CENPSD-1157 (REV.1)

CEOG COMBUSTION ENGINEERING OWNERS GROUP

TECHNICAL JUSTIFICATION FOR THE ELIMINATION OF THE POST-ACCIDENT SAMPLING SYSTEM FROM THE PLANT DESIGN AND LICENSING BASES FOR CEOG UTILITIES

[CEOG TASK 1072]

MARCH 2000

PREPARED FOR THE
COMBUSTION ENGINEERING OWNERS GROUP

BY

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EXECUTIVE SUMMARY

This document provides the technical basis for the elimination of regulatory requirements for the Post Accident Sampling System (PASS) as promulgated by NUREG-0737, "Clarification of TMI Action Plan Requirements." In particular, this document supports the position that elimination of PASS sampling of RCS fluids and containment sump and atmosphere will not adversely impact accident response and mitigation, or emergency preparedness procedures and actions.

In Combustion Engineering Owners Group (CEOG) plants, the PASS performs various sampling and analysis functions that were put into place after the Three Mile Island, Unit 2 (TMI-2) accident. The TMI-2 accident was the catalyst for several regulatory guidance documents that culminated in the issuance of NUREG-0737. However, increased knowledge of accident phenomenology and the considerable amount of operating experience that have been gained in the years since NUREG-0737 was issued have led to a better understanding of degraded core behavior and the role that a PASS would play in various accident scenarios. This better understanding supports the conclusion that PASS does not play a significant role in controlling the plant emergency management response to severe accidents. In certain instances, use of PASS can even degrade the plant emergency response by diverting limited resources to non-essential activities and/or creating a radiation release pathway into the auxiliary building. It has also been determined that the role of the PASS in emergency planning is minimal and primarily confirmatory.

The costs associated with maintaining many of the PASSs are accelerating due to obsolescence and year 2000 (Y2K) issues. Maintenance costs alone vary between $50K to over $100K per plant per year. In addition, obsolescence issues and Y2K concerns can result in per plant capital costs in excess of $500K. The integrated cost of maintaining the PASS over the remaining lifetime of the CEOG plants will be in excess of 20 million dollars. Thus, the PASS is considered to be a costly, resource intensive system and as discussed in the report, the system is no longer considered risk-beneficial and has no significant beneficial impact on plant emergency management response.

The value of the PASS in post accident plant response has been questioned for more than a decade. In the recent past, the value of the PASS has been further diminished by the development and implementation of the CEOG Severe Accident Management Guidance (SAMG) which relies on "in-plant" instrumentation to trend severe accident progression, track recovery actions and establish core damage assessments. In light of current knowledge of severe accident phenomenology and progression, and accident mitigation strategies, PASS requirements were reviewed to determine their contribution to overall plant safety and accident recovery. The review considered the current role of the PASS in all relevant accident management programs. Specifically, the review assessed the accident progression with respect to the plant Abnormal and Emergency Operating Procedures (EOPs), the Severe Accident Management Guidance and the Emergency Plan.

Based on the current understanding of core damage accidents and the current guidance for bringing the plant to a safe, stable state following such an accident, it is recommended that all of the current PASS sample requirements be eliminated. In the event that containment hydrogen trending is needed during a severe accident, containment hydrogen concentration will remain available via the post accident hydrogen analyzers and will not require use of the PASS to determine hydrogen concentrations. These analyzers have been previously approved by the NRC for this purpose (See the Safety Evaluation Report in CE-NPSD-415-A).

In addition, it is recommended that non-PASS monitoring capability be modified in two specific areas to replace PASS monitoring. In the monitoring of boron concentration as specified in the EOPs, it is recommended that sampling of reactor coolant be downgraded from a requirement (in selected procedures) to a recommended action with the expectation that the action can be performed via the Normal Sampling System (NSS). Furthermore, the capability of the NSS or other comparable system to monitor low levels of fuel damage (1-5% failed fuel) should be confirmed by each CEOG plant.

In conclusion, the regulatory requirements for the PASS can be eliminated for CEOG plants without degrading the plant emergency plan. The changes recommended in this report will result in an accident management strategy which meets the initial intent of the Post-TMI guidance by relying on the recently implemented SAMGs to manage severe accident risk and site survey monitoring to support modification of emergency plan protective action recommendations. Thus, the potential risk to public health and safety due to a beyond design basis event is minimized with all essential sampling functions accomplished via qualified non-PASS equipment.

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LIST OF ACRONYMS

AMGs	-	Accident Management Guidelines
AOPs	-	Abnormal Operating Procedures
ARMs	-	Area Radiation Monitors
BOC	-	Beginning of Cycle
CDA	-	Core Damage Assessment
CDAM	-	CDA Methodology
CEA	-	Control Element Assembly
CETs	-	Core Exit Thermocouples
CHLAs	-	Candidate High Level Actions
CVCS	-	Chemical Volume & Control System
EALs	-	Emergency Action Levels
EOPs	-	Emergency Operating Procedures
EPGs	-	Emergency Procedure Guidelines
ERDS	-	Emergency Response Data System
FF	-	Failed Fuel
FRG	-	Functional Recovery Guidelines
GE	-	General Emergency
ICC	-	Inadequate Core Cooling
ISLOCAs	-	Interfacing System LOCAs
LOCA	-	Loss of Coolant Accident
LPZ	-	Low Population Zone
NC	-	Natural Circulation
NOUE	-	Notification of an Unusual Event
NSS	-	Normal Sampling System
ORGs	-	Optimal Recovery Guidelines
PARs	-	Protective Action Recommendations
PASS	-	Post Accident Sampling System
PSA	-	Probabilistic Safety Assessment
RAS	-	Recirculation Actuation Signal
RCP	-	Reactor Coolant Pump
RCS	-	Reactor Coolant System
RG	-	Regulatory Guide
RVLMS	-	Reactor Vessel Level Monitoring System
RWST	-	Refueling Water Storage Tank
SAE	-	Site Area Emergency
SAMG	-	Severe Accident Management Guidance
SAMGs	-	Severe Accident Management Guidelines
SBO	-	Station Blackout
SCC	-	Stress Corrosion Cracking

LIST OF ACRONYMS CONTINUED

SDM	-	Shutdown Margin
SGTRs	-	Steam Generator Tube Ruptures
SIS	-	Safety Injection System
TMI-2	-	Three Mile Island Unit 2
TSC	-	Technical Support Center
TSP	-	Trisodium Phosphate Dodecahydrate
UHJTCs	-	Unheated Junction Thermocouples
WOG	-	Westinghouse Owner's Group

1.0 OBJECTIVE

This document provides the technical basis for the elimination of regulatory requirements for the Post Accident Sampling System (PASS) as promulgated by NUREG-0737, "Clarification of TMI Action Plan Requirements," (Reference 1). In particular, this document supports the position that elimination of PASS sampling of Reactor Coolant System (RCS) fluids and containment sump and atmosphere will not adversely impact accident response and mitigation, or emergency preparedness procedures and actions.

2.0 OVERVIEW

The PASS installed at Combustion Engineering Owners Group Pressurized Water Reactors (PWRs) are designed to perform a multitude of sampling and analysis functions. These functions were designed for, and intended to be used in, post accident situations and were put into place after the Three Mile Island Unit 2 (TMI-2) accident. TMI-2 was the catalyst for several regulatory guidance documents that culminated in the issuance of NUREG-0737. However, in the years since NUREG-0737 was issued, a considerable amount of knowledge and operating experience has been gathered about core behavior and the role that a PASS would play in various accident scenarios. Increased understanding of severe accident phenomenology, coupled with a clearer vision of the severe accident management process, confirms that the PASS does not play a significant role in controlling the plant emergency management response to severe accidents. In certain instances, use of the PASS can even degrade the plant emergency response by diverting limited resources to non-essential activities and/or create a radiation release pathway into the auxiliary building. The role of the PASS in emergency response is minimal and primarily confirmatory.

The costs associated with maintaining many of the PASSs are accelerating due to obsolescence and year 2000 (Y2K) issues. Today, at many CEOG plants the PASS is the plant system requiring the most resources to maintain and operate. The cost of maintaining the PASS varies from $50K to over $100K per plant per year. Obsolescence issues and Y2K concerns may result in per plant capital costs at CEOG utilities in excess of $500K for the PASS. The anticipated costs are impacted by the degree of automation associated with the PASS system. The distribution of CEOG current and planned expenditures (including Y2K "fixes") on the PASS units are summarized in Table 2-1. Table 2-2 summarizes the PASS capabilities for CEOG member utilities. It is estimated that over the remaining lifetime of the CEOG member units, elimination of regulatory requirements associated with PASS will result in savings well in excess of twenty million dollars. In addition, elimination of PASS related calibrations, surveillances, and maintenance will reduce unnecessary radiation exposure to plant personnel (Reference 31) and remove a potential radionuclide release pathway following a beyond design basis event. The PASS is viewed by the CEOG and much of the nuclear utility industry as costly, resource intensive and, as will be discussed in further detail, non-risk beneficial.

The remainder of this report is arranged as follows:

Section 3: Explores various features of the PASS regulatory basis, including the initial basis for the PASS and the subsequent evolution and re-interpretation of associated NUREG-0737 requirements.

Section 4: Provides an overview discussion of the inter-relationships of the Emergency Operating Procedures (EOPs), Abnormal Operating Procedures (AOPs), Severe Accident Management Guidelines (SAMGs) and Emergency Planning.

Section 5: Presents the current use of the PASS by CEOG members and provides the detailed technical basis for eliminating PASS requirements. The existing CEOG accident management and emergency preparedness programs are discussed with respect to PASS utilization.

Section 6: Focuses on a sample-by-sample evaluation of the PASS requirements. Recommendations to delete specific PASS requirements are made and where necessary, non-PASS alternatives to meet the intent of 10CFR50.54 are discussed.

Section 7: Summarizes the CEOG PASS elimination proposal, along with plant specific means to confirm that emergency preparedness is not adversely affected (i.e. a negative 10CFR50.54(q) determination).

