of plant emergency response, including specific documents and procedures used, in order to remove all references to PASS. Specifically, Emergency Operating Procedures (EOP), Severe Accident Management Guidelines (SAMG), Emergency Plans (EP), Abnormal Operating Procedures (AOP), and Core Damage Assessments (CDA) will need to be revised. Implementation of the recommendation to eliminate PASS regulatory requirements will also necessitate the revision of those plant documents using applicable plant change procedures. Review by each individual utility of the plant specific accident response procedures provides assurance that this action does not compromise the efficacy of the response.

3.0 REVIEW OF ACCIDENT MITIGATION GUIDANCE

The Post Accident Sampling System is used to provide information for managing accidents. The accident management procedures were reviewed for the Three Mile Island (TMI), Crystal River Unit 3 (CR-3), Oconee Nuclear Station (ONS) and Davis-Besse (DB) Nuclear Power Plants to determine how the procedures use the PASS. The purpose of the review was to determine what accident mitigation actions rely on information provided by the PASS. The review did not investigate whether or not the PASS is used in another capacity, which could be part of the licensing bases. Such a review would have to be conducted by the individual utilities.

The procedures for managing accidents are divided into three categories:

a. Emergency Operating Procedures,
b. Severe Accident Management Procedures and
c. Emergency Plan Implementation Procedures

The results of the review for each of the three categories are provided in the following sections and summarized in Appendix A. Appendix A is a broad listing of EOP steps that may eventually lead to a sample, but in most cases such samples are not required for operator actions.

3.1 Emergency Operating Procedures

The NRC requested symptom oriented Emergency Operating Procedures to be implemented following the TMI-2 accident. The B&W Owners Group developed generic Emergency Operating Procedure (EOP) Technical Bases. B&W plant Emergency Operating Procedures are derived from these EOP Technical Bases. The EOP Technical Bases are specific to B&W plant design in identifying accident in progress and providing the plant operators specific accident mitigating instruction based on symptoms and trends of the event in progress. Each utility used the EOP Technical Bases to develop a set of plant specific Emergency Operating Procedures (EOPs). As a consequence, the plant specific EOPs are similar. They monitor similar system parameters and use the parameters in similar ways.

In order to identify the role of the PASS in the plant EOPs, a review of individual utility EOPs was performed. The plant EOPs were reviewed to identify which of the parameters that can be measured by PASS are specified in EOP action statements. Appendix A tabulates the results of this review, which is a broad listing of EOP steps that may eventually lead to a sample, but in most operator actions are specified in the EOPs without the need for sample results. In some instances, the EOPs incorporate specific Abnormal Operating Procedures (AOPs) by reference as part of the mitigating procedure. The review included such AOPs.

In general, the EOPs do not directly use the PASS for making transient mitigation decisions. Wherever possible, they use in-plant instrumentation because of the general need for making decisions as quickly as possible. When samples are taken they rely on the normal sampling system wherever possible because of the familiarity of the normal sample system.

The only parameters that can be measured by the PASS that were used by the EOPs at all plants are:

RCS boron
Containment atmosphere hydrogen and
Containment sump boron

In addition to the above, RCS and containment radionuclides are measured in the CR-3 EOPs, containment sump radionuclides are measured in the TMI EOPs, and the containment sump pH is measured in the ONS and TMI-1 EOPs.

3.1.1 RCS Boron

The EOPs specify measuring the RCS boron concentration for three specific situations and for a general assessment of the boron status. The specific situations include:

a. RCS cooldown
b. RCS dilution due to addition of a non-borated water source, and
c. Stuck control rod(s)

In each situation, the boron concentration is used to confirm adequate shutdown margin to protect the core from a return to criticality.

Although the EOPs specify measuring RCS boron, the inability to obtain boron concentration measurements would not result in an unsafe plant condition. During most RCS cooldown scenarios, the option is provided to start boration if boron concentration information is not available rather than wait for boron concentration measurement results. Except for natural circulation cooldown, where procedures may not allow cooling below a minimum temperature unless adequate boron concentration is confirmed, the cooldown can continue with addition of boric acid and no confirmation of boron concentration.

For an uncontrolled cooldown or dilution, the approach to criticality would be slow such that other corroborative indications to verify that the reactor is shut down and will remain shutdown can be used such as control rod position indication, reactor power decreasing or stable and positive indication of flow into the RCS of high boron concentration fluid. These indications would prompt the operator to add more boron. The RCS system and its accident mitigating features are designed such that adequate shutdown margin is assured following a reactor trip and one stuck control rod drive mechanism assembly (CRDM). The failure of more than one CRDM would be indicated to prompt the operator to provide additional boron to the RCS. In addition, to accommodate RCS fluid contraction during cooldown and to makeup for RCS inventory losses due to leaks, borated water will be added by various emergency cooling systems such as high pressure injection

(HPI), low pressure injection (LPI), and core flood tank (CFT). These sources all contain a high concentration of boron to assure core subcriticality is maintained.

In general, the letdown line will not be isolated and the normal sampling system is expected to support the sampling requirements recommended in the EOPs and not the PASS. The normal sampling system can be used for the vast majority of accidents anticipated since only minimal fuel failure is expected. Incidents with fuel failures significant enough to prevent use of the normal sampling system are likely to be the result of LOCAs and the shutdown margin should be adequately controlled for these events via the injection of highly borated ECCS water.

3.1.2 Containment Hydrogen

The EOPs require trending the containment hydrogen concentration for initiation of the containment hydrogen control system and for a general assessment of the containment hydrogen status. This measurement determines the possibility of approaching flammable concentration of hydrogen in the containment so that actions can be initiated to alleviate this condition. However, should the action to initiate the hydrogen control system not be taken, the health and safety consequences of any potential burn are negligible.

Although the PASS can measure the containment hydrogen concentration, the EOPs rely on the containment hydrogen monitors for that capability.

3.1.3 Containment Sump Boron

The EOPs specify sampling the containment sump water to detect boron dilution due to addition of any non-borated water to the containment and for a general assessment of the boron status. The boron measurement is to assure adequate concentration to protect the core from a return to criticality during containment sump recirculation. Although the EOPs specify measuring RCS boron concentration, the inability to obtain the measurements would not result in an unsafe plant condition. For a dilution occurrence, other corroborative indications to verify that the reactor is shutdown and will remain shutdown can be used such as control rod position indication, reactor power decreasing or stable and positive indication of flow into the RCS of high boron concentration fluid.

Without the PASS, the sump water boron concentration can be readily and adequately estimated by knowing the amounts of water added to the sump and their respective boron concentrations.

3.1.4 Other Uses

CR-3 EOPs specify measuring RCS and containment atmosphere radionuclides. The RCS radionuclides are measured only to assess radionuclide status and do not direct any operator actions. The steam generator is isolated for tube rupture accident that uses radionuclide information to assess dose equivalent I^{131} status. The containment atmosphere radionuclides are measured to determine if the I^{131} concentration is low enough to permit stopping the containment spray. The operator can leave the spray pumps on if the concentration cannot be measured.

The TMI-1 EOPs specify measuring the containment sump radionuclides. The containment sump radionuclides are measured only to assess radionuclide status and do not provide initiating conditions for operator actions.

The ONS and TMI EOPs specify measuring the containment sump pH. In the ONS EOPs, the containment sump pH is measured only to assess sump pH status and does not provide any initiating conditions for operator actions. In the TMI-1 EOPs, the containment sump pH is measured to determine if NaOH should be added to increase pH. If the pH cannot be measured, then the pH value can be estimated with a sufficient degree of accuracy from the volumes and chemistries of water going to the sump.

3.2 Plant Severe Accident Management Procedures

3.2.1 Background

B&W produced a generic guide for use by the owners of B&W-type plants in developing plant-specific severe accident management procedures. The document is intended to provide guidance to personnel in the control room and Technical Support Center (TSC) in the event of a severe accident. A severe accident is one involving catastrophic fuel rod failure, core degradation, and fission product release into the reactor vessel/containment/environment.

3.2.2 Guideline Summary and Use

The purpose of the plant Severe Accident Management Guidelines (SAMG) (or SAG) is to rapidly guide the user through a structured process in the event of a severe accident. This process seeks to determine the status of the reactor coolant system and of containment by detecting one or more of the symptoms expected to result from possible plant damage conditions. The condition of the reactor coolant system is categorized as badly damaged (BD) or ex-vessel (EX). The condition of the containment is categorized as closed and cooled (CC), challenged (CH), impaired (I), or bypassed (B). With the condition of the plant determined, the next process step is the selection of one or more coping strategies from a prioritized list of Candidate High Level Actions (CHLA). Each CHLA is intended to limit core, vessel and containment damage, return the plant to a controlled state, and prevent or minimize the release of radiation to the public.

In addition to the prioritized list of immediate actions, the SAMG includes discussions of long-term concerns as additional considerations. The long-term concerns described in the SAMG do not have any impact on the immediate actions to control the accident situation. The SAMG merely acknowledges that performance of CHLAs may cause, for example, changes in water chemistry if non-reactor coolant grade water sources are used. Extended use of these sources will eventually require that water chemistry be addressed in order to prevent degradation of plant components over a relatively long period of time.

CHLAs for effective accident management will always take precedence over any actions to address long-term concerns. Therefore, SAMG does not require sampling and analysis to evaluate these parameters.