Table 2-1:

PASS Projected Per Unit Yearly O&M Costs and Anticipated Short Term Capital Costs

PLANT	O&M Costs K$/year/unit	Anticipated Short Term Capital Costs for Upgrade** (K$ per unit)
Fort Calhoun Station	125	100
Palisades	75	250
Millstone Point 2	not available	not available
St. Lucie 1	65	10
St. Lucie 2	65	10
ANO-2	50	580****
Waterford 3	not available	not available
San Onofre 2 & 3	120 ***	150 ***
Palo Verde 1, 2 & 3	83 *	>$50 per unit
Average Yearly Maintenance per plant for CEOG Utility	87	

* $250K/yr for 3 units

** Costs reflect Y2K upgrades and replacement of aging equipment

*** PASS is common to both Units 2 and 3

**** $830K savings will be realized when considering changes to both ANO-1 and ANO-2

Table 2-2:
Summary of PASS Capabilities for CEOG Member Utilities*

	Reactor Coolant				Containment	
Plant	**Boron Conc.**	**Radionuclide Isotopics**	**RCS Dissolved Gas/ Hydrogen**	**Chlorides**	**Radionuclide Isotopics**	**Hydrogen**
Fort Calhoun Station	In-Line	In-Line	In-Line Total Gas	In-Line	Grab	Grab
Palisades	Grab	Grab	In-Line H_2	Grab	Grab	In-Line H_2 Analyzer ***
Millstone Point Unit 2	Grab	Grab	In-Line Total Gas	Grab	Grab	Grab
St. Lucie 1	Grab	Grab	In-Line H_2	Grab	Grab	In-Line H_2 Analyzer ***
St. Lucie 2	In-Line	Grab	In-Line Total Gas	Grab	Grab	In-Line H_2 Analyzer ***
ANO-2	Grab	In-Line	In-Line H_2	Grab	In-Line	In-Line H_2 Analyzer ***
Waterford 3	In-Line	Grab	In-Line H_2	Grab	Grab	In-Line H_2 Analyzer ***
San Onofre 2 & 3	Grab **	In-Line	In-Line H_2	Grab	In-Line	In-Line H_2 Analyzer ***
Palo Verde 1, 2 & 3	In-Line	Grab	Grab	Grab	Grab	In-Line H_2 Analyzer ***

* As Required by NUREG-0737, In-Line Systems are backed up by grab samples

** The primary sample capability for boron is In-Line boronometer. However, due to drift and accuracy problems, grab sample is normally used.

*** Plants use RG 1.97 approved containment H_2 monitors to meet PASS requirements. Use of In-Line containment H_2 monitors to replace PASS sample was accepted by the NRC in Reference 8.

3.0 EVOLUTION OF THE PASS REGULATORY BASIS

3.1 Post TMI Requirements

The Post Accident Sampling System (PASS) is designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place after the Three Mile Island Unit 2 (TMI-2) accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids with potentially high levels of radioactivity, without exceeding the radiation exposure limits for plant personnel.

The accident at TMI-2 indicated several deficiencies in the ability of the power reactors licensed in the United States to respond to serious accidents. Among these were the need for:

- Additional post accident instrumentation,
- Increased understanding of existing instruments when used beyond the design basis,
- Better emergency response procedures and a post accident sampling system.

In combination, these changes would better enable the plant operating staff to cope with a beyond design basis accident. The NRC's first attempt at defining the lessons learned from the TMI experience was published in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations," September 1979 (Reference 2). Following the initial recommendations NRC issued clarifications. These documents specifically stated that the purpose of implementing post-accident sampling capability was to "...improve efforts to assess the course of an accident...". In 1980, the TMI-2 action plan was published in NUREG-0660, "NRC Action Plan Development as a Result of the TMI-2 Accident," (Reference 3). PASS capability was included as part of this action plan.

The final requirements for post-accident sampling were published in NUREG-0737, "Clarification of TMI Action Plan Requirements," (Reference 1). NUREG-0737 is the clarification of TMI-related actions in the TMI-2 Action Plan, NUREG-0660. Section II.B.3 of NUREG-0737 outlined the requirements for post-accident sampling capability. Eleven specific criteria were defined for the development of a PASS. These criteria are summarized in Table 3-1.

Table 3-1: Summary of PASS Criteria defined by NUREG-0737 Section II.B.3	
Number	**Criteria**
1	Licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for the sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.
2	The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the three hour time frame established above, quantification of the following: Certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g. noble gases, iodines, cesiums and nonvolatile isotopes) Hydrogen levels in the containment atmosphere.* Dissolved gases (e.g. H_2), chloride and boron concentration of liquids. Alternatively, have inline monitoring capabilities to perform all or part of the above analyses.
3	Reactor coolant and containment atmosphere sampling during post accident conditions shall not require an isolated auxiliary system to be placed in operation in order to use the sampling system.
4	Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gas with unpressurized reactor coolant samples. The measurement of total dissolved gases or H_2 gas in the reactor coolant samples is considered adequate.
5	Time for chloride sample to be performed is dependent upon two factors: (a) if the plants' coolant water is seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days.
6	The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposure to any individual exceeding the criteria of GDC 19.
7	The analysis of primary coolant samples for boron is required for PWRs.
8	If inline monitoring is used for any sampling and analytical capability…the licensee shall provide backup sampling through grab samples and shall demonstrate the capability of analyzing the samples.
9	....the radiological and chemical sample analysis capability shall include provisions to: • Identify and quantify the isotopes of nuclide categories….corresponding to the source terms given in RG 1.3, 1.4 and 1.7 • Restrict background levels of radiation in the radiation and chemical analysis facility...such that the sample analysis will provide results with an acceptably small error.
10	Accuracy, range and sensitivity shall be adequate to provide pertinent data to the operator in order to describe the radiological and chemical status of the reactor coolant systems.
11	In the design of the PASS and analysis capability, consideration should be given to ..: Provisions for purging sample lines… The ventilation exhaust from the sampling station should be filtered with charcoal absorbers and HEPA filters.

* Monitoring of containment hydrogen is required by NUREG-0737. Containment oxygen measurement is recommended in Reference 4. However, this feature may be reliably assessed via use of "in plant" instruments and analytical procedures (See section 6.9).

In brief, NUREG-0737 required that the licensee establish an onsite radiological and chemical analysis capability to provide quantification of the following:

- Certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of core damage (e.g. noble gases; iodines and cesiums, and non-volatile isotopes),

- Hydrogen levels in the containment atmosphere,

- Dissolved gases (e.g. H_2), chloride (time allotted for analysis subject to discussion below) and boron concentration of the reactor coolant,

- Samples were to be taken and analyzed in less than 3 hours (24 hrs for chloride),

- The analysis capability could be inline or via grab samples. Sampling restrictions were provided with respect to time, doses and accuracy.

Systems and processes designed to meet the PASS criteria varied greatly throughout the nuclear industry. Many utilities opted to meet the time and radiological dose requirements using inline monitoring capability with minimal personnel interface. Other utilities constructed systems and processes that were based on manual sampling and analysis.

The NUREG-0737 requirements were also included to a degree in Regulatory Guide (RG) 1.97, Revision 3 (Reference 4) which contains a list of variables to be monitored during and following an accident. This RG expanded the scope of the PASS to monitor the containment sump, added the requirement to analyze liquid samples for pH and to analyze RCS samples for oxygen content. Of the added variables considered under RG 1.97 only one of the variables (RCS soluble boron concentration) was identified as a Type B variable. A Type B variable is defined as a process parameter that affects the accomplishment of a safety function. Three variables: RCS radioactivity, RCS gamma spectrum analysis and containment hydrogen concentration were considered Type C variables (that is, indicates the potential for or actual break of barriers to fission product release). The remaining variables were designated Type E and may be used to determine the magnitude of the release of radioactivity. RG 1.97 also refers to General Design Criterion (GDC) 64, "Monitoring of Radioactivity Releases," (Reference 5a). This GDC required that a means be provided to monitor containment atmosphere, spaces containing components for recirculation of LOCA fluid, effluent discharge paths, and the plant environs for radioactivity that may be released from postulated accidents.

In addition to the above explicit requirements, the PASS system was integrated within the license via 10CFR50.54 (References 5b) references to 10CFR50.47 "Emergency Plans" and 10CFR50 Appendix E: "Emergency Planning and Preparedness for Production and Utilization Facilities," (References 5c and 5d). Specifically, the PASS was integrated to varying degrees into the NRC emergency classification scheme which required 1) the diagnosis of failure of the fission product barriers as contained in 10CFR50.47 (b)(4), and 2) development of appropriate plant emergency action levels (EALs). In addition, PASS may also have been used to satisfy the 10CFR50.47(b)(9) requirement that "Adequate methods, systems and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use." A similar requirement may be found in 10CFR50 Appendix E, IV.E.2.

Appendix E, Section VI of 10CFR50 includes the requirement that in the event of an accident, the plant must be capable of transmitting specified data to the NRC emergency response center. This data will be input into the Emergency Response Data System (ERDS). In particular, Appendix E Section VI.1.a(i) element 5 requires that the ERDS include reactor coolant activity, containment radiation level, condenser air removal radiation level, effluent radiation monitors and process radiation monitor levels. However, no specific PASS data is required.

3.2 PASS Licensing Basis

PASS requirements were backfit onto all plants. The full licensing basis of the PASS includes all documented commitments related to the eleven criteria of NUREG-0737. The documented commitments primarily consist of the licensees' stated compliance to the specific criteria of NUREG-0737 item II.B.3 and RG 1.97. For some plants it may also include commitments made during the post-implementation review and subsequent PASS inspections. The PASS requirements appear in the Administrative Controls section of the plant technical specifications which very generally states:

"A program shall be established, implemented and maintained which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

- Training of personnel,
- Procedures for sampling and analysis,
- Provisions for maintenance of sampling & analysis."

Some plants also have a license condition or Confirmatory Order addressing the PASS. As a result of the technical specification commitment, the PASS is a condition of license per 10CFR50.54 and any changes to the PASS that are reflected in the technical specifications must be performed in accordance with 10CFR50.90 (Reference 5e). From the previous discussions, it appears that PASS is also a requirement associated with 10CFR50.54(q) "Emergency Preparedness." Thus, elimination of the PASS requires confirmation that no decrease in the effectiveness of the emergency plans will result from the removal/downgrade of the PASS.

3.3 Re-evaluation and Regulatory Assessments of PASS Requirements

PASS was implemented along with many other post-TMI-2 requirements. Taken in consort with these other requirements (RV level indication, improved recovery procedures, increased monitoring capability per RG 1.97, etc.) the role of the PASS appears redundant. Results of PASS samples are believed, at best, to confirm information readily available via alternative plant indications. Consequently, almost from its inception, the risk importance of the PASS has been challenged.

In 1986, the NRC published a contractor assessment of the regulatory requirements associated with 10CFR50 (Reference 6). This assessment was intended to identify those regulations, that if deleted or appropriately modified, would improve the efficiency and effectiveness of the NRC regulatory program without adversely impacting safety. The report identified 45 regulatory requirements that fell into this category. PASS requirements were identified as an area with a high level of concern associated with the regulatory burdens. This review was followed up by a more detailed assessment of PASS regulatory requirements. The assessment examined relaxation of the time requirements for samples and the elimination of some samples. However, these changes were not pursued. The PASS requirements reviewed for modification included:

- Relaxation of Time Requirement,
- Elimination of Radiological Analysis,
- Elimination of Containment Hydrogen Analysis,
- Elimination of RCS Dissolved Gas Analysis,
- Elimination of Heat Tracing Measures.

Relaxation in all of the above areas were considered to have a marginal impact on the risk to the public. However, due to the perceived lack of economic benefit to the utility none of the above were strongly recommended for operating plants. It was noted that for future plants where the capital costs would be incurred, PASS requirements would most likely be significantly reduced.

In 1990, the CEOG funded an exploratory initiative to review the PASS regulatory basis and identify potential areas for regulatory relief. This task culminated in the formulation of a generic PASS relief request which was submitted to the NRC in 1993 as a topical report CEN-415, Revision 1-A (Reference 8).

The CEOG topical report requested both clarification and relaxation of PASS requirements. This relief request was approved in part by the NRC. CEN-415, Revision 1-A provided clarification of the NUREG-0737 requirements. The purpose of this report was to identify alternative methods/approaches to meeting the intent of the requirements in NUREG-0737 II.B.3.

Specific relaxation was requested from PASS related items such as:

- Deletion of the measurement of sump pH,

- Utilization of current safety grade containment hydrogen monitors for analysis and trending of containment hydrogen concentration,

- Deletion of the requirement for heat tracing of the containment atmosphere sample line,

- Modification of RCS total hydrogen/total gas sampling methodology to include the option of using either the PASS or normal sampling system,

- Deletion of the RCS oxygen analysis requirement from PASS,

- Definition of sample points to be limited to the containment atmosphere and the reactor coolant system,

- Clarification that the time requirement in NUREG-0737 is a recommendation which applies to the first sample only,

- Additionally, preparations such as suiting up, radiation measurements, etc. may be performed prior to deciding to sample, and therefore, "prior to starting the clock".

The conclusions of CEN-415, Revision 1-A, which were approved by the NRC, in summary are:

- The elimination of analysis for pH of the sump water was found acceptable since pH is not required by NUREG-0737 and reliable means exist for pH control.

- Safety grade hydrogen monitors in the containment were considered to provide adequate capability for monitoring post accident hydrogen.

- Sampling of iodine was not required within the PASS (See also Reference 26); therefore, the heat tracing of sample lines may be removed provided core damage assessment procedures are based on representative containment atmosphere xenon and krypton activities.

- Oxygen concentration is not required by NUREG-0737; therefore, need not be part of PASS.

- The only sampling requirement is the ability to obtain a sample of the reactor coolant and containment atmosphere under all accident conditions. No specific sample points are prescribed.

Relief regarding the RCS hydrogen sample was not allowed (even though this was a likely candidate for relief per NUREG/CR-4330). Relief from the sampling time requirement was also not granted. It was further noted that "*As the decision could be made anytime during the accident management phase, PASS should be designed for obtaining analytical results within three hours from the beginning of the accident*". In the specific instance of boron concentration measurement, it was noted that for plants with neutron flux instrumentation complying with Category 1 criteria of RG 1.97, "postponing measurement of boron concentration to 8 hours after the end of power operation," could be justified.

Additional plant specific relief from PASS requirements was submitted by Westinghouse utilities including TVA (Reference 9) and most recently, Wolf Creek (Reference 10). This later application was the culmination of a Westinghouse Owner's Group (WOG) effort to effect PASS relaxation by rewriting their Core Damage Assessment (CDA) procedure to exclude PASS input (Reference 11). The WOG effort proposes the elimination of the regulatory requirements for PASS samples, with the exception of reactor coolant boron and in some cases chloride, since they are not required to ensure long term plant stability for accident mitigation and recovery. The renewed interest in PASS relief from the WOG (as well as by the CEOG) is to avert large capital (up to $2 million for WOG plants with in-line PASSs, Reference 34) and maintenance expenses associated with maintaining PASS compliant to both NUREG-0737 commitments and Y2K. Since there was no explicit discussion of RCS dissolved gas/hydrogen samples in the plant Technical Specification, Wolf Creek prepared and completed a 50.59 evaluation for eliminating this PASS sample requirement from the Updated Safety Analysis Report (Reference 7).

3.4 Advanced Reactor Certification

Relaxation of PASS requirements has also been pursued in the advanced reactor arena with mixed results. The System 80+ evolutionary PWR design certification project succeeded in relaxing specific PASS requirements, including delaying the time required for the first sample of most items to 24 hours (Reference 12). Specific relaxations accepted by the NRC included:

- Sampling for boron concentration is not needed for the first eight hours after an accident. This was based on the fact that this measurement is used only to confirm the accident mitigation measures and conditions of the core obtained by other methods and do not need to be performed in an early phase of an accident. Neutron flux monitoring instrumentation which complies with

Category I criteria of RG 1.97, will have fully qualified redundant channels that monitor neutron flux over the required power range.

- Radionuclide samples could be postponed until 24 hours following an accident. This delay was considered acceptable since these measurements provide information used in evaluating the condition of the core for the purpose of post-accident assessment, and would not be used in establishing early plant emergency responses. Analogous information will be made available during the accident management phase by monitoring other pertinent variables.

AP600 licensing experience resulted in a vague commitment for the licensee to have a PASS system capable of obtaining and analyzing source term data promptly and continuously in order to assess and refine dose measurements, and to confirm or modify initial protective action recommendations. No specific relief from NUREG-0737 was taken (Reference 13).

3.5 Recent Regulatory Positions

A clear recognition exists that the initial intent of using PASS sampling as a means to guide and quantify with precision the consequences of a severe accident are neither practical nor needed. Adoption of post TMI "fixes" (such as the Reactor Vessel Level Monitor) and implementation of SAMG at all reactors in the United States have enabled plant personnel to use familiar plant instruments to both take actions to mitigate severe accidents and to trend their progression.

Accordingly, the NRC has directed considerable effort towards the understanding of the relationship between PASS and various emergency planning procedures. Specifically, these procedures involve establishing EALs, providing Protective Action Recommendations (PARs) to State and local authorities, providing data to the ERDS and establishing a CDA. To this end, it was the NRC staff's opinion that elimination of PASS requirements should be justified subject to 10CFR50.54q (Decrease in the effectiveness of the emergency plan) (Reference 19).

This report demonstrates that the PASS may be eliminated without a degradation of the emergency planning procedures. The following sections explore the current uses of PASS samples at CEOG member utilities and establish technical bases for elimination of existing PASS requirements. Where appropriate and where the "intent" of PASS sample remains valid, alternative methods to accomplish these PASS-related functions are recommended.

3.6 Impact of PASS on Related Regulations

Installation and implementation of PASS has resulted in unanticipated impacts on plant severe accident performance. Based on plant-specific experience using PASS, several utilities have included leakage from PASS during post-accident plant sampling in demonstrating adherence to 10CFR100.11 and GDC-19 exposure guidelines. For example, a 350 cc/hr leakage from the PASS is estimated to contribute approximately 2 rem to the thyroid doses at the exclusion area boundary and low population zone. PASS leakage is also estimated to contribute 1.7 rem to the Control Room thyroid dose.

Recently, a design flaw in the PASS delivery system at Fort Calhoun Station was identified and resolved. Prior to the issue resolution, PASS operation could potentially overpressurize the waste disposal system containment isolation valves. The design change involved installation of pressure relief valves. Although the revised design meets PASS requirements, operation of the PASS during a severe accident may increase radiation exposure to plant personnel during beyond design basis events and reduce accessibility to the auxiliary building while the PASS is in operation.

4.0 INTER-RELATIONSHIP OF ACCIDENT MANAGEMENT AND EMERGENCY RESPONSE PROGRAMS AT CEOG UTILITIES

This section provides a high level overview of the relationship among the various programs and procedures CEOG plants currently employ to ensure public health and safety is protected when responding to and mitigating reactor accidents. The purpose of this exercise is to clarify the roles and inter-relationships of these programs and procedures as they relate to the ability of the plant to effectively:

1. Diagnose and trend the consequence of design and beyond design basis accidents (including severe accidents),
2. Provide adequate information to state and local authorities and,
3. Provide appropriate Protective Action Recommendations (PARs).

In the context of this discussion, operating nuclear reactors respond to accidents with a comprehensive set of programs and procedures. The programs are generally hierarchical in that less severe events invoke fewer programs and/or associated procedures. These programs have been either a direct or indirect result of post-TMI-2 recommendations. The first lines of defense for responding to an accident are the plants Emergency Operating Procedures (EOPs)* and associated Abnormal Operating Procedures (AOPs). These procedures provide direction to the operator, using Class 1E "in-plant" instrumentation to diagnose the event and take effective mitigating actions. The CE EOPs include both Optimal Recovery Guidelines (ORGs) for cases when a specific type of event is diagnosed and Functional Recovery Guidelines (FRG) which are focused entirely on a Safety Function restoration concept (Reference 14). These procedures are implemented upon accident initiation. As will be discussed in detail later, neither of these procedures relies on PASS information to ensure plant safety functions are met. Entry into the EOPs/AOPs would also be concurrent with initiation of the plant emergency plan.

In the event that the EOP actions are found to be ineffective in controlling the event, the site emergency director will recommend that the Technical Support Center (TSC) enter the SAMGs. The implementation of SAMGs at member utilities is relatively recent. Currently SAMGs have been implemented at all CEOG member utilities. SAMGs reflect an integration of recent understanding of severe accident phenomenology with insights gained from plant specific Level 1 and Level 2 PSAs. Thus, they are inherently focused on quickly understanding the event progression and protecting the next fission product barriers. SAMGs also provide guidance on mitigation of ongoing releases. In short, these guidelines provide a structured process for diagnosing, and mitigating the consequences of a severe accident. To effectively perform this task the CEOG SAMGs and associated Calculation Aides have drawn on key information and guidance from the CEOG Core Damage Assessment methods. However, the SAMGs differ from the Core Damage Assessment Methodology (CDAM), in that the focus of the CDAM is post-accident quantification of the extent of core damage, whereas SAMGs are structured to make instantaneous assessments of the level of core degradation and the direction of the core melt progression or recovery. While PASS information is allowed within the scope of the SAMGs, the guidance is directed towards reliance on in-plant instruments including Core Exit Thermocouples (CETs), Reactor Vessel Level Monitoring System (RVLMS) and its associated Unheated Junction Thermocouples (UHJTCs), Hot leg RTDs, RCS pressure sensors, containment pressure, temperature, hydrogen concentration and radioactivity (as measured by Area Radiation Monitors (ARMs)) and various ex-core neutron detectors. In the SAMGs, when hydrogen monitoring is called for, use of the Class 1E in-line plant containment hydrogen analyzers is recommended.