The SAMG includes calculation aids and forms for collecting and recording measured plant parameters. These are useful in trending analysis to confirm CHLA efficacy and to identify any negative effects of a CHLA.

It is recognized that during the course of the accident management process, there may be times when information is insufficient or needed equipment is unavailable. At such times, the SAMG allows the performance of generic actions. Generic actions seek to preserve the next radiological barrier and do not require determination of a particular plant condition. When the delayed information becomes available, allowing a reliable plant assessment, execution of the appropriate SAMG CHLA may begin or resume.

Parameter measurement to detect the symptoms of a plant damage condition relies on fixed in-plant instrumentation in preference to manual sampling and analysis. The reasons are that sampling requires more time, often results in a measurement that is not representative and non-conservative, risks sample line plugging and iodine plateout or release, and exposes personnel to greater radiological dose. The process sampling and analysis for radionuclides are simply too slow to provide useful information about the rapidly changing conditions that exist during core damage accidents. SAMGs are designed for use from the time serious core overheating is detected until a controlled stable state is reached. This period of time is dominated by rapidly changing transient conditions, which the slow process of acquiring and analyzing samples cannot rapidly and continuously monitor. Alternative instrumentation is available in the form of the in-line instrumentation of the RCS and containment. RCS and containment instrumentation are part of the complement of inadequate core cooling instrumentation, designed for "harsh" environments, and expected to survive core damage accidents. Therefore, except for the monitoring of hydrogen concentration, SAMG requirements for PASS do not exist.

3.2.3 PASS Parameters Considered By Severe Accident Management Guidelines

To determine if sampling with PASS is required during severe accident mitigation, plant specific reviews of SAMG documents were performed. The results are tabulated in Appendix B, Review of Plant Severe Accident Management Guidelines. Individual plants may have other plant specific licensing bases relating to PASS which will need to be addressed by each plant.

Considering the parameters that can be measured by PASS, only containment hydrogen is measured. The following paragraph explains why SAMG requires that parameter to be measured and how that measurement is used. Use of PASS in the measurement process is not specified, or is discouraged due to radiological hazards. Other methods exist to determine the parameter, and the inherent advantages of these other methods make them preferable to the use of PASS.

3.2.3.1 Containment Hydrogen

Oxidation of fuel clad material at the high temperatures experienced during core damage accidents generates hydrogen that can accumulate in the containment. Sources outside the RCS can also cause hydrogen to accumulate in the containment. Possible sources include the chemical reaction of boron with aluminum, and radiolytic decomposition of water. Increasing levels of hydrogen in the containment may subsequently ignite. Hydrogen combustion results in a containment pressure increase. This is one of the ways containment integrity can be challenged. The actual risk is determined by the concentrations of hydrogen and steam. These concentrations are evaluated using computational aids to quantify the risk to containment to determine appropriate mitigating actions. Bounding conditions, on-line containment hydrogen monitors, or gas sample analyses are acceptable methods for determining the hydrogen concentration. No specific method is identified by the SAMGs.

3.3 Emergency Plan Implementing Procedures

A review was made of various Emergency Plan Implementation Procedures (see Appendix C) and Core Damage Assessment Procedures (see Appendix D) to assess how the PASS is being used in Site Emergency Plans at the ONS, TMI, DB and CR-3 nuclear plants. In general, the Post Accident Sampling System may be used to a limited extent for classifying accidents, for assessing core damage, and for predicting offsite dose.

3.3.1 Classifying Accidents

The plant Emergency Plans use various Emergency Action Levels (EALs) to classify accidents into one of four emergency classes: Notification of Unusual Event, Alert, Site Area Emergency, and General Emergency. For each emergency class, the Emergency Plans specify the appropriate Protective Action Responses (PARs). Among the various EALs are a few based on the amount of radionuclides in the reactor coolant as a quantifier of fuel damage for which PASS could be used to determine. These EALs are:

a. the reactor coolant activity exceeding technical specification limits and
b. the reactor coolant activity exceeding 300 μCi/gram dose equivalent I^{131} (about 2-5% failed fuel).

These EALs are based on recommendations in NUREG-0654/FEMA REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" and NUMARC NESP-007, "Methodology for Development of Emergency Action Levels". NUREG-0654 proposes an EAL value equal to the plant technical specification limit for reactor coolant activity associated with iodine spiking as a criterion for declaring an "Unusual Event" classification. The NUREG also uses a value of 300 μCi/gram dose equivalent I^{131} in the reactor coolant as a criterion for declaring an "Alert" classification. Using the 300 μCi/gram dose equivalent I^{131} emergency action level is also discussed in NUMARC

NESP-007. (If other EALs are present the emergency classification could be higher than "Unusual Event" or "Alert" classifications.)

To determine if the EAL for reactor coolant radionuclide values exceed the plant technical specification limits, the normal sample system would be used rather than PASS. The PASS would normally be used to determine if the reactor coolant radionuclide values exceed the 300 μCi/gram dose equivalent I^{131}. Therefore, the present Emergency Plans assume the availability of PASS to support the emergency action level classification.

If fuel overheating occurs when the reactor coolant radionuclide values exceed the 300 μCi/gram dose equivalent I^{131}, then other indicators such as the core exit thermocouples would be used to assess the extent of fuel damage. Therefore, the only events for which PASS could provide value would be events, such as reactivity excursion or mechanical damage, which cause some cladding damage with radionuclide values exceeding 300 μCi/gram dose equivalent I^{131} , but not causing an indication of fuel overheating. However, for these events, other indicators of failed fuel can be correlated to the degree of failed fuel such that PASS would not be needed. These indicators could include letdown radiation monitors (or normal sampling system). In addition, these other indicators would also tend to be better for determining an emergency classification because they can provide a quicker assessment of fuel damage than PASS can (due to the timing issues associated with taking PASS samples, i.e., sample line flushing).

The results of plant-specific emergency classification procedures are provided in Appendix C.

3.3.1 Assessing Core Damage

The plant Emergency Plans use various methods to assess core damage. These methods include:

a. assessing core damage based on core exit thermocouple indications,
b. containment radiation monitor readings,
c. containment hydrogen concentration and
d. radionuclide analysis of reactor coolant.

The first two methods (a and b) use plant instrumentation and therefore do not rely on PASS. The third method (c) uses the post accident hydrogen monitors and therefore, does not rely on PASS. The core exit thermocouples and hydrogen monitors can not be used for assessing very low core damage conditions. The containment radiation monitors may detect very low core damage conditions. However, for these conditions normal plant systems would be used. Thus PASS would not be used. These plant systems could include the normal sample system or letdown radiation monitors. Their measurements would be correlated to the degree of core cooling.

The last method (d) relies on PASS to obtain and analyze a reactor coolant sample. In general, the last method is used only as a confirmatory method. Obtaining a PASS sample is not considered an urgent or high priority task since the other methods of core damage assessment provide adequate information for all short-term actions. In addition the other methods are simpler and more expeditious to perform. The PASS sample was originally intended to provide a more precise assessment of core damage. However, now that the post accident sampling processes are better understood, it is recognized that the sampling process has inaccuracies such that a core damage assessment made by PASS may be no better than one made by the other methods.

The results of a review of plant-specific core damage assessment procedures are provided in Appendix D.

3.3.2 Predicting Offsite Dose

Protective Action Responses (PARs) of the Emergency Plan are, in part, dependent on offsite dose assessments. The methods for making offsite dose assessments during the early phases of an accident generally use radiation monitors or some other real time instrument. Subsequent offsite dose assessments, during and after release, will use radionuclide measurements made at the release point or by field teams. In general, PASS measurements of the containment atmosphere radionuclides would be used to predict what the offsite dose would be if the containment atmosphere were to be released. However, should PASS not be available, other methods are available for estimating the radionuclide content of the containment atmosphere. Therefore, the only function for PASS in regard to offsite dose assessment, would be confirmatory information after the plant has stabilized. For this particular function a dedicated PASS sample system can be replaced by contingency plans for obtaining and analyzing highly radioactive samples of the containment atmosphere. The plans would detail the plant's existing sampling capabilities and what actions (e.g., assembling temporary shielding) may be necessary to obtain and analyze highly radioactive samples. The contingency plans would not have to be demonstrated.

4.0 EVALUATION OF PASS ELIMINATION

The PASS is designed to sample different parameters for various purposes. This section assesses each parameter that is measured. The assessment includes a discussion of the purpose for measuring each parameter and a justification as to why the parameter does not need to be measured by PASS. Each assessment also includes the NUREG 0737 and Regulatory Guide 1.97 requirement. In each assessment, the justification shows either the measurement is not needed or the purpose of the measurement can be satisfied by an alternate method. Table 4-1 provides a summary of assessments for PASS parameters. The table indicates if the parameters need to be measured by PASS and, if not, the table provides a justification as to why it is not needed. As indicated by the table, none of the parameters need to be measured by PASS. Although not needed to be sampled, sampling some parameters could be beneficial, As shown by the table, sampling is only beneficial during long term cooling and not for all parameters. If desired, these beneficial sample measurements could be accomplished using non-dedicated sampling equipment that does not have the strict requirements of PASS.

4.1 RCS Dissolved Gases

The purpose of sampling the reactor coolant system (RCS) for dissolved gases is to determine if the potential for void formation within the RCS highpoints (vessel dome and hotlegs) exists due to dissolved gases coming out of solution during system depressurizing. The void formations can lead to uncovering the core or prevent natural circulation.