* SONGS Units use the term Emergency Operating Instructions (EOIs)

Another feature of the CEOG SAMGs is the ability to focus on public risk by highlighting risk significant accident conditions. Public risk is primarily dominated by the state of the containment. So long as an intact containment exists (and the containment is not bypassed) radioactive releases to the public will have negligible health consequences. The SAMG highlights those containment situations where the containment has a potential for failure, the containment is already impaired (due to open penetrations or other leakage) or the containment is bypassed. This information may be readily factored into the Emergency Planning Process for the development of PARs. Determinations of the containment state is based on transient event progression data, RCS and containment pressures, potential threats to containment integrity and offsite and vent point radiological release surveys. As discussed above, when hydrogen monitoring is needed use of the Class 1E in-line plant containment hydrogen analyzers is recommended.

The information from the SAMGs is capable of being integrated into the Emergency Plan. When this is done, the emergency planners will be informed as to the state of the core, the progression of the event, the prospects for recovery and the existing and projected containment status. The site emergency plan as required in Reference 5c and 5d directs plant personnel to assess and classify the event, estimate the degree of core damage and formulate offsite Protective Action Recommendations (PARs). The intent of these actions is to ensure that the health and safety of the public is preserved.

Revision 3 to Regulatory Guide 1.101 endorses two emergency planning classification schemes (See References 17 and 37). Four emergency classifications are identified: Notification of an Unusual Event (NOUE), Alert, Site Area Emergency (SAE) and General Emergency (GE). Examples of initiating conditions for selection of these classifications are captured in RTM-96 (Reference 20) and the current NEI Guidance document (Reference 21). Classification of events normally is done quickly and should be able to be done based on in-plant instrumentation. For most events numerous appropriate indictors are directly available to the plant staff, such that an effective diagnosis may be made. In practice, while PASS monitoring is indicated as input for the emergency classification process, the delays in taking and analyzing the sample render it ineffective in this context.

Notification of an Unusual Event include events with very limited clad damage or transients with iodine spiking. However, even in designation of an NOUE via sampling, the actual sampling (if it is to be performed) may be done via the Normal Sample System or the designation may be established via some other non-PASS approach.

The Alert classification for most CE plants includes a high reactor coolant sample activity > 275 to 400 μCi/cc I^{131} (See Appendix C). The monitoring of doses at this level typically is assigned to the PASS system, however alternate means for establishing this declaration are possible. This classification was initially intended to cover those transients that result in localized core damage (~ 2-5% failed fuel) without significant RCS inventory loss. Such events include reactivity addition transients (main steam line breaks and CEA ejection) and seized rotor accidents. Estimated doses are based on design basis considerations associated with localized fuel damage caused by a number of fuel rods locally experiencing a departure from nucleate boiling (DNB). As will be discussed later, more realistic analyses suggest fuel damage from these events would be negligible. In any event, delays associated with the PASS sample process typically render this emergency classification unused. Instead, Alert declarations are typically based on event characteristics or alternate sampling approaches.

The last two emergency classifications, Site Area Emergency and General Emergency, readily emerge when the event poses additional challenges to plant fission product barriers. While many existing EALs indirectly rely on PASS input, due to references to lesser emergency classes and the CDA (to be discussed later), in actual practice the data collection and interpretation inherent within the EOPs and/or SAMGs will provide abundant information to the plant staff such that these assessments may be made quickly and accurately.

A second feature of the emergency plan is the requirement to formulate PARs. PARs require consideration of:

- Extent of core damage,
- Fission product inventory released to containment,
- Containment status (e.g. closed and cooled, challenged, impaired or bypassed),
- Site meteorology.

Decisions to be made include the need for, and extent of, evacuation and need for shelter. The initial PAR, as indicated in Supplement 3 to NUREG-0654 should be based on the event progression and other plant conditions. The subsequent PAR decision process typically includes consideration of measured or projected dose at various distances from the plant. In order to establish the extent of core damage, the emergency planners typically utilize the CDA procedures based on CET readings, and containment radiation.

Treatment of fission product releases will vary dependent on the type and location of the release. Releases that bypass the containment are typically measured via site and field radiation surveys. Bypass events are rare scenarios that allow transport of fission products directly from the RCS into the environment (e.g. auxiliary building). These events are typically associated with large releases of fission products. Bypass events with core damage will rapidly escalate to a SAE or GE. Offsite thyroid doses will be governed by plateout in the bypass lines and on equipment and walls at the release points. If the release point is via a steam generator with an open main steam safety valve, Steam Generator (SG) plateout and scrubbing (if the SG is flooded) would also contribute to dose reductions. Dose predictions during these scenarios will be difficult and the availability of PASS would not reduce the uncertainty associated with these estimates.

Projections of intact, challenged and impaired containment fission product releases (and in particular iodine releases) may be established based on considerations of iodine isotopic decay, available fission product removal systems (e.g. containment sprays and charcoal filters) and expected leakage rates. Calculations are typically established by adjusting ARM readings to reflect the expected fission product containment composition of radioisotopes. The methodology assumes a radioisotope distribution. Physical uncertainties in the release and transport process, among other issues, will tend to cause predictions of releases to be different from those actually experienced. Among the issues that contribute to this difference include:

- Impact of particulate filtering in the leakage pathways (that is if the leakage is due to cracks in the containment, what percentage of CsI will filter out in the 3 foot thick structure),
- Imprecise knowledge of the containment leak rate,
- Uncertainty in the airborne composition and constituents.

PASS airborne radioisotope samples, while quantitative, will not ensure added validity to the assessment due to inherent issues associated with the location of the sample point and distribution of iodines in the RCS and containment, CsI plateout in the sample line, time delay between measurement and analysis, and consequences of containment pressure changes on iodine re-evolution from the sump. If sample line plateout is significant (as would be expected) analyzed samples could significantly underpredict airborne iodides.

Based on the above, it can be concluded that while precise dose projections cannot be expected, projections will be adequate to reasonably assess public risk and define appropriate actions. Both NUREG-0737 and Reg. Guide 1.97 require having monitoring and sampling capability for all identified plant release paths. Additionally, some plants include relatively sophisticated dose survey equipment for assessment of doses due to particulates, iodines and noble gases (SuperParticulate, Iodine, Noble Gas (SPING) detectors) released at various site locations. To confirm the projected doses, emergency plans also include area radiation surveys including the assessment of iodine releases. These surveys vary from plant to plant. Data from site and off-site surveys can be used effectively to both assess off-site releases and improve dose projections.

Another element integrated into the emergency plan is the CDA methodology. With the implementation of the SAMG, the role of the CDA is now primarily a post-accident, after the fact quantification. It should also be noted that the CDA need not be precise. However, it should provide sufficient information to the plant staff to assess, along with other indicators, the level of potential volatile material released from the core. Specifically, the results of the CDA should support longer term PARs. To meet this objective it is sufficient that the CDA be able to distinguish among core damage events that result in negligible (< .1% failed fuel (FF)), small (< 5% FF), and significant (> 10%-20% FF) fission product releases from the fuel. Gross core damage is not generally identified in typical PARs, but may be readily assessed via many non-PASS CDA approaches.

The CEOG CDA was conceived as a four (4) pronged graded approach to quantifying core damage. The CEOG approach relies on CETs, area radiation monitors, hydrogen production and radioisotope sampling. The first two assessments may, together, be used to distinguish the continuum of core damage states ranging from a core with a limited number of fuel rod failures to one in which gross core melting has occurred. It should be noted that once significant fuel overheating has occurred, the core will have released all of its noble gases and a large quantity (up to 50%, Reference 22) of its initial iodine and cesium inventory. The hydrogen production CDA is a parallel evaluation and confirms the level of core damage by estimating the extent of cladding oxidation. The last CDA, radioisotope sampling, was intended to be a detailed precise methodology for quantifying core damage. The procedure involves isotopic measurements obtained via the PASS, followed by an extensive spectroscopic analysis of the results. Based on current understanding of fission product plateout and transport, there is little expectation that the sample will provide sufficiently accurate information to improve upon the assessment that would be obtained via the simpler CDAs integrated within the SAMG accident progression.

The last item included in the context of emergency planning is the interface of plant accident data with the Emergency Response Data System (ERDS). The requirements of ERDS are defined in 10CFR50 Appendix E, Section VI. All CEOG plants conform to ERDS requirements without reliance on PASS data.

Following a severe accident, eventually there will be a need to enter the containment building for assessment and clean-up. Plants are required to establish containment re-entry procedures. These procedures may be used to control personnel access to the containment following severe accidents.

Procedures currently in use at CEOG plants require pre-entry radiation assessments. PASS is not specifically required. It is expected that long term containment re-entry and clean up programs may require use of portable equipment to be brought on site.

In summary, the NRC has mandated that plants implement a robust set of programs and procedures to respond to severe accident situations. As knowledge was gained in severe accident dynamics and fission product behavior, the focus on the type and amount of information needed to mitigate a severe accident has evolved. Consequently, more recent programs (notably the SAMGs) have effectively superseded various aspects of older programs (notably the CDAs). This evolution has resulted in timely and robust accident management and emergency planning strategies that currently have minimal reliance on the PASS and that could readily be divorced from PASS sampling entirely. The remainder of this document focuses on the various elements of the plant accident response program and addresses the emergency response actions and the impact of eliminating the existing PASS.

5.0 ASSESSMENT OF THE UTILIZATION AND EFFICACY OF THE PASS WITHIN THE CEOG ACCIDENT MANAGEMENT AND EMERGENCY PLANNING PROGRAMS/PROCEDURES

This section provides a description of how the CEOG implementation of PASS (consistent with the direction and intent of NUREG-0737) is generally incorporated into CEOG member EOPs, severe accident management and the various elements of the plant emergency plan. While the text of this section discusses these issues in general, a plant specific listing of PASS-related references in the relevant procedures is provided in Appendices A through E. For convenience, the highlights of this survey are summarized in Table 5-1.

The remainder of this section is divided into four sub-sections. Each sub-section focuses on a particular program which could (or does) use PASS sample data. The first of these sub-sections deals with issues associated with the use of samples (and potential use of PASS) in EOPs* and AOPs. The second sub-section concentrates on potential PASS related overlap with severe accident management guidance. The third sub-section deals with the various potential uses of PASS samples in emergency planning issues. These include, the role of PASS in regard to triggering initiating conditions for declaring emergency classes formulating PARs and performing the core damage assessment. The last sub-section discusses current containment re-entry procedures.

* Alternate designations may also be used for similar programs. For example, SCE refers to these procedures as Emergency Operations Instructions (EOIs).