Dissolved gas sampling is required by NUREG-0737 and Reg. Guide 1.97. However, NUREG/CR-4330 indicates that this requirement can be eliminated based on installation of RCS highpoint vent systems and reactor vessel level instrumentation.

Knowledge of the dissolved gas content would not prevent void formation, nor aid the operator in the elimination of the void. The RCS void formation will result in inadequate core cooling. This condition is easily eliminated by the RCS highpoint vent system. Procedures and training can be instituted for detection of void formation and elimination.

Currently, plants either have an automatic gas sampling system or manually perform this function. In either case, the sample results are not timely and not accurate due to small sample sizes that are taken to minimize undue exposure. Therefore, it has no practical significance in accident monitoring and management.

Thus, following an accident, sampling for dissolved gases in the RCS is not required to achieve a safe and stable state. Based on the above discussion of the need and practical use of post accident sampling capabilities for reactor coolant dissolved gases, this function of the PASS system can be eliminated from all operating B&W-designed plants.

4.2 RCS Hydrogen

The purpose of sampling the reactor coolant system (RCS) for dissolved hydrogen is to determine if the potential for void formation within the RCS highpoints (vessel dome and hotlegs) exists due to non-condensable gases coming out of solution during cooldown and depressurization. The void formation can disrupt natural circulation cooling that might be used for post accident long-term decay heat removal.

Dissolved gas sampling is required by NUREG-0737 and Reg. Guide 1.97. However, NUREG/CR-4330 indicates that this requirement can be eliminated based on installation of RCS highpoint vent systems and reactor vessel level instrumentation.

The amount of dissolved hydrogen can be a leading indicator of dissolved fission products and non-condensable gases due to fuel cladding deterioration during an accident. For plants with highpoint vents, hot leg level instrumentation, and the Reactor Vessel Level Instrument System (RVLIS) or continuous head-to-hot leg vent line (Davis-Besse), RCS void formation is easily detected. Thus, knowledge of the hydrogen concentration would not prevent void formation, nor aid the operator in the elimination of the void. Procedures and training can be instituted for detection of void formation and elimination using the RVLIS system when depressurizing to promote natural circulation mode of cooling. For plants with continuous head-to-hot leg vent line, void formation in the vessel dome is eliminated by design.

Thus, following an accident, sampling for hydrogen concentration in the RCS is not required to achieve a safe and stable state. Based on the above discussion of the need for post accident sampling capabilities for reactor coolant hydrogen concentration, this function of the PASS system can be eliminated from all operating B&W-designed plants.

4.3 RCS Oxygen

The purpose of sampling the reactor coolant system for dissolved oxygen content is to assess the potential for stress corrosion cracking of the RCS stainless steel piping promoted by high chlorides concentration.

Post accident sampling for determination of oxygen concentration in the RCS is recommended by NUREG–0737 and is required by Reg. Guide 1.97. This requirement is imposed whenever the RCS chloride concentration exceeds 0.15 ppm.

The oxygen concentration in the presence of chlorides will enhance stress corrosion cracking of the RCS stainless steel components (long term application). High chloride concentration will also cause low pH. The RCS pH control is performed by either automatic addition of buffering solutions through the containment spray system or by adding trisodium phosphate to the sump that is circulated into the reactor.

Sampling for oxygen concentration is not used for RCS pH control or any other accident mitigating measures. Therefore, post accident sampling capabilities for reactor coolant oxygen concentration can be eliminated from all operating B&W-designed plants.

4.4 RCS pH

The purpose of sampling the reactor coolant system for pH is to determine whether a chloride induced stress corrosion cracking of the stainless steel components will occur and to assure that radioactive iodine is retained in the coolant.

The reactor coolant system post accident sampling for pH is not required by NUREG–0737, however, it is a Reg. Guide 1.97 requirement.

Low pH values (below 7.0) is a leading indication of potential for long term stress corrosion cracking of the stainless steel components (long term application). The reactor coolant water with a low pH in conjunction with the presence of chlorides will enhance stress corrosion cracking of the RCS stainless steel components. Another reason to determine RCS pH is that it provides an indication of radioactive iodine retention potential.

The RCS pH control is typically performed by either an automatic addition of buffering solutions through the containment spray (sodium hydroxide) system or by adding trisodium phosphate to the sump that is circulated back into the reactor.

The RCS pH measurement is not required for any accident mitigating action. Therefore, post accident sampling capabilities for reactor coolant pH can be eliminated from all operating B&W-designed plants.

4.5 RCS Chlorides

The purpose of sampling the reactor coolant system for chloride concentration is to assist and assure that chloride induced stainless steel stress corrosion cracking will not occur in the long term.

Sampling for chloride concentration is required by NUREG-0737 and Reg. Guide 1.97.

It has been shown that high concentration of chlorides in the reactor coolant system will cause stress corrosion cracking of the stainless steel components (long term application). During an accident chlorides may be introduced into the primary coolant by external sources of untreated or non-demineralized water such as sump water, brackish river water, or seawater to continue injection of water into the reactor coolant system.

Since the use of external sources of water is a deliberate operator action, the chloride concentration of the resultant circulation reactor coolant water can be estimated and adjusted using buffering solutions (pH control) to prevent long term stainless steel corrosion. Adjusting the RCS pH using buffering solutions such as sodium hydroxide or trisodium phosphate additives does not require chloride concentration measurement. Furthermore, the addition of passive or active buffering solutions to control reactor coolant pH is a predetermined parameter derived from conservatively calculated coolant composition and RCS conditions.

The RCS chloride concentration measurement is not required for any accident mitigating action or RCS chlorides control action, therefore, post accident sampling capabilities for reactor coolant chloride concentration can be eliminated from all operating B&W-designed plants.

4.6 RCS Boron

The purpose of obtaining the reactor coolant boron concentration is to assure that adequate shutdown margin is being maintained in the core and no possibility of return to criticality exists during cold shutdown process.

The post accident sampling of the reactor coolant system for boron concentration is required by NUREG-0737 and Reg. Guide 1.97.

Plant emergency operating procedures do not rely on RCS boron concentration measurement to maintain shutdown margin. The B&W plant EOPs provide adequate measures for RCS boration to maintain shutdown margin through accident mitigation and accident recovery stages without relying on post accident sampling system's boron measurement. In addition, alternate indications of adequacy of post accident shutdown margin are control rod drive position indication, reactor power indication, negative startup rate indication, and concentrated boron addition indication.

Based on the above discussions the RCS boron measurement is not required for any accident mitigating action, therefore, post accident sampling capabilities for reactor coolant boron measurement can be eliminated from all operating B&W-designed plants.

4.7 RCS Conductivity

The purpose of measuring the reactor coolant system conductivity is to verify RCS pH measurement.

The post accident sampling the reactor coolant system for conductivity measurement is not required by NUREG-0737 or Reg. Guide 1.97.

A review of B&W plants accident management guidelines or emergency plans reveals that the RCS conductivity is not utilized.

The RCS conductivity is not required for any accident mitigating action, therefore, post accident sampling capabilities for reactor coolant conductivity can be eliminated from all operating B&W-designed plants.

4.8 RCS Radionuclides

The purpose of sampling the reactor coolant system for radionuclides is to determine the integrity of the fuel cladding during an accident.

Post accident sampling of the reactor coolant system for radionuclides measurements is required by NUREG-0737 and Reg. Guide 1.97.

The reactor coolant radionuclide measurement will support the accident classification scheme used by plants. Exceeding a preset limit of 300 μCi/cc (equivalent I^{131}) RCS iodine concentration is an indication of 5 to 10% fuel cladding failure. In addition to RCS radionuclide measurement, other indications exist that provide early warning of cladding failures and an indication to escalate emergency classification. Other indicators that support accident escalation level are high core exit thermocouple readings, low reactor core level indication, high containment radiation level indication, loss of sub-criticality, loss of RCS subcooling margin, and high RCS letdown radiation level.

In some special cases methods listed above may not provide indication of fuel failure (e.g. fuel cladding failures due to debris induced mechanical failures). In such cases the normal sample system can be used for EAL classification for conditions equal or less than 300 μCi/cc. If does rates preclude getting a sample from the normal sample sink, then a minimum of an Alert classification would be obvious.

The use of alternate measures described above will typically result in a more conservative evaluation of accident classification. The use of radionuclide samples will not only result in a less expedient and less timely process, it will result in added exposure to plant personnel.

Based on the above discussion and justifications for alternate measures available to classify accident levels, it is recommended that the post RCS radionuclide measurement function can be eliminated from all operating B&W-designed plants.

4.9 Containment Atmosphere Hydrogen

The capability to remove grab samples of containment atmosphere for analysis of hydrogen levels in the containment atmosphere is specified in NUREG 0737 item II.B.3 and in RG 1.97 revision 3. NUREG 0737 specifies a capability to obtain and analyze containment hydrogen within 3 hours. The purpose of measuring the containment hydrogen concentration is to trend the accumulation of hydrogen to determine the potential for hydrogen combustion in the containment.

Containment hydrogen concentration monitors are required by 10CFR50.44(b)(1), NUREG-0737, and regulatory Guide 1.97. These monitors are generally relied upon to meet the data reporting requirements of 10CFR50 Appendix E. These monitors are required to be functional within 30 minutes after the initiation of safety injection and have a range of 0 to 10 volume percent.

Based on specific containment design and generally larger containment size some B&W plant licensees have obtained regulatory relief for the 30-minute time limit. The hydrogen monitor operability time limit has been replaced with a functional requirement that allows the licensee the flexibility to determine the appropriate time limit for providing indication of hydrogen concentration in the containment. Once the need for a hydrogen measurement has been determined, the installed monitors would provide a measurement quicker than a PASS sample.

The hydrogen monitors can be used instead of PASS to determine the containment atmosphere hydrogen concentration during the initial phases of an accident. However, the potential exists for the hydrogen concentration to exceed the range of the hydrogen monitors in the long term. If the emergency management guidance relies on knowing the containment hydrogen concentration for concentrations above the range of the hydrogen monitors (off-scale) then contingency plans should be provided for obtaining and analyzing such samples (long term application). However, maintaining dedicated equipment for measuring off-scale samples is not necessary.

Based on the above assessment of the need for post accident sampling of containment atmosphere hydrogen, the containment atmosphere hydrogen sampling function is considered unnecessary and can be eliminated from all operating B&W-designed plants. If the emergency management guidance relies on knowing the containment hydrogen concentration for concentrations above the range of the hydrogen monitors then contingency plans can be provided for obtaining and analyzing containment atmosphere hydrogen after the plant conditions have stabilized (long term application).

4.10 Containment Atmosphere Oxygen

The capability to remove grab samples of containment atmosphere for analysis of oxygen in the containment atmosphere is specified in RG 1.97 revision 3. NUREG-0737 recommends measuring oxygen concentration, but does not require the measurement.

The containment atmosphere oxygen concentration is measured to evaluate the potential for combustion of the containment hydrogen. The only potential source of oxygen in the post accident environment is radiolysis of the sump water. This source of oxygen is not expected to cause a significant increase of oxygen above that initially existing in the containment.

The accident mitigation guidance does not rely on or use a measurement of the containment oxygen concentration.

Based on the above assessment of the need for post accident sampling of containment oxygen, the oxygen sampling function is considered unnecessary and can be eliminated from all B&W-designed plants.

4.11 Containment Atmosphere Radionuclides

The capability to remove grab samples of containment radionuclides is specified in RG 1.97 revision 3, and in NUREG 0737. The containment atmosphere radionuclide measurement is used to estimate the degree of core damage, to assess offsite dose and for post accident recovery.

Sampling the containment atmosphere is not an accurate method for determining the degree of core damage for several reasons.

a. A significant portion of the fission products can be deposited on RCS internal surfaces and not be released to the containment.

b. If the containment depressurizes, additional fission products could be released to the containment.

c. A representative sample cannot be made because there is no true representative sample point, plate-out of aerosols in sample lines may occur, or time delays may result in obtaining samples during non-stable phases of the accident.

For assessing core damage, other plant indicators are available which provide more rapid indications of actual or projected core damage and have the required accuracy. Such indicators are the core exit thermocouples, containment radiation and containment hydrogen. Containment samples can be used for supplementary information for assessing core damage after the plant conditions have stabilized. Dedicated equipment for this function is not necessary. However, contingency plans should be provided for obtaining and analyzing highly radioactive samples.

For assessing offsite releases, site survey capability is available that provides a better offsite dose assessment. Site surveys are applicable to all accidents and can measure at specific release points.

For post accident recovery, containment samples can be used to assess the containment environment. Dedicated equipment for this function is not necessary. The accident has been mitigated, so plans can be devised for obtaining and analyzing samples of containment atmosphere during post accident recovery activities.

Based on the above assessment, the containment atmosphere radionuclides sampling function is considered unnecessary and can be eliminated from all B&W-designed plants. Other plant indicators may be used to provide core damage assessment and to assess offsite dose. For post accident recovery (long term application), contingency plans can be provided for obtaining and analyzing highly radioactive samples of containment atmosphere after the plant conditions have stabilized.

4.12 Containment Sump pH

The capability to remove grab samples of containment sump water for analysis of pH is specified in RG 1.97 revision 3, but is not specified in NUREG 0737. The purpose of measuring the pH is to assure that it is maintained within an acceptable range to limit stress corrosion cracking of stainless components and to maintain iodine retention in the containment sump water.

The post accident sump water pH is maintained in an alkaline range either by passive pH control or by containment spray additives. Should the containment spray not be activated then the pH value can be estimated with a sufficient degree of accuracy from the volumes and chemistries of water going to the sump.

Based on the above assessment of the need for post accident sampling of reactor sump pH, the reactor sump pH sampling function is considered unnecessary and can be eliminated from all operating B&W-designed plants.

4.13 Containment Sump Chlorides

The capability to remove grab samples of containment sump water for analysis of chlorides is specified in RG 1.97 revision 3, but is not specified in NUREG 0737. The purpose of measuring the chlorides is to assure that the high concentrations of chlorides are not in the containment sump. High concentrations are undesirable because they can cause stress corrosion cracking of stainless components and affect iodine retention in the containment sump water.

The source of chlorides is minimal, except for plants with a potential for infiltration of brackish water (i.e., plants with a single barrier between the cooling water and the containment). In addition chloride concentrations of water being added to the containment sump are known such that the resulting concentration of chlorides in the sump water can be estimated with a sufficient degree of accuracy. The containment chloride concentration is not used in any short-term accident mitigation guidance, however, it is used to mitigate long term effect of stress corrosion due to chloride presence.

Based on the above assessment of the need for post accident sampling of reactor sump chlorides, the reactor sump chloride sampling function is considered unnecessary and can be eliminated from all operating B&W-designed plants.

4.14 Containment Sump Boron

The capability to remove grab samples of containment sump water for analysis of radionuclides is specified in RG 1.97 revision 3, but is not specified in NUREG 0737. The purpose of sampling the containment sump for boron content is to assure reactor subcritically should sump water be used in the recirculation mode to cool the core.

For most situations only water from the RCS, Borated Water Storage Tank and Core Flood Tanks will accumulate in the containment sump. The boron concentration maintained in the Borated Water Storage Tank and the Core Flood tanks is sufficient to establish a sump boron concentration which, when mixed with water in the RCS, will assure subcriticality during the recirculation mode. This is true any time during the fuel cycle. If unborated water should be introduced into the containment sump, the resulting sump water boron concentration would be reduced. However, the sump water boron concentration can be readily and adequately estimated by knowing the amounts of water added to the sump and their respective boron concentrations.

Based on the above assessment of the need for post accident sampling of reactor sump boron, the reactor sump boron sampling function is considered unnecessary and can be eliminated from all operating B&W-designed plants.

4.15 Containment Sump Radionuclides

The capability to obtain grab samples of containment sump water for analysis of radionuclides is specified in RG 1.97 revision 3, but is not specified in NUREG 0737. The purpose of sampling the containment sump for radionuclide content is to enable predictions of offsite dose due to leakage from emergency core coolant system recirculation flow path. The sampling is also used to provide an indication of the degree of core damage.

In regard to emergency response, estimating the degree of core damage should use real-time indications. The time necessary to provide a radiological sump sample and to provide an analysis of the radionuclides is too long to use the containment sump radionuclides as a practical method for estimating core damage. For assessing core damage, other plant indicators are available which provide more rapid indications of actual or projected core damage and have the required accuracy. Such indicators are the core exit thermocouples and containment radiation.

In regard to offsite dose predictions, the containment sump radionuclides cannot be accurately determined nor can they be determined in a timely manner. For assessing offsite dose, site survey capability is available and provides a better offsite dose assessment. Site surveys are applicable to all accidents and can measure at specific release points.

Based on the above assessment of the need for post accident sampling of reactor sump radionuclides, the reactor sump radionuclides sampling function is considered unnecessary and can be eliminated from all operating B&W designed plants.

TABLE 4-1
PASS PARAMETER NEED SUMMARY

Pass Parameter	Purpose	Is Measurement Needed?	Could Measurement Be Beneficial?	Justification For Not Needing Measurement
1. RCS dissolved gases	Proximity to RCS void formation.	No	No	RCS dissolved gas measurement is not necessary with ability to measure RCS gas void and to vent void as necessary. Use High point vent, hot leg level instrumentation, RVLIS.
2. RCS Hydrogen	Proximity to RCS void formation. Leading indicator of fuel clad deterioration.	No	No	RCS hydrogen measurement is not necessary with ability to measure RCS gas void and to vent void as necessary. Use High point vent, hot leg level instrumentation, RVLIS.
3. RCS Oxygen	Assess potential for stress corrosion cracking of stainless steel components in the presence of high chloride concentration.	No	Only during long term mitigation	SCC not a concern when maintaining proper pH. Therefore, RCS oxygen measurement not necessary with pH control.
4. RCS pH	Assure radioactive Iodine is retained in the RCS and assess potential for stress corrosion cracking of stainless steel components.	No	Only during long term mitigation	RCS pH measurement not necessary with pH control. RCS pH control is performed either automatically using containment spray, passively using trisodium phosphate baskets, or manually by adding chemicals to the sump.