Table 5-1: Utilization of PASS Within Plant Programs

Parameter	EOP Safety Functions: Reactivity Control	EOP Safety Functions: Combustible Gas Control	EOP Safety Functions: Other Safety Functions	EOP Long Term Actions	SAMGs	EPlan
RCS BORON CONCENTRATION	1. Used to verify adequate SDM 2. Confirm boron concentration prior to RCP restart during SGTR event. 3. Monitor for boron dilution during SGTR event.	N/A	N/A	SD Margin reactivity control verification. Reactivity control verification may be accomplished via alternate means such as: 1) Monitoring reactor neutron count rate/power 2) Emergency boration 3) CEA position	Use optional. PASS may be used as an alternate means to confirm reactivity control for "In Vessel" Recovery for core damage sequences with "intact" geometry. Not needed for severely degraded cores	N/A
RCS TOTAL GAS AND HYDROGEN CONCENTRATION	N/A	Not used when taking actions for eliminating RCS voids.	N/A	N/A	N/A	Used as "fine tuning" in current CDAM application. may be eliminated without impacting CDA.
pH OF REACTOR COOLANT	N/A	N/A	N/A	Post LOCA pH controlled passively via TSP baskets. pH control mitigates SSC and Iodine revolatilization. pH control does not require PASS sample	N/A	N/A
RCS CHLORIDES	N/A	N/A	N/A	Minimize Stress Corrosion Cracking (SCC)	N/A	N/A

Table 5-1: Utilization of PASS Within Plant Programs

Parameter	EOP Safety Functions			EOP Long Term Actions	SAMGs	EPlan
	Reactivity Control	Combustible Gas Control	Other Safety Functions			
RADIONUCLIDES IN REACTOR COOLANT	N/A	N/A	Awareness of RCS activity helpful for anticipating radiological consequences in auxiliary building prior to initiating SDC. PASS Radioisotopic assessment not required.	N/A	No	I^{131} equivalent activity primarily used to differentiate between no fuel damage and low fuel damage conditions (<5% FF) (See Note 2). Used for NOUE/ALERT. Not directly used for SAE, GE declarations. PASS information not expected to be available in time to declare EAL. Information may be used to support alternative CDA.
CTMT HYDROGEN	N/A	For most plants redundant Safety Grade hydrogen monitors are used to track containment hydrogen. (1) Monitoring provides no significant design basis safety function.	N/A	PASS is backup to redundant Safety Grade hydrogen monitors. (2)	Hydrogen measurement may be used to track core damage and assess containment failure potential. PASS is backup to redundant Safety Grade hydrogen monitors. Most CE PWRs have eliminated PASS H_2 sample in accordance with CEN-415-A.	Used to support "Hydrogen" CDA. CTMT hydrogen concentration may be assessed via non-PASS requirements

(1) Function primarily "investment protection" (averting damage to H_2 Recombiners)

(2) In practice, ALERT declaration will be made on other accident parameters. Use of PASS for low levels of CD (< 5% FF) may be largely replaced via alternate measurement techniques including use of NSS.

Table 5-1: Utilization of PASS Within Plant Programs						
Parameter	**EOP Safety Functions**			**EOP Long Term Actions**	**SAMGs**	**EPlan**
	Reactivity Control	**Combustible Gas Control**	**Other Safety Functions**			
CTMT RADIONUCLIDES	N/A	N/A	Radio-isotopics not required (3)	No	No	Used in existing CDAM for estimating severe core damage conditions. Varies among plants. Does not affect E-plan actions, EALs, PARs. Trending of radiation releases accomplished via use of ARMs and calculational tools. Isotopics, if desired, may be assessed via SPING monitors and other site surveys. PASS serves as an optional input in some existing programs.

(3) Containment activity monitored via ARMs used for (1) criteria for securing containment sprays, (2) estimate release rates for purge and venting operations, (3) projecting radiation levels outside containment, and (4) preparation of RWP for containment entry.

5.1 Use of PASS Samples in Plant EOPs/AOPs

5.1.1 Overview of CE Emergency Procedure Guidelines

CEOG member plant Emergency Operating Procedures are derived from the CE Emergency Procedure Guidelines (EPGs). The EPGs are a collection of the best available technical information to be used for writing Emergency Operating Procedures (EOPs). The CE EPGs were modified several times since TMI-2. The intent of the EOPs (and associated AOPs and other implementing procedures) is to allow operators to trend the event progress and take appropriate mitigative actions.

The CE EPGs are generally divided into two parts, Functional Recovery Guidelines (FRG) and Optimal Recovery Guidelines (ORGs). The ORGs are event specific emergency procedures. Functional Recovery Guidelines are functional in nature and are focused at providing operators sufficient information to assess, control and trend the status of the plants' safety functions without knowing what event is in progress. The CE EPGs define nine safety functions. These are:

1. Reactivity Control,
2. RCS Inventory Control,
3. RCS Pressure Control,
4. Core Heat Removal
5. RCS Heat Removal,
6. Containment Isolation,
7. Containment Temperature and Pressure Control,
8. Containment Combustible Gas Control and
9. Maintenance of Vital Auxiliaries.

FRG are intended for use when the event cannot be diagnosed. If a firm diagnosis of the event can be ascertained, the operators would use Optimal Recovery guidance available. Guidance contained in the ORGs optimizes the mitigating actions by sequentially listing the operator actions necessary to strategically address that symptom set.

In order to identify the role of the PASS in the plant EOPs, a review was performed of the CE EPGs (Reference 14), as well as, individual utility EOPs/AOPs. This review indicated three uses of the post accident sampling of RCS coolant or the containment atmosphere during certain events. These are:

1. Determination of boron concentration of RCS to ensure adequate shutdown margin.

2. Determination of containment hydrogen concentration to determine appropriate time to initiate the hydrogen recombiners or purge.

3. Determination of RCS activity, prior to entry into shutdown cooling.

As will be discussed below, all of the recommended sample uses are confirmatory in nature (that is the outcome of the sampling does not result in direct action) and the sample or the verification may be established via non-PASS systems/equipment.

5.1.2 Evaluation of EPG/EOP Sampling Recommendations

5.1.2.1 Evaluation of Post Accident Determination of Boron Concentration of RCS Coolant to Ensure Adequate Shutdown Margin

5.1.2.1.1 Basis for Boron Sampling

The reactor coolant system contains some level of boron, in the form of boric acid, for use as a chemical shim to control reactor power. The RCS boron concentration within the reactor coolant is initially based on the reactivity of the core. At the beginning of a fuel cycle boron concentrations may be in excess of 1000 ppm, while at the end of a fuel cycle, the boron concentration may be negligible (< 50 ppm). Following a safety injection signal, the emergency core cooling pumps inject borated water from a Refueling Water Storage Tank (RWST) which contains a high concentration of boric acid. Any water injected into the containment via spillage also originates in the RWST. If Safety Injection (SI) continues to operate (e.g. a LOCA), the RCS boron concentration will approach the RWST boron concentration.

In assessing the need for a boron sample, several accident sequences are considered. Boron concentration is needed to confirm adequate shutdown margin to protect the core from a return to criticality and maintain it subcritical. CE PWRs are designed such that adequate shutdown margin is assured when the plant is tripped and hot with no more than one Control Element Assembly (CEA) stuck out. The typical generic success criteria for reactivity control is to use the chemical volume and control system or Safety Injection System (SIS) and ensure that the reactor power is < **[--%]** and constant or decreasing:

And all CEAs are inserted, OR
Boration rate > [40 gpm] and reactor power decreasing, OR
Adequate shutdown margin established per technical specifications and reactor power constant or decreasing

While reactivity control is essential following all accidents, the concern over boron concentration is of primary concern following: Steam Generator Tube Ruptures (SGTRs), due to the possibility of backflow of unborated secondary water into the RCS; a station blackout when there is no way to inject boric acid into the RCS; and large LOCAs without the ability to implement hot side/cold side injection (due to high boron concentration issues). The role of boration and boron sampling following these events are discussed in the paragraphs below.

5.1.2.1.1.a Steam Generator Tube Rupture with backflow of secondary liquid into the RCS

During a SGTR event the RCS boron concentration may be diluted by inflow of secondary side inventory. The RCS pressure is normally controlled to be greater than, and close to, the pressure in the secondary side of the SG to minimize the backflow through the ruptured tube. Analyses have shown that a reactivity addition threat may occur due to the formation of a slug of low boron concentration water in the affected loop RCS cold legs during a SGTR under natural circulation conditions and with potential backflow of secondary side water into the primary side. Therefore, the EPGs discourage the use of SG backflow during natural circulation except when needed to control the level in the affected SG. The EPGs include precautions to disable RCPs in the affected loop and requires sampling for boron whenever backflow is occurring. The RCS boron concentration is required likewise to be determined prior to restarting an RCP following natural circulation operations whenever SG backflow has occurred. The intent of these actions is to avoid sweeping a possible slug of low boron concentration water from the cold leg to the core.

It is expected that events currently requiring boron sampling will have minimal (or no) core damage and the sampling would be performed via the NSS. Even so, it is recognized that it will be difficult to get a representative RCS sample with the Reactor Coolant Pumps (RCPs) off. This is further complicated by the fact that the sample port is in the hot leg and the boron concentration of primary interest is in the cold leg. Therefore, the EPG strategies are designed to minimize the deboration threat for worst case situations, even when the RCS boron concentration is unknown.

In all cases, the reactivity of the core is monitored constantly via the ex-core nuclear instrumentation and start-up range neutron detectors to confirm that a subcritical condition is maintained. In addition, the operator is trained to initiate and maintain emergency boration whenever there is any doubt about reactivity control.

5.1.2.1.1.b Station Blackout

Monitoring RCS boron concentration during a Station Blackout condition is necessary because there is no means available to provide makeup to the RCS at the same time the RCS is cooling down. As the RCS temperature decreases the shutdown margin decreases. The EPGs provide instructions to stop the cooldown if there is a possibility that continued cooldown will result in a loss of shutdown margin and a return to criticality.

For these events, when power is restored, boron samples may be obtained via the use of the NSS. While taking a boron sample is a recommended action in most EOPs, it is not necessary to know the current boron concentration via sampling. Instructions are provided for predicting the change in Shutdown Margin (SDM) based on the anticipated change in RCS temperature.

5.1.2.1.1.c Large LOCAs

The plant's automatic response to large LOCAs is to provide large quantities of highly borated water into the reactor vessel. Appropriate operation of the minimum safety systems (as defined in the EOPs) will ensure that the plant will remain subcritical. Since the long term core cooling mechanism for some LOCAs is boiloff, the boric acid dissolved in the injection water would tend to concentrate in the RCS. As the system cools the boric acid solubility decreases and precipitation of the boric acid may occur. To avert this situation, the EOPs instruct the operator to realign safety injection to discharge water to both the reactor pressure vessel hot and cold sides simultaneously. This action will limit the buildup of boron in the core region.