Pass Parameter	Purpose	Is Measurement Needed?	Could Measurement Be Beneficial?	Justification For Not Needing Measurement
5. RCS Chlorides	Assure that chloride induced stress corrosion cracking of the stainless steel components will not occur.	No	Only during long term mitigation	SCC not a concern when maintaining proper pH. Therefore, chloride measurement not necessary with pH control. RCS chlorides can be manually estimated using chlorides concentration values of external sources of coolant.
6. RCS Boron	Assure adequate shutdown margin is maintained.	No	Only during long term mitigation	Determine shutdown margin by CRDM position indications, reactor negative startup rate indication, and reactor wide range power indication. RCS boration instructions do not rely on RCS boron concentration measurement.
7. RCS Conductivity	Verify RCS pH control measures.	No	No	Conductivity measurement is not necessary for RCS pH control.
8. RCS Radionuclides	Core damage assessment	No	No	Estimate Core damage by Core exit thermocouples, RCS level measurement, high containment radiation level, loss of sub-criticality, loss of sub-cooling margin, and high RCS letdown radiation level.

TOPICAL RECOMMENDATION

Parameter	Purpose	Is Measurement Needed?	Could Measurement Be Beneficial?	Justification For Not Needing Measurement
9. Containment Atm Hydrogen	Proximity to Hydrogen combustion	No	Only during long term mitigation	Use H2 monitor for short term and long term Use non-dedicated sample system for long term if high H2 concentration
10. Containment Atm Oxygen	Proximity to Hydrogen combustion	No	No	Estimate oxygen based on hydrogen concentration and partial pressure of steam
11. Containment Atm Radionuclides	Core damage assessment	No	Only during long term mitigation	Estimate radionuclides based on other parameters such as core outlet temperature and containment radiation. Use non-dedicated sample system with contingency plan
	Offsite dose assessment	No	Only during long term mitigation	Use site survey and specific release point surveys
	Post accident recovery	No	Only during long term mitigation	Use non-dedicated sample system for long term with no contingency plan
12 Sump pH	Stress chloride cracking Iodine retention	No	Only during long term mitigation	SCC and iodine retention not a concern when maintaining proper pH. pH measurement not necessary when pH maintained by passive pH control or containment spray additive (add buffer if spray does not activate) Estimate pH from volumes and chemistries of water going to sump.

Parameter	Purpose	Is Measurement Needed?	Could Measurement Be Beneficial?	Justification For Not Needing Measurement
13 Sump Chlorides	Stress chloride cracking Iodine retention	No	Only during long term mitigation	SCC and iodine retention not a concern when maintaining proper pH. Therefore, chloride measurement not necessary with pH maintained. pH maintained by passive pH control or containment spray additive (add buffer if spray does not activate) Estimate chloride from volumes and chemistries of water going to sump
14 Sump Boron	Subcriticality in recirc	No	Only during long term mitigation	Estimate boron from volumes and chemistries of water going to sump.
15 Sump Radionuclides	Core damage assessment	No	No	Estimate radionuclides based on other parameters such as core outlet temperature and containment radiation.

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5.0 REFERENCES

5.1 NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980

5.2 NUREG-0578, TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations," September 1979

5.3 NRC Regulatory Guide 1.97, Revision 3, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environmental Conditions During and Following an Accident," May 1983

5.4 Title 10 Code of Federal Regulations, Part 50 Appendix A, "General Design Criteria"

5.5 Title 10 Code of Federal Regulations, Part 50 Section 54, "Condition of License"

5.6 Title 10 Code of Federal Regulations, Part 50 Section 47, "Emergency Preparedness"

5.7 Title 10 Code of Federal Regulations, Part 50 Appendix E, "Emergency Planning and Preparedness"

5.8 Letter, Stuart Richards, Director NRC Division of Licensing Project Management to Ralph Phelps, Chairman CE Owners Group, Subject: Acceptance for Reference of the Combustion Engineering Joint Application Report, CE NPSD-1157, Revision 1, "Technical Justification for Elimination of the Post-Accident Sampling System from the Plant Design and Licensing Bases for CEOG Utilities" (TAC No. MA5661), Dated May 16, 2000

5.9 Letter, Stuart Richards, Director NRC Division of Licensing Project Management to Karl Jacobs, Chairman Westinghouse Owners Group, Subject: Safety Evaluation to Topical Report WCAP-14986, Revision 1, Westinghouse Owners Group Post Accident Sampling System Requirements" (TAC No. MA4176), Dated June 14, 2000

5.10 NUREG/CR-4430, "Review of Light Water Reactor Requirements, Volume 3:Assessment of Selected Regulatory Requirements That May Have Marginal Importance to Risk," March 1987

5.11 NUREG-0654, Rev 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, " October 1980

5.12 NUMARC/NESP-07, "Methodology for Development of Emergency Action Levels," April 1992

5.13 B&W Owners' Group Operator Support Committee. Generic Severe Accident Guideline (and Technical Basis Document). BWNT 69-1224353-01. 11 March, 1994.

5.14 Letter 0CAN079901 from Mr. CRH of Entergy to NRC, Subject: Arkansas Nuclear One-Unit 1 and Unit 2, Docket Nos. 50-313 and 50-368, License Nos. DPR-51 and NPF-6, "Proposed Changes for Relief from Technical Specification and NUREG-0737 Requirements," July 14, 1999

5.15 "Severe Accident Management Guidance Technical Bases Report," Volume 1, Fauske & Associates, Inc., Final Report, December 1992, EPRI Research Project 3051-2

5.16 Letter Stephen Dembek, NRC Section Chief of Licensing Project Management to Lou Liberatori, Chairman WOG Steering Committee, Subject: SER Related to Topical Report WCAP-14696-A, Rev. 1, "WOG Core Damage Assessment Guidance", September 2, 1999

5.17 B&W Owners Group Document 74-152419 rev 9, "Emergency Operating Procedure Technical Bases"

APPENDIXES

A. EMERGENCY OPERATING PROCEDURE REVIEW

B. REVIEW OF SEVERE ACCIDENT MANAGEMENT GUIDELINES

C. EMERGENCY CLASSIFICATION PROCEDURE REVIEW

D. CORE DAMAGE ASSESSMENT PROCEDURE REVIEW

APPENDIX A
EMERGENCY OPERATING PROCEDURE
REVIEW
For parameters which can be measured by PASS

6.0 Parameter	**Plant**	**Procedure**	**Step**	**Purpose Of Sample**
RCS Boron	ONS	SG Tube Leak Section 504 (Rev. 29)	42	To determine if boron concentration adequate for cooldown
RCS Boron	ONS	SG Tube Leak Section 504 (Rev. 29)	54	To determine if boron concentration adequate for cooldown
RCS Boron	ONS	SG tube leak Section 504 (Rev. 29)	68	To determine if RCS boron dilution occurring due to SGTR back leakage
RCS Boron	ONS	SG Cooldown With Sat. RCS, CP-602 (Rev. 29)	48	To asses RCS boron status
RCS Boron	ONS	HPI Cooling Cooldown CP-603 (Rev. 29)	29	To asses RCS boron status
RCS Boron	ONS	Solid Plant Cooldown CP-604 (Rev. 29)	43	To determine if boron concentration adequate for cooldown
RCS Boron	ONS	Subcooled cooldown CP-605 (Rev. 29)	1.4	To determine if boron concentration adequate for cooldown
RCS Boron	TMI	OTSG Tube Rupture 1210-5 (Rev. 31)	3.26	To determine if boron concentration adequate for cooldown
RCS Boron	CR-3	VSSV EOP-02 (Re. 5)	3.3	To determine if minimum requirement met for stuck rod(s)
RCS Boron	CR-3	Excessive Heat Transfer EOP-05 (Rev. 4)	3.33	To determine if boron concentration adequate for cooldown

6.0 Parameter	Plant	Procedure	Step	Purpose Of Sample
RCS Boron	CR-3	Natural Circ. Cooldown EOP-09 (Rev. 3)	3.15	To determine if boron concentration adequate for cooldown
RCS Boron	CR-3	Post Trip Stabilization EOP-10 (Rev. 4)	3.5	To assess reactor shutdown if no SR channel available
RCS Boron	CR-3	SGTR EOP-06 (Rev. 10)	3.43	To determine if boron concentration adequate for cooldown
RCS Boron	CR-3	Post Trip Stabilization EOP-10 (Rev. 4)	3.8	To assess RCS boron status

APPENDIX A
EMERGENCY OPERATING PROCEDURE
REVIEW
For parameters which can be measured by PASS

7.0 Parameter	Plant	Procedure	Step	Purpose Of Sample
RCS Boron	CR-3	Post Accident Boron Concentration Management, EM-225B (Rev. 7)	All	To assess RCS boron status
RCS Boron	DB	Supplementary Action DB-OP-02000 (Rev. 5)	4.1.2	To determine if minimum requirement met for stuck rod(s)
RCS Boron	DB	Overcooling DB-OP-02000 (Rev. 5)	7.39	To determine if boron concentration adequate for cooldown
RCS Boron	DB	Saturated SG Cooldown DB-OP-02000 (Rev. 5)	11.1 1.1	To determine if boron concentration adequate for cooldown
RCS Boron	DB	Solid Plant SG Cooldown DB-OP-02000 (Rev. 5)	13.6	To determine if boron concentration adequate for cooldown
RCS Radionuclide	CR-3	SGTR EOP-06 (Rev. 10)	3.6	To assess dose equivalent I^{131} status
RCS Radionuclide	CR-3	Post Trip Stabilization EOP-10 (Rev. 4)	3.8	To assess dose equivalent I^{131} status

7.0 Parameter	Plant	Procedure	Step	Purpose Of Sample
Containment Hydrogen	ONS	SG Cooldown With Sat. RCS CP-602 (Rev. 29)	48	To asses containment hydrogen status
Containment Hydrogen	ONS	HPI Cooling Cooldown CP-603 (Rev. 29)	29	To asses containment hydrogen status
Containment Hydrogen	TMI	SBLOCA 1210-6 (Rev. 22)	2.35	To determine when to start hydrogen recombiners
Containment Hydrogen	TMI	LBLOCA 1210-7 (Rev. 22)	2.26	To determine when to start hydrogen recombiners
Containment Hydrogen	TMI	RCS Superheat 1210-8 (Rev. 19)	2.12	To determine when to start hydrogen recombiners

APPENDIX A
EMERGENCY OPERATING PROCEDURE
REVIEW
For parameters which can be measured by PASS

8.0 Parameter	Plant	Procedure	Step	Purpose Of Sample
Containment Hydrogen	CR-3	ICC EOP-07 (Rev. 7)	3.16	To assess containment hydrogen status
Containment Hydrogen	CR-3	LOCA Cooldown EOP-08	3.64	To assess containment hydrogen status
Containment Hydrogen	CR-3	LOCA Cooldown EOP-08 (Rev. 8)	3.84	To assess containment hydrogen status
Containment Hydrogen	CR-3	LOCA Cooldown EOP-08 (Rev. 8)	3.50	To assess containment hydrogen status
Containment Hydrogen	CR-3	LOCA Cooldown EOP-08 (Rev. 8)	3.103	To assess containment hydrogen status
Containment Hydrogen	CR-3	EOP Enclosures EOP-14 (Rev. 7)	Encl. 21	To assess containment hydrogen status
Containment Hydrogen	CR-3	LOCA cooldown EOP-08 (Rev. 8)	3.114	To assess containment hydrogen status
Containment Hydrogen	DB	LBLOCA DB-OP-02000 (Rev. 5)	10.19	To determine when to start containment hydrogen control
Containment Hydrogen	DB	MU/HPI Cooling DB-OP-02000 (Rev. 5)	12.8	To determine when to start containment hydrogen control
Containment Hydrogen	DB	Subcooled SG Cooldown DB-OP-02000 (Rev. 5)	13.10	To determine when to start containment hydrogen control

8.0 Parameter	Plant	Procedure	Step	Purpose Of Sample
Containment Atmosphere Radionuclides	CR-3	LOCA Cooldown EOP-08 (Rev. 8)	3.34	To determine if I131 concentration is low enough to permit stopping containment spray
Containment Sump Radionuclide	TMI	SBLOCA 1210-6 (Rev. 22)	2.36	To asses containment sump radionuclide status

APPENDIX A
EMERGENCY OPERATING PROCEDURE
REVIEW
For parameters which can be measured by PASS

9.0 Parameter	Plant	Procedure	Step	Purpose Of Sample
Containment Sump Radionuclide	TMI	LBLOCA 1210-7 (Rev. 22)	2.27	To asses containment sump radionuclide status
Containment Sump pH	ONS	SG Cooldown With Sat. RCS CP-602 (Rev. 29)	48	To asses containment sump pH status
Containment Sump pH	ONS	HPI Cooling Cooldown CP-603 (Rev. 29)	29	To asses containment sump pH status
Containment Sump pH	TMI	SBLOCA 1210-6 (Rev. 22)	2.36	To determine when to add sodium hydroxide
Containment Sump pH	TMI	LBLOCA 1210-7 (Rev. 22)	2.27	To determine when to add sodium hydroxide
Containment Sump Boron	TMI	LBLOCA 1210-7 (Rev. 22)	2.27	To asses containment sump boron status
Containment Sump Boron	TMI	SBLOCA 1210-6 (Rev. 22)	2.36	To asses containment sump boron status
Containment Sump Boron	ONS	SG Cooldown With Sat. RCS, CP-602 (Rev. 29)	14	To determine if RCS boron dilution occurring due to addition of non-borated water
Containment Sump Boron	ONS	HPI Cooling Cooldown CP-603 (Rev. 29)	11	To determine if RCS boron dilution occurring due to addition of non-borated water
Containment Sump Boron	ONS	Enclosure 7.11 (Rev. 29)	8	To asses containment sump boron status

9.0 Parameter	Plant	Procedure	Step	Purpose Of Sample
Containment Sump Boron	ONS	CD Following LOCA CP-601 (Rev. 29)	18	To determine if RCS boron dilution occurring due to addition of non-borated water

APPENDIX A
EMERGENCY OPERATING PROCEDURE
REVIEW
For parameters which can be measured by PASS

10.0 Parameter	Plant	Procedure	Step	Purpose Of Sample
Containment Sump Boron	CR-3	LOCA Cooldown EOP-08 (Rev. 8)	3.50	To assess containment sump boron status
Containment Sump Boron	CR-3	LOCA Cooldown EOP-08 (Rev. 8)	3.64	To assess containment sump boron status
Containment Sump Boron	CR-3	LOCA Cooldown EOP-08 (Rev. 8)	3.84	To assess containment sump boron status
Containment Sump Boron	CR-3	LOCA Cooldown EOP-08 (Rev. 8)	3.103	To assess containment sump boron status
Containment Sump Boron	CR-3	LOCA Cooldown EOP-08 (Rev. 8)	3.114	To assess containment sump boron status
Containment Sump Boron	CR-3	EOP Enclosures EOP-14 (Rev. 7)	Encl. 19	To assess containment sump boron status
Containment Sump Boron	DB	LBLOCA DB-OP-02000 (Rev. 5)	10.19	To assess containment sump boron status
Containment Sump Boron	DB	MU/HPI Cooling DB-OP-02000 (Rev. 5)	12.8	To assess containment sump boron status
Containment Sump Boron	DB	Subcooled SG Cooldown DB-OP-02000 (Rev. 5)	13.10	To assess containment sump boron status

APPENDIX B

REVIEW OF SEVERE ACCIDENT MANAGEMENT GUIDELINES

For parameters which can be measured by PASS

Parameter	**Plant**	**Procedure**	**Section / Step**	**Purpose of Sample**	**PASS Sample Required**
Containment Hydrogen	CR3	Severe Accident Guideline, Rev. 1	Chapter III – Diagnosis and Mitigation – Plant Damage Condition: Oxidized/Badly Damaged (OX/BD) Section 1.0 Symptoms	Measurement of increasing H_2 in containment is used as a symptom of prolonged core uncovery for core highly oxidized / badly damaged core condition.	NR (a) (m)
Containment Hydrogen	CR3	Severe Accident Guideline, Rev. 1	Chapter III – Diagnosis and Mitigation – Plant Damage Condition: Ex-Vessel (EX) Section 1.0 Symptoms	Measurement of increasing H_2 in containment is used as a symptom for molten core material outside of the reactor vessel.	NR (b) (m)
Containment hydrogen	CR3	Severe Accident Guideline, Rev. 1	Chapter III – Diagnosis and Mitigation – Plant Damage Condition: Challenged (CH) Section 1.0 Symptoms	Measurement of significant and increasing containment H_2 (e.g., $\geq 3\%$) is used as a symptom that the containment integrity is challenged.	NR (c) (m)
Containment Hydrogen	TMI1	TMI-1 Severe Accident Guideline, Rev. 1	Section 2.1.1 OX/BD Symptoms	Measurement of increasing H_2 in containment is used as a symptom of prolonged core uncovery for core highly oxidized / badly damaged core condition.	NR (d) (m)
Containment Hydrogen	TMI1	TMI-1 Severe Accident Guideline, Rev. 1	Section 2.2.1 EX Symptoms	Measurement of increasing H_2 in containment is used as a symptom for molten core material outside of the reactor vessel.	NR (e) (m)

Parameter	Plant	Procedure	Section / Step	Purpose of Sample	PASS Sample Required
Containment Hydrogen	TMI1	TMI-1 Severe Accident Guideline, Rev. 1	Section 2.3.1 CH Symptoms	Measurement of increasing H_2 in containment is used as a symptom that the containment integrity is challenged.	NR (f) (m)
Containment Hydrogen	DB	DBSAMG, Rev. 0	Chapter III – Diagnosis and Mitigation A. Plant Damage Condition Oxidized and Badly Damaged (OX/BD) 1.0 Symptoms	Measurement of increasing H_2 in containment is used as a symptom of prolonged core uncovery for core highly oxidized / badly damaged core condition.	NR (g) (m)

APPENDIX B
Review of Plant Severe Accident Management Guidelines
For parameters which can be measured by PASS