The EOPs do not require analysis of the RCS for boron prior to initiating simultaneous hot and cold leg injection. It is done based on time of event initiation only. The time to initiate hot side cold side injection varies among CEOG utilities from a minimum time of 2 hours after event initiation to times well in excess of one day. In any event, the boron concentration as a function of time may be estimated via calculation since all injection flowrates, boron sources and boiloff rates are known. Design basis estimates of boron accumulation is also computed in safety analyses and is directly integrated into the timing estimate for accomplishing the hot side cold side injection operator action. Furthermore, the boron sample of reactor coolant in the RCS hot leg or the emergency sump will not be representative of the core region. Therefore, the results of the sample should not alter operator recovery actions. In fact, if not properly interpreted, the lower boron concentrations in the hot leg or the sump may lead to incorrect conclusion that the core is inadequately borated, as happened at TMI-2 (See Reference 23, Appendix BB).

5.1.2.1.2 Evaluation of Boron Sampling Requirements within EPGs/EOPs

CEOG EPGs and plant specific EOPs and associated implementing procedures currently indicates the following uses of boron concentration sampling:

1. *Ensuring that the reactor remains shutdown during a cooldown:*
   Each ORG and the FRG has steps that direct the operator to "Borate the RCS to maintain the reactor shutdown throughout the cooldown." Implicit in this direction is the ability to determine boron concentration initially and periodically throughout the cooldown.

2. *Verification of Reactivity Control Safety Function Status Check:*
   All of the reactivity control safety function status checks contain acceptance criteria that ask: "Is adequate shutdown margin established?" Implicit in this direction is the ability to determine boron concentration for use in the reactivity balance calculation.

3. *Confirming RCS water is borated prior to Reactor Coolant Pump Restart:*
   Some EOPs require the operator to confirm adequate boration (via a sample) prior to RCP restart. The intent of this check is to ensure a slug of underborated water is not sent to the core. It is generally expected that the NSS will be available to make this assessment.

4. *Ensuring adequate boration of the RCS following backflow during a SGTR:*
   Some CE EOPs require the operator to ensure the boration in the RCS following SGTR with secondary side backflow into the primary side. This assessment may be established via alternate means including calculations. Since samples are "required" every 15 to 30 minutes, the EOPs tacitly assume a NSS sample (PASS samples may take up to several hours per sample).

5. *Ensuring that the reactor remains shutdown during a cooldown in Station Blackout (SBO):*
   This is a variation of item 1 which is adapted specifically for a SBO. In SBO, the operator is directed to verify that the reactor will remain shutdown at 50 °F intervals less than indicated T_{cold} and boron concentration at the time of event initiation. In a blackout, there is no way to borate the RCS. The purpose is to predict the reactivity change associated with negative reactivity feedback due to the change in moderator temperature. If it looks like the reactor may go critical if the cooldown continues, then the operator is directed to stop the cooldown.

For non-LOCA accident scenarios the boron concentration verification, when and if performed, is expected to be accomplished via use of the NSS. Typical instructions may require samples as often as every 1/2 hour while the ability to take the processed samples may exceed 1 1/2 hours. In all cases where boron concentration is directly or indirectly referred to, there are always other corroborative indications that can be used by the operator to verify that the reactor is shutdown and will remain shutdown. For example, it is noted in the EOPs that return to criticality may be averted by verifying CEA position indication and insertion limits, ensuring that reactor power is lowering or stable, confirming a negative startup rate, or confirming positive indication of the flow of a high concentration of boron into the RCS.

When sources of boron are known, analytical assessment of SDM can be made based on injection of known borated sources into the RCS/Containment. In cases where undefined amounts of non-borated water enters the containment (for example via breaks in the component cooling water system) the operator will be alerted to additional unexpected quantities of water in the containment via increasing containment water level monitors and decreasing component cooling water inventories. The operator will then take actions to inject adequate boron into the RCS and confirm decreasing reactor power and/or a negative core startup rate.

5.1.2.2 Evaluation of the use of PASS to determine containment hydrogen concentration to determine appropriate time to initiate the hydrogen recombiners or purge

5.1.2.2.1 Basis for Hydrogen Concentration Sample

Following selected design basis LOCAs, hydrogen may be introduced into the containment atmosphere as a result of:

1. Oxidation of the zircaloy cladding during the core uncovery and recovery process,
2. Radiolysis of water in the containment and RCS due to the presence of fission products within the reactor coolant, and
3. Oxidation of galvanized steels, zinc based paints and aluminum materials in the containment due to extended exposure to high temperature steam environment.

The timing and significance of the containment hydrogen levels varies based on (1) the severity of the LOCA (time of core uncovery, number of fuel pins failed), (2) assumed levels of hydrogen production due to radiolysis, and (3) the effectiveness of containment heat removal. The containment design basis requires that the containment hydrogen level be maintained below the hydrogen dry air lower flammability limit of 4 v/% hydrogen (Reference 24) for 30 days following the most limiting large LOCA (with minimum safeguards available and assuming conservatively high hydrogen production rates due to radiolysis and oxidation). This is typically reflected in the EOPs by requiring the initiation of hydrogen recombiners (if available) once the containment hydrogen levels reach 0.5 to 2 volume percent hydrogen (dry air). (It should be noted that not all CE plants require action in 30 days and this step may not be in the EOP. Also since the consequence of low concentration burns are not considered significant to large dry containments, the actual significance of this step, as it relates to accident management and the protection of public safety, is minimal).

5.1.2.2.2 Evaluation of Containment Hydrogen Sampling Requirements within the EPGs/EOPs

The EOPs reference hydrogen concentration insofar as it is used to establish the global containment concentration so that the hydrogen recombiners will be started prior to the containment atmosphere reaching a flammable condition. For the more likely (of the low probability) design basis LOCAs, the need to start hydrogen recombiners in a thirty day period following the event is very low. If the ignition of a flammable mixture [≥ 4 v/%] results, no harm will be done to the containment or to safety grade equipment needed to continue with long term decay heat removal. In any event, CE plants can monitor containment hydrogen via safety grade hydrogen monitors which are PASS independent. This option was granted to the CEOG via a 1993 PASS relief request (See Reference 8). In summary,

1. The likelihood of needing to take the action to initiate recombiners is unlikely,
2. Should the action to initiate the hydrogen recombiner not be taken, the health and safety consequences of any potential burn are negligible, and finally
3. If the measurement is desired (to protect the recombiners), the containment hydrogen concentration may be established via safety grade hydrogen monitors that have no reliance on the

PASS.

Consequently, it is recommended that if the hydrogen recombiner operation is retained, the hydrogen concentration sample be continued from the perspective of "investment protection" (i.e. not wishing to damage the recombiner). However, PASS is not required to perform this task.

5.1.2.3 Evaluation of the Normal Sampling System for Events with limited Core Damage

The Nuclear Sampling System (NSS) is capable of sampling the reactor coolant with failed fuel (See Table 5-2). The ability of the NSS to sample at greater fuel failure levels will improve as time passes during an accident. After one day (24 hours) has elapsed, the radiation level associated with a gap release is expected to be reduced by a factor of 5. Thus RCS samples could be taken via the NSS for core damage events with ~ 5 to 10% failed fuel (See Table 5-2). This range of sampling capability is sufficient for the vast majority of accidents anticipated within the EOPs. Incidents with greater failed fuel fractions 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 RWST water. In general, the Nuclear Sampling System is expected to support the sampling requirements recommended in EOPs.

Table 5-2:
Typical Capability of Normal Sampling System

Plant	Anticipated Operational Range (% Failed Fuel, FF)**	Comments
Fort Calhoun Station, OPPD	~ 7	May be extended to greater range via use of smaller (35 ml) sample bottles.*
St. Lucie Unit 1	***	May sample up to 10 man-rem (covers > 275µC/gm of I^{131}
St. Lucie Unit 2		
Waterford Unit 3	***	-
Arkansas Nuclear One, Unit 2	3-4% Fuel overheat >5% Failed Fuel	Existing procedures utilize NSS up to 0.025% FF
San Onofre Units 2 & 3, SCE	> 10	NSS isolation valve close on high radiation (~ 350 mrem/hour)
Palo Verde Units 1, 2 & 3, APS	***	> 1% Failed Fuel (minimum estimate)
Millstone Unit 2	***	-
Palisades	***	-

* FCS has the capability to sample RCS coolant containing 6.7 µC/gm of I^{131} with an exposure of ~ 8 mrem.
** Failed Fuel (FF) Anticipated operational range based on fuel rod fission products "gap" releases.
*** Conversion to failed fuel not available at this time.

PASS samples were not generally intended for post-accident monitoring. The PASS was required to be capable of taking and analyzing a sample ~ 3 hours after the initiation of core melt (See Reference 26). No other timing requirement is imposed on this sampling. ALARA considerations and potential leakage issues may result in minimizing the frequency for RCS sampling.

5.1.3 Evaluation of the use of the PASS for post-accident RCS activity determination prior to entry into Shutdown Cooling

The EPG cautions that the operators should be aware of the coolant activity prior to shutdown cooling (SDC) entry. The intent of this action is to ensure that the plant staff is aware of the core and RCS status prior to SDC entry so that unnecessary personnel exposure is avoided. The time of entry into SDC is based on the plant conditions and availability of alternate means of decay heat removal. Entry into SDC will not be prevented based on high radiation levels in the RCS coolant. Specific radioisotopic assessment is not required. This awareness may be established via dose measurements in the vicinity of RCS coolant sampling locations, the event progression, and/or CDA based on CET readings, ARMs and containment hydrogen. Thus, PASS samples are not required to meet the intent of this action.

5.1.4 Conclusions

The ability to establish reactivity control via monitoring of shutdown margin and recriticality should be retained in the EOPs. Boron sampling of RCS fluid should be retained as a recommended action in the EOPs with the expectation that any sampling to be performed will be accomplished via the NSS. During the time the NSS cannot be used to perform these assessments, alternate strategies should be used to ensure reactivity control. PASS measurements should not be required. An assessment of existing NSSs indicates that the systems would be continuously available for most events considered in the EOPs. For events leading to significant core damage (up to about 10% fuel failure) normal sampling capability may be reestablished within 24 hours following reactor trip. In practice, the need for hydrogen concentration measurement within the EOPs is not required to protect the public health and safety (See example, Reference 35). This measurement can be retained in the EOPs from the perspective of "investment protection;" however, alternate non-PASS methods are available (i.e. RG 1.97, hydrogen analyzers) for accomplishing this assessment that have previously been approved by the NRC.

5.2 Use of PASS within the CEOG Severe Accident Management Guidelines (SAMGs)

5.2.1 Background

The CEOG SAMGs provide a structured process for the plant operating staff to prioritize, select and implement actions to mitigate the consequences of a severe core damage sequence. Entry conditions into the SAMGs vary, but typically SAMGs are entered when EOPs are no longer capable of successfully coping with the event. The appropriate SAMG response to a severe accident is focused on the implementation of one or more of approximately 14 Candidate High Level Actions (CHLAs, See Table 5-3, See also Reference 15).

Accident Management Guidelines (AMGs) have been developed by ABB for the CE Owners Group. The intent of the AMGs is to provide organized information to the utility Technical Support Center (TSC) during the various stages of a severe accident. The CEOG Generic AMGs are structured as guidance material to be utilized solely by personnel in the TSC, for use in mitigating, trending and diagnosing the progression of severe accidents.

There are no specific entry conditions into the CEOG AMGs. Typically the AMGs are opened and the process for diagnosing the plant condition is initiated based on a decision from the site emergency director. This decision will be based on many factors including the current plant emergency class, the type of accident in progress, the readiness of the TSC and input from the plant shift supervisor and other control room personnel.

The generic AMGs utilize a multi-phase process that is depicted illustratively in Figure 5-1. The CEOG AMG uses a phased, symptom-based process for diagnosing the state of the core and prioritizing the implementation of the proposed candidate high level actions for severe accident mitigation. The process also includes a diagnostic step that categorizes the core damage event into one of eight states that have implications for both mitigation strategies and emergency preparedness actions. The phases of the CEOG SAMG are summarized below.

Phase 1: SAMG Data Collection and Initial Diagnosis

One of the most important features of the CE generic AMGs is the ability to explicitly diagnose the condition of the plant during a severe accident using available in-plant instrumentation. This allows the TSC to make their recommendations to the control room based on the knowledge of how the accident has evolved and what the current status of the plant is at any time. In Phase 1 of the AMG process, data from the plant is collected and two flowcharts are solved in order to "bin" the accident onto what is referred to as the "plant condition matrix." This matrix approach is a modification of that outlined in the EPRI Technical Basis Report (Reference 27). The CEOG SAMG matrix utilized a 2 X 4 array composed of two RCS damage states and four containment states (Table 5-4). The two RCS designates are called "BD" and "EX" for "badly damaged" and "ex vessel" respectively. In addition, the RCS condition can also be diagnosed as being "Not In A Severe Accident." The four containment designates are "CC" for closed and cooled, "CH" for challenged, "I" for impaired and "B" for bypassed. These matrix locations are illustrated in Table 5-4. The two flowcharts attempt to solve the plant condition for the RCS and the containment respectively. Although the Phase 1 process considers only a very limited amount of "in-plant" data to reach an initial diagnosis, (See Table 5-5) it is sufficiently robust in many instances to yield a solution that requires no further verification.

Phase 2: Verification of Initial Diagnosis

Phase 2 is a detailed verification of the Phase 1 diagnosis. This phase is optional and entry into Phase 2 depends on the level of confidence associated with the Phase 1 diagnosis. Phase 2 differs from Phase 1 in that more plant information (obtained from in-plant data sources and without PASS sampling) is formally factored into the diagnosis phase and is therefore better able to pick up more subtle diagnosis issues for plant state categorization and matrix location selection.

Phase 3: Candidate High Level Action Implementation and Assessment

Phase 3 provides actions to the control room and includes guidance as to how to concurrently assess the effects of those actions on the overall plant condition. Phase 3 involves the analysis and potential recommendation for implementation of one or more CHLAs for each of the matrix locations as a means for mitigating the severe accident. For any CHLA, the available information in Phase 3 includes such topics as initiation criteria, termination criteria, benefits of actions, associated cautions, equipment requirements, etc. Typically, Phase 3 also identifies calculational aids and Selected Technical Issue Report results to assist in determining the efficacy of implementing the CHLA and post-CHLA tracking information. Based on the outcome of Phase 3 strategies, the TSC may take any of a number of actions allowed within the AMGs, including a move to the final location within the AMG process known as "Restorative AMGs."

Restorative Accident Management Guidelines

If, at any time during the AMG process, insufficient information exists to confidently diagnose the matrix location that applies to the severe accident in progress, then the TSC may choose to exit that part of AMGs and rely instead on the section known as "Restorative AMGs". This alternate path is considered to be the preferred path of the AMG process if the TSC is awaiting the restoration of sufficient equipment or instrumentation to facilitate a reliable diagnosis on the plant condition matrix. Whereas the Phase 3 actions have some level of optimization based on the diagnosed plant matrix location, Restorative AMGs provide state independent guidance for the CHLA implementation. Restorative AMGs are focused on saving the next intact radiological barrier. They are also sufficiently fluid to allow the TSC to exit back to the plant condition matrix once a reliable diagnosis can be reached.

The information from Phase 1, Phase 2, Phase 3 and Restorative AMGs together comprise what is known as "CE Generic AMGs." Plant specific versions of the CEOG AMGs have been implemented at all CEOG member utilities.

Figure 5-1: CEOG Severe Accident Management Process

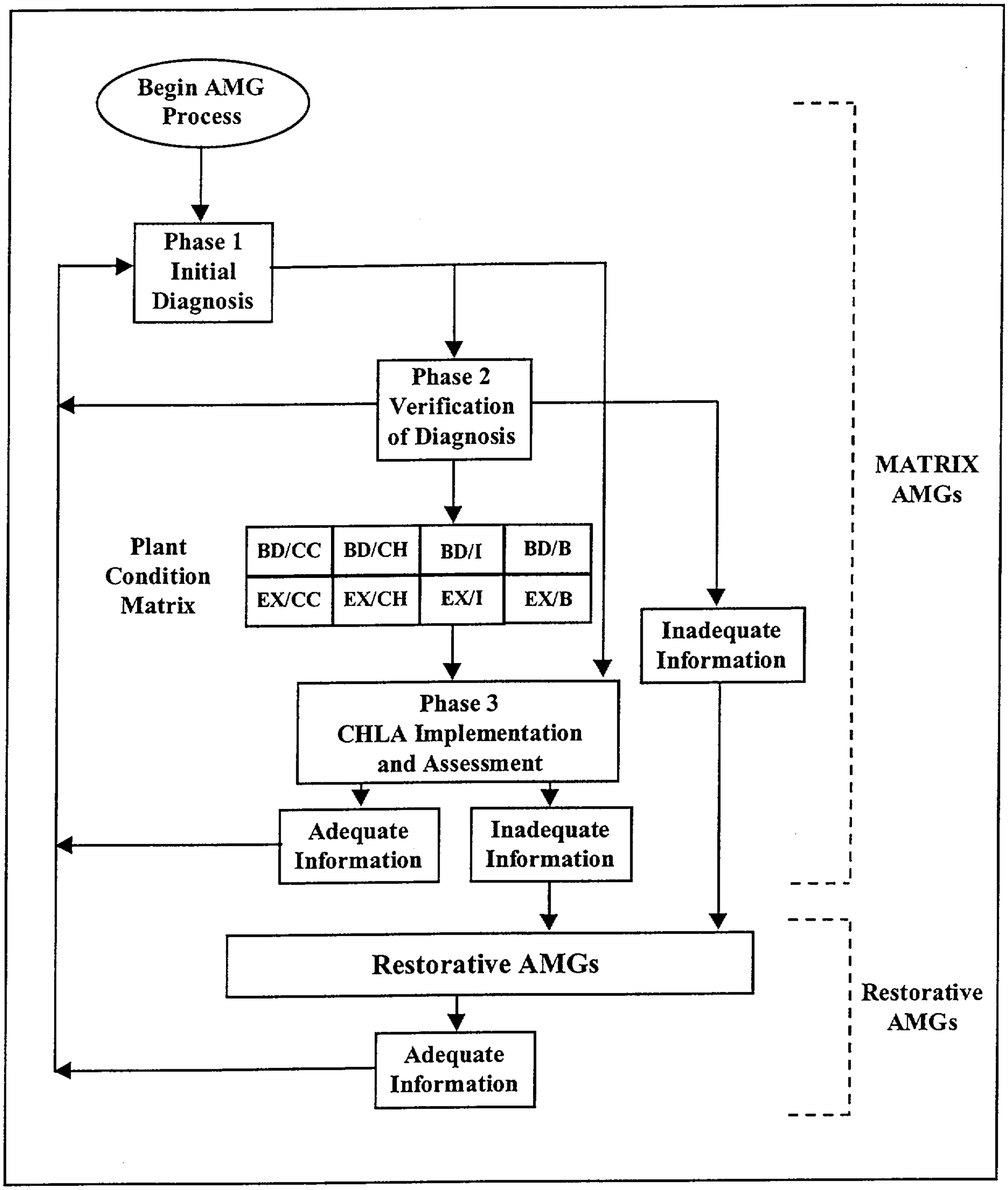

Table 5-3 :

Typical List of CEOG AMG Candidate High Level Actions

1. Inject into RCS
2. Inject into the Steam Generators
3. Depressurize the Steam Generators
4. Restart Reactor Coolant Pumps
5. Vent the RCS
6. Flood Reactor Cavity
7. Depressurize the RCS
8. Spray into Containment
9. Operate Containment Fan Coolers
10. Operate Hydrogen Recombiners
11. Spray outside of Containment
12. Spray the Auxiliary Building
13. Flood the Auxiliary Building
14. Vent Containment

Table 5-4:

CEOG SAMG Plant Condition Matrix

RCS	**Containment**			
	Closed & Cooled (CC)	**Challenged (CH)**	**Impaired (I)**	**Bypassed (B)**
Badly Damaged (BD)	BD/CC	BD/CH	BD/I	BD/B
Core Debris Ex vessel (EX)	EX/CC	EX/CH	EX/I	EX/B

Table 5-5: Typical CEOG SAMG Initial Diagnosis Data Table				
Time Started				
Core exit temperature (°F)[1]				
Vessel water level above core? (Y/N) [1]				
RCS pressure (psia) [1]				
Rapid increase in containment pressure? (Y/N)				
Radiation level in containment – high range CTMT area rad monitors (R/hr) [1]				
Rapid increase in the ex-core power range detector current (pico-amps) (Y/N) [2]				
Was a SGTR diagnosed? (Y/N)				
If a SGTR was diagnosed, is the affected S/G isolated? (Y/N)				
Was a LOCA outside containment diagnosed? (Y/N)				
If there is a LOCA outside containment, has it been isolated? (Y/N)				
Rapid drop in containment pressure? (Y/N)				
If there is a drop in containment pressure is it due to heat removal? (Y/N)				
If there is a drop in containment pressure is it due to controlled or uncontrolled venting? (CV/UV)				
Radiation level outside containment (mR/hr).				
Is containment threatened based on Containment Challenged Calc. Aid? (Y/N)				
Containment pressure (psig).				
Sump Level (ft)				
Containment Hydrogen Concentration (Vol. %)				
LOCA location[3]				
LOCA size (ft^2) [3]				
Safety Injection Flowrate (gpm)				
RWST Level (ft)				
Tail Pipe Temperatures (°F)				
Time Completed				
RCS condition from flowchart				
Containment condition from flowchart				

Notes:
1. Included in ICC instrumentation complement.
2. Ex-core detector to be compared with normal post-shutdown performance.
3. Optional

5.2.2 Evaluation of the Impact of PASS on SAMGS

The SAMGs were intentionally developed with minimal reliance on PASS. This philosophy was largely adopted for the entire industry effort since sufficient non-PASS plant instrumentation is available to provide timely information to the Technical Support Center (TSC) for use in the accident decision making process. Additional reasons for excluding PASS from playing a significant role in the SAMGs decision making process include uncertainties in the PASS measurements due to the delays in 1) obtaining and assessing the sample, 2) difficulties in interpreting sample results due to sample point placement issues, and 3) sample line plugging and iodine plateout. To meet the decision making needs of the SAMGs, the SAMGs must rapidly trend the progression of a core damage event. To this end, the SAMGs rely on RCS and containment instrumentation available to the operations staff. Much of the instruments are within the inadequate core cooling instrumentation complement and are expected to survive the "harsh" environments associated with severe accidents.