Parameter	Plant	Procedure	Section / Step	Purpose of Sample	PASS Sample Required
Containment Hydrogen	DB	DBSAMG, Rev. 0	Chapter III – Diagnosis and Mitigation B. Plant Damage Condition Ex-Vessel (EX) 1.0 Symptoms	Measurement of increasing H_2 in containment is used as a symptom for molten core material outside of the reactor vessel.	NR (h) (m)
Containment Hydrogen	DB	DBSAMG, Rev. 0	Chapter III – Diagnosis and Mitigation D. Plant Damage Condition Containment Challenged (CH) 1.0 Symptoms	Measurement of significant (e.g., $\geq 3\%$) and increasing CTMT H_2 is used as a symptom that the containment integrity is challenged.	NR (i) (m)
Containment Hydrogen	ONS	Oconee Severe Accident Guidelines, Rev. 0	Chapter III – Diagnosis and Mitigation PDC OX/BD Mitigation Section 1.0 Symptoms	Measurement of increasing H_2 in containment is used as a symptom of prolonged core uncovery for core highly oxidized / badly damaged core condition.	NR (j) (m)
Containment Hydrogen	ONS	Oconee Severe Accident Guidelines, Rev. 0	Chapter III – Diagnosis and Mitigation PDC EX Mitigation Section 1.0 Symptoms	Measurement of increasing H_2 in containment is used as a symptom for molten core material outside of the reactor vessel.	NR (k) (m)
Containment Hydrogen	ONS	Oconee Severe Accident Guidelines, Rev. 0	Chapter III – Diagnosis and Mitigation PDC CH Mitigation Section 1.0 Symptoms	Measurement of significant (e.g., $\geq 3\%$) and increasing containment H_2 is used as a symptom that the containment integrity is challenged.	NR (l) (m)

Notes:

R - Determining a PASS parameter is required to maintain the effectiveness of the procedure. That is, the procedure calls for different actions based on different values and the PASS parameter is the only parameter that can be used for determining the correct action.

NR - Determining the value of a PASS parameter is addressed, but is not required to maintain the effectiveness of the procedure.

(a) Determining prolonged core uncovery is derived by several indications including:
Hot leg and Rx vessel level indication off-scale low when all RCPs are off (RC-163A/163B-LR1, RC-164A/164B-LR1).
Source range indication (NI-1, NI-2, NI-14, NI-15) trending up.
Core injection flow $<W_{vap}$ (saturated vapor flow).
Increasing H_2 in containment (WS-10-CR,WS-11-CR).
T_{hot} RTD increasing (RC-4A-TI4-1, RC-4B-TIR1).
Incore thermocouple temperatures increasing (RC-171/172-TR).
RCS void trend approaching 100% when RCPs are on (RC-169-XR).

(b) Determining that molten core material is outside the reactor vessel is derived by several indications including:
Primary Symptom - RCS pressure decrease and containment pressure / temperature increase (RC-158-PI2, RC-159-PI2, BS-16-PI, BS-17-PI, BS-90-PI, BS-91-PI).
Other confirming symptoms include:
Containment radiation level increasing (RM-G29, RM-G30).
Containment H_2 concentration increasing (WS-10-CR, WS-11-CR).
CO_2 and possible CO concentrations in the containment (no direct measurement available).

(c) Determining challenge to containment integrity is derived by several indications including:
Primary Symptom - High and increasing containment pressure and temperature and containment isolation was successful (BS-16-PI, BS-17-PI, BS-90-PI, BS-91-PI; AH-536/537/538/539-TIR).
Other confirming symptoms include:
Containment H_2 significant (e.g., $\geq 3\%$) and increasing (WS-10-CR, WS-11-CR).
No indication of water accumulation in the containment (WD-301-LI, WD-302-LI).
RCS pressure high ($\geq$ 500 to 600 psig) challenging OTSG tubes, and therefore containment (RC-158-PI2, RC-159-PI-2).

(d) Determining prolonged core uncovery is derived by several indications including:
RCITS off-scale low.
Source range indication trending up.
Core injection flow <W_{vap}.
Increasing H_2 in containment.
T_{hot} RTD increasing.
Incore thermocouple temperatures increasing.

(e) Determining that molten core material is outside the reactor vessel is derived by several indications including:
<u>Primary Symptom</u> – Nearly simultaneous RCS pressure decrease and containment pressure/temperature increase.
Other confirming symptoms include:
Containment radiation level increase.
Containment H_2 concentration increase.
CO_2 and possible CO concentrations in the containment.

(f) Determining challenge to containment integrity is derived by several indications including:
<u>Primary Symptom</u> – Reactor isolation complete in combination with high and increasing containment pressure and temperature.
Other confirming symptoms include:
Containment H_2 significant (e.g., $\geq$ 3%) and increasing.
No indication of water accumulation in the containment with an indication of no RCS injection (e.g., failure of containment spray).
RCS pressure high ($\geq$ 500 to 600 psig).

(g) Determining prolonged core uncovery is derived by several indications including:
Hot Leg Monitoring System (HLLMS) off-scale low.
Source range indication trending up.
Core injection flow <W_{vap} (saturated liquid flow).
Increasing H_2 in CTMT.
T_{hot} RTD increasing.
Incore thermocouple temperatures increasing.

(h) Determining that molten core material is outside the reactor vessel is derived by several indications including:

Primary Indication – RCS pressure decrease and CTMT pressure/temperature increase.

Other confirming symptoms include:

CTMT radiation level increase.
CTMT H_2 concentration increase.

(i) Determining challenge to containment integrity is derived by several indications including:

Primary Indication – CTMT isolation complete in combination with high and increasing CTMT pressure and temperature.

Other confirming symptoms include:

CTMT H_2 significant (e.g., $\geq$ 3%) and increasing.
No indication of water accumulation in CTMT.
RCS pressure high ($\geq$ 500 to 600 psig).

(j) Symptoms of prolonged core uncovery include:

RVLIS off-scale low.
Source range indication trending up.
Core injection flow <W_{vap}. Refer to W_{vap} calculational aid to estimate flowrate.
Increasing H_2 in containment. Refer to RP/0/B/1000/18, CORE DAMAGE ASSESSMENT, for estimated H_2 concentrations.
T_{hot} RTD increasing.
CETC temperatures increasing.
Erratic incore neutron detector response (i.e. large spike or off-scale high/low readings).

(k) Determining that molten core material is outside the reactor vessel is derived by several indications including:

Primary Indication – RCS pressure decrease and containment pressure/temperature increase.

Other confirming symptoms include:

Containment radiation level increase. (RIA-57 & 58)
Containment H_2 concentration increase. (H_2 concentrations of 0-10% are available from the H_2 analyzer. Higher concentrations can be measured by Chemistry through analysis of a PAGS sample. Radiation dose should be considered prior to requesting this manual analysis.

CO_2 and possible CO concentrations in the containment. (ONS has no preplanned method for determination of CO and CO_2 concentrations.)

(l) Determining challenge to containment integrity is derived by several indications including:

Primary Indication – Reactor isolation complete (Refer to EOP Section 505, ES Checklist) in combination with high and increasing containment pressure and temperature.

Other confirming symptoms include:

Containment H_2 significant ($\geq$ 3%) and increasing. (H_2 concentrations of 0-10% are available from the H_2 analyzer. Higher concentrations can be measured by Chemistry through analysis of a PAGS sample. Radiation dose should be considered prior to requesting this manual analysis.)
No indication of water accumulation in the containment.
RCS pressure high ($\geq$ 500 to 600 psig).

(m) Hydrogen level in the containment atmosphere is normally measured with hydrogen monitors for concentrations from 0 to 10 %. Determination of hydrogen levels above 10 % is by gas sample. Detection of hydrogen level above 10 % has no impact on the choice of CHLA to be executed.

APPENDIX C
EMERGENCY CLASSIFICATION
PROCEDURE REVIEW
For parameters which can be measured by PASS

Parameter	Plant	Procedure	Step	Purpose of Sample	PASS Sample Required
Containment Hydrogen	CR-3	Duties of Emergency Coordinator EM-202, Rev. 66	Step 4.0.2 Enclosure 1	Determine if > 4 % hydrogen for unusual event classification for containment potential loss for Fission Product Barrier. If other EALs are reached the classification can be higher.	(a)(b)
RC Radionuclide	CR-3	Duties of Emergency Coordinator EM-202, Rev. 66	Step 4.0.2 Enclosure 1	Determine if > 300 μCi/gram DEI for alert classification fuel clad loss for Fission Product Barrier. If other EALs are reached the classification can be higher.	R (a)
RC Radionuclide	CR-3	Duties of Emergency Coordinator EM-202, Rev. 66	Step 4.0.2 Enclosure 1	Determine if > 1 μCi/gram DEI (> 48 hours) or > 100/E bar μCi/gram (> 48 hours) for unusual event classification for fuel clad degradation for system malfunction (modes 1,2,3,4,5)	(a)(c)
RCS Radionuclides	DB	Emergency Classification RA-EP-01500, Rev. 1	Tab 1.B.2	Determine if > 300 μCi/gram DEI for alert classification for very high coolant activity (all modes)	R
RCS Radionuclides	DB	Emergency Classification RA-EP-01500, Rev. 1	Tab 1.B.1	Determine if > 1 μCi/gram DEI (>48 hrs) or > 100/E bar μCi/gram for unusual event classification for high RC activity requiring plant shutdown per T.S. (mode 1,2)	(c) (e)
RCS Radionuclides	DB	Emergency Classification RA-EP-01500, Rev. 1	Tab 1.B.4	Determine if > 300 μCi/gram DEI and other EAL for GE classification for core damage with other plant conditions making a release of large amounts or radioactivity possible (all modes)	R

Parameter	Plant	Procedure	Step	Purpose of Sample	PASS Sample Required
RCS Radionuclides	DB	Emergency Classification RA-EP-01500, Rev. 1	Tab 1.B.3	Determine if > 1 μCi/gram DEI or > 100/E bar μCi/gram (> 48 hrs) and other EAL (CET superheat) for SAE classification core damage with inadequate core cooling (all modes)	(c) (e)
RCS Radionuclides	DB	Emergency Classification RA-EP-01500, Rev. 1	Tab 1.C.1	Determine if > 300 μCi/gram DEI and other EAL for GE classification for loss of 2 of 3 fission product barriers with a potential loss of the third (all modes)	R

APPENDIX C
EMERGENCY CLASSIFICATION
PROCEDURE REVIEW
For parameters which can be measured by PASS

Parameter	Plant	Procedure	Step	Purpose of Sample	PASS Sample Required
Containment Hydrogen	ONS	Emergency Classification RP/0/B/1000/001, Rev. 8	Enclosure 4.1	Determine if > 9 % hydrogen for unusual event classification for containment potential loss for Fission Product Barrier. If other EALs are reached the classification can be higher. (modes 1,2,3,4)	(b)
RC Radionuclide	ONS	Emergency Classification RP/0/B/1000/001, Rev. 8	Enclosure 4.1	Determine if > 300 μCi/gram DEI for alert classification fuel clad loss for Fission Product Barrier. If other EALs are reached the classification can be higher. (modes 1,2,3,4)	R
RC Radionuclide	ONS	Emergency Classification RP/0/B/1000/001, Rev. 8	Enclosure 4.2	Determine if > 5 μCi/ml DEI for unusual event classification for fuel clad degradation for system malfunction (all modes)	(c)
Containment Hydrogen	TMI	Emergency Classification and Basis EPIP-TMI-01, Rev. 7	Exhibit 2	Determine if > 4 % hydrogen for unusual event classification for containment potential loss for Fission Product Barrier. If other EALs are reached the classification can be higher.	(b)
RC Radionuclide	TMI	Emergency Classification and Basis EPIP-TMI-01, Rev. 7	Exhibit 2	Determine if > 2500 μCi/cc for alert classification fuel clad loss for Fission Product Barrier. If other EALs are reached the classification can be higher.	R (d)
RC Radionuclide	TMI	Emergency Classification and Basis EPIP-TMI-01, Rev. 7	Exhibit 1	Determine if > .35 μCi/gram DEI (>48 hours) or > 100/E bar μCi/gram or > 60 μCi/grm DEI for unusual event classification for fuel clad degradation for system malfunction (power operation)	(c) (f)

Parameter	Plant	Procedure	Step	Purpose of Sample	PASS Sample Required
RC Radionuclide	TMI	Emergency Classification and Basis EPIP-TMI-01, Rev. 7	Exhibit 1	Determine if > .35 μCi/gram DEI (>48 hours) or > 100/E bar μCi/gram or > 275 μCi/grm DEI for unusual event classification for fuel clad degradation for system malfunction (hot shutdown)	(c) (f)

Notes:

R – Determining the value of a PASS parameter is required to maintain the effectiveness of the procedure. That is, the procedure calls for different actions based on different values and the PASS parameter is the only parameter that can be used for determining the action.

NR - Determining the value of a PASS parameter is addressed but is not required to maintain the effectiveness of the procedure.

(a) The same sample requirement is restated in Radiological Emergency Response Plan Table 8.1

(b) Method of measuring hydrogen concentration not stated. However, EAL limit within range of hydrogen monitors so assume hydrogen monitor will be used.

(c) Method of measuring radionuclide not stated. However, EAL limit within range of normal sample system so assume normal sample system will be used.

(d) 2500 μ/Ci total RCS activity corresponds to approximately 300 μCi/cc DEI ^{131}. This is approximately 5% fuel clad damage.

(e) DB Tech spec 3.4.8

(f) TMI Tech spec 3.1.4

APPENDIX D
CORE DAMAGE ASSESSMENT
PROCEDURE REVIEW
For parameters that can be measured by PASS

Parameter	Plant	Procedure	Step	Purpose of Sample	Sample Required
RCS Radionuclides	CR-3	Post Accident Sampling and Analysis of the RCS CH-632A, Rev. 4	Enclosure 2	Estimate amount of core damage.	NR (d)
Containment Atmosphere Radionuclides	CR-3	Post Accident Sampling and Analysis of Reactor Building Atmosphere CH-631, Rev. 2	Enclosure 2	Estimate amount of core damage	NR (d)
Containment Sump Radionuclides	CR-3	Post Accident Sampling and Analysis of Reactor Building Sump CH-632D, Rev. 4	Enclosure 2	Estimate amount of core damage	NR (d)
Containment Hydrogen	DB	Emergency Technical Assessment RA-EP-02320, Rev. 1	6.5	Assess amount of core damage	NR (a) (d)
RCS Radionuclides	DB	Emergency Technical Assessment RA-EP-02320, Rev. 1	6.6	Assess amount of core damage	NR (d)
Containment Atmosphere Radionuclides	DB	Emergency Technical Assessment RA-EP-02320, Rev. 1	6.6	Assess amount of core damage	NR (d)
Containment Sump Radionuclides	DB	Emergency Technical Assessment RA-EP-02320, Rev. 1	6.6	Assess amount of core damage	NR (d)
Containment Hydrogen	ONS	Core Damage Assessment RP/0/B/1000/18, Rev. 2	1.1	Determine core damage symptom. Little or no hydrogen concentration (condition zero) indicates only mechanical core damage condition (fuel non-overheating)	NR (b) (c)

APPENDIX D
CORE DAMAGE ASSESSMENT
PROCEDURE REVIEW
For parameters that can be measured by PASS

Parameter	Plant	Procedure	Step	Purpose of Sample	Sample Required
Containment Hydrogen	ONS	Core Damage Assessment RP/0/B/1000/18, Rev. 2	1.2	Determine core damage symptom. Increasing hydrogen concentration 0 to .5% (condition one) indicates cladding damage condition (fuel overheating)	NR (b) (c)
Containment Hydrogen	ONS	Core Damage Assessment RP/0/B/1000/18, Rev. 2	1.3	Determine core damage symptom. High hydrogen concentration .2 to 2.9% (condition two) indicates core damage for fuel over-temperature (fuel overheating)	NR (b) (c)
Containment Hydrogen	ONS	Core Damage Assessment RP/0/B/1000/18, Rev. 2	1.4	Determine core damage symptom. High hydrogen concentration 1.4% and greater (condition three) indicates core damage condition for fuel melt (fuel overheating)	NR (b) (c)
RCS Radionuclides	ONS	Core Damage Assessment RP/0/B/1000/18, Rev. 2	1.4	Determine core damage symptom. Measure barium and/or lanthanum >/+ 1 μCi/ml (condition three) indicates core damage condition for fuel melt (fuel overheating)	NR (c)
RCS Radionuclides	ONS	Core Damage Assessment RP/0/B/1000/18, Rev. 2	Enclosure 4.1	Estimate amount of fuel damage for condition zero based on I-131 concentration and I-131/I-133 ratio using normal sample	R (e)
RCS Radionuclides	ONS	Core Damage Assessment RP/0/B/1000/18, Rev. 2	Enclosure 4.1	Estimate amount of fuel damage for condition zero if PASS sample has been taken	NR
RCS Radionuclides	ONS	Core Damage Assessment RP/0/B/1000/18, Rev. 2	Enclosure 4.2	Estimate amount of fuel damage for condition one, two or three	NR (d)
Containment Atmosphere Radionuclides	ONS	Core Damage Assessment RP/0/B/1000/18, Rev. 2	Enclosure 4.2	Estimate amount of fuel damage for condition one, two or three	NR (d)

Parameter	Plant	Procedure	Step	Purpose of Sample	Sample Required
Containment Hydrogen	ONS	Core Damage Assessment RP/0/B/1000/18, Rev. 2	Enclosure 4.2	Estimate amount of fuel damage for condition one, two or three	NR (d)
containment hydrogen	TMI	Method for Estimating Extent of Core Damage Under Severe Accident Conditions TDR No. 431, Rev. 2	Section II	Assess amount of core damage	NR (b) (d)
RCS Radionuclides	TMI	Method for Estimating Extent of Core Damage Under Severe Accident Conditions TDR No. 431, Rev. 2	Section III	Assess amount of core damage	NR (d)

Notes:

R – Determining the value of a PASS measurable parameter is required to maintain the effectiveness of the procedure. That is, the procedure calls for different actions based on different values and the PASS measurable parameter is the only parameter that can be used.

NR - Determining the value of a PASS measurable parameter is addressed but is not required to maintain the effectiveness of the procedure.

(a) Use Hydrogen monitors

(b) Method of measuring hydrogen concentration not stated. However, EAL limit within range of hydrogen monitors so assume hydrogen monitor will be used

(c) Core damage symptom is derived from several alternate symptoms including:

- Core exit thermocouple temperature
- Reactor vessel level
- Containment radiation level
- Letdown storage tank radiation
- Containment hydrogen concentration
- Containment sump level
- RCS radionuclide analysis

(d) One of several methods for estimating amount of Fuel Failure which can include:

- Radiochemical analysis or RCS
- Containment radiation detectors
- Containment hydrogen concentration
- Core exit thermocouples

(e) Normal sample system used