5.2.2.1 Identification and Trending of Core Melt Progression

To trend the onset and progression of a core melt event, the SAMGs rely on various instrument sensors within the RCS. These include the Reactor Vessel Level Monitoring System (RVLMS) for trending of voiding in the upper plenum, and CETs, hotleg RTDs and Unheated Junction Thermocouples (UHJTCs) of the RVLMS for trending the core heatup transient. Analyses to support the initial CEOG CDA indicate CETs provide an effective mechanism for following the core uncovery event from incipient core damage to significant levels of fuel pellet overheating. For most domestic CE designs (with the exception of the Palo Verde Units), the CETs are top mounted (inserted from the upper head). This positioning allows operation of these instruments to about 2300 ^{o}F[1]. CET readings in that range correlate to fuel temperatures approaching 3000 ^{o}F.

The RVLMS tracks the local liquid inventory via the temperature difference between a heated and unheated junction thermocouple. When covered with water this device indicates a low temperature difference. However, once uncovered, the heated junction thermocouple will show a rapid increase in temperature relative to the unheated junction thermocouple.

The use of differential thermocouples in the RVLMS provides still additional trending capability. While the primary intent of the RVLMS is to trend approaching uncovery and/or the presence of non-condensables in the upper head, the absolute temperature from the unheated thermocouple junction may be separately monitored to assess gas temperatures in the upper plenum. Since these devices are further removed from the core than the CETs they will survive longer than the CETs. These lower expected temperatures in the upper plenum may be clearly seen by reviewing Tables 5-6 and 5-7. As a result, these instruments can trend core uncovery well into fuel melting. Additionally upon recovery, these sensors may still be operable and trend the recovery process and identify stable coolable conditions.

Core damage may also be indirectly trended via measurements of hydrogen generation and containment radiation. Core relocation within the reactor pressure vessel may be inferred from use of ex-core detectors. Ex-vessel corium relocation may be established using ex-core neutron detectors and containment pressure and temperature. Calculation aides for the use of startup and power range detectors to detect core relocation are incorporated into the CEOG SAMG discussion of technical issues (Reference 15) and plant specific SAMGs.

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[1] The ABB-CE Palo Verde design employs bottom-mounted CETs. The performance capabilities of these CETs are similar to those identified for the CETs installed in the Westinghouse PWRs (See Reference 11).

5.2.2.2 Assessment of Approach to Recriticality and Boron Precipitation

Re-criticality may be a concern following core damage events which require use of unborated water for core cooling, or which involve a large introduction of unborated water to the containment (Component Cooling Water system rupture).

CE and WOG SAMGs allow the introduction of unborated water to cool the core when borated water is unavailable. To better understand the threat associated with the introduction of non-borated water into the core, analyses were performed to support the EPRI Technical Basis Report. (References 23 and 27). This study confirmed that controlled introduction of unborated water into the post-accident reactor core (where the fuel geometry is undeformed) will prevent a return to recriticality. To this end, cautions are provided by the SAMGs to restrict water delivery rates. For delivery rates on the order of ~ 1000 gpm the reactivity addition rate associated with the introduction of clean water is sufficiently slow, such that the boiling feedback mechanism will limit the power generation and prevent further core degradation. In this case, the plant staff is highly aware of the potential for recriticality and existing plant instrumentation is effective in monitoring the potential for a return to criticality.

As the core begins to deform, its ability to reach criticality is diminished, and with sufficient loss of geometry, the addition of unborated water will not impact power production. As discussed in the EPRI Severe Accident Management Technical Basis Report (Reference 23, Appendix B), continued degradation of the core will not result in re-criticality due to both the compaction of the core material and the boron buildup in the core debris. Thus, the need for boron sampling in severe core damage scenarios resulting in core material relocation is minimal.

Boric acid precipitation may become possible following large cold leg LOCAs which cannot implement hot side/cold side safety injection strategies in a timely fashion. Hot leg/cold leg post-LOCA injection guidance is well proceduralized and times to implement boration strategies are sufficiently long that appropriate piping alignments may be established well in advance of the core reaching the boric acid precipitation point. For the cold leg LOCAs of concern, CE plants have operational hotleg injection pathways. If boric acid precipitation is suspected due to an inability to implement hot side injection, lower boration (or unborated) water may be introduced into the RCS. Recriticality is highly unlikely; however, it may be tracked via startup range monitors. This situation would likely place the operator in the Functional Recovery Guidelines. It should be noted that boron sampling would not necessarily produce desired results since the sample occurs in the hotleg which will not be representative of the core condition. This situation occurred at TMI-2 when hotleg sampling appeared to the operators to indicate a poorly borated core with a potential criticality concern, while in reality boron was concentrating in the core.

5.2.3 Monitoring of Containment Hydrogen

5.2.3.1 Basis for SAMG Containment Hydrogen Monitoring

Following a severe accident hydrogen is monitored in the containment, to both (1) help establish the degree of core degradation, and (2) assess the potential for a containment threat due to hydrogen combustion. Containment hydrogen monitoring to establish core damage is part of the CEOG plant CDA methodology. For accidents that do not bypass the containment (or do not exhibit significant containment loss of isolation) and have large breaks in the primary system, (assuming the absence of prior burns) the hydrogen in containment roughly correlates to the level of core-wide zircaloy cladding oxidation.

Oxidation of cladding occurs at elevated core temperatures (in excess of 1800 ^{O}F). In the CE CDA methods, core-wide oxidation, CET readings and fuel rod failures are related as shown in Figures 5-2a and 5-2b.

The second use of hydrogen monitoring is to establish the existence of a potential containment threat. The combustion of large quantities of accumulated hydrogen in the containment (generally in excess of 75% reacted zircaloy cladding) may raise containment pressures to levels where the integrity of the containment (particularly penetrations and other containment weak points) may be threatened. By having knowledge of the global hydrogen accumulation, operators may take actions to control containment hydrogen levels via controlled containment cooldowns combined with venting, or by attempting to ignite the hydrogen under conditions less favorable to combustion (e.g., wet atmosphere, lower hydrogen concentration).

5.2.3.2 Evaluation of Hydrogen Monitoring within SAMGs

Based on the discussion in 5.3.3 it is recommended that the ability to monitor the containment atmosphere for hydrogen be retained as a NUREG-0737 requirement. However, this capability is currently being performed by redundant safety Class 1E containment hydrogen monitors which are not part of the PASS system. Approval for this option was already granted by the NRC in their SER on CEN-415.

5.2.4 Monitoring of RCS and Containment Activity

SAMGs do not include specific reference to monitoring RCS activity (and in particular, RCS radioisotopes). Containment activity is monitored in the SAMGs as a means of assessing the level of fission product releases to the containment and in assessing the effectiveness of fission product removal mechanisms. Containment monitoring is accomplished via redundant Class 1E Area Radiation Monitors . The CDAM has approximately correlated the ARM readings with various levels of fuel damage (See for example Figure 5-3).

SAMGs also monitor system vents (stacks) and bypasses. Monitoring of radioactivity releases may be established via fixed in plant equipment or offsite doses.

5.2.5 Conclusions

The initial intention of the PASS was to provide information to the plant staff for use in trending and mitigating severe core damage events. In practice, limitations inherent in a post accident sampling process, coupled with increased availability of equipment and information, have resulted in a more restrictive use of the PASS and the development of SAMGs without reliance on PASS. Therefore, the elimination of the PASS sampling has no impact on the ability of the SAMGs to perform its function and adversely impact accident management.

5.3 Use of PASS within the CEOG Plant Emergency Plans and Core Damage Assessment

The NRC conducted a review of emergency plans from several US utilities from the perspective of assessing the role of the PASS in their program. This work was reported in Reference 6. The reference indicated that provisions for the use of the PASS are integrated into these procedures and the plant CDA Methodology (CDAM). However, the review also concluded that PASS information would be expected to lag other indicators of plant status that drive the emergency class declarations and formulation of PARs. It was further concluded that adequate core damage assessment can be made without reliance on a

PASS. Thus, while the PASS was integrated in the site emergency plans, elimination of PASS sample requirements would not significantly affect the plans' effectiveness.

This section reviews and evaluates the present use of the PASS in CEOG member utility site emergency plans. Where appropriate, recommendations are made to eliminate PASS samples or to provide equivalent information via alternate methods.

5.3.1 Use of PASS in the Declaration of Emergency Classes

5.3.1.1 Background: Emergency Classifications

10CFR50.47 requires that as part of the site emergency plan that the plant be able to identify appropriate EALs (in order to classify emergencies), establish PARs, and trend and project radiation releases (See Reference 17). Plants have included a number of processes for meeting these objectives. EALs are typically based on parameters affecting fuel state, RCS state and containment state. 10CFR50 Appendix E, Section C, "Activation of Emergency Organization," states that EALs are to be based on:

1. Site radiation monitoring information,
2. Offsite radiation monitoring information, and
3. Readings from plant sensors capable of indicating a potential emergency (such as containment pressure and relevant ECCS data).

The emergency classes shall include:

1. Notification of Unusual Event (NOUE),
2. Alert,
3. Site Area Emergency and
4. General Emergency.

In practice, CEOG emergency plans do not rely on EAL determination based on PASS samples. This is due to the longer time it would take to make a proper assessment using the PASS and issues associated with interpretation of PASS samples. EALs are typically based on the following parameters as appropriate and available:

1. Core exit thermocouple readings
2. RVLMS readings
3. Containment hydrogen concentration
4. Gross RCS activity
5. Radiation readings from Area Radiation Monitors
6. Off site doses (based on projection or site survey)
7. Containment pressure
8. Event related diagnostics and associated barrier failure assessment
9. Occurrence of observable events (fire, seismic, etc.)
10. RCS iodine concentration

Figure 5-2a: % of Fuel Rods with Ruptured Clad vs Core Clad Oxidation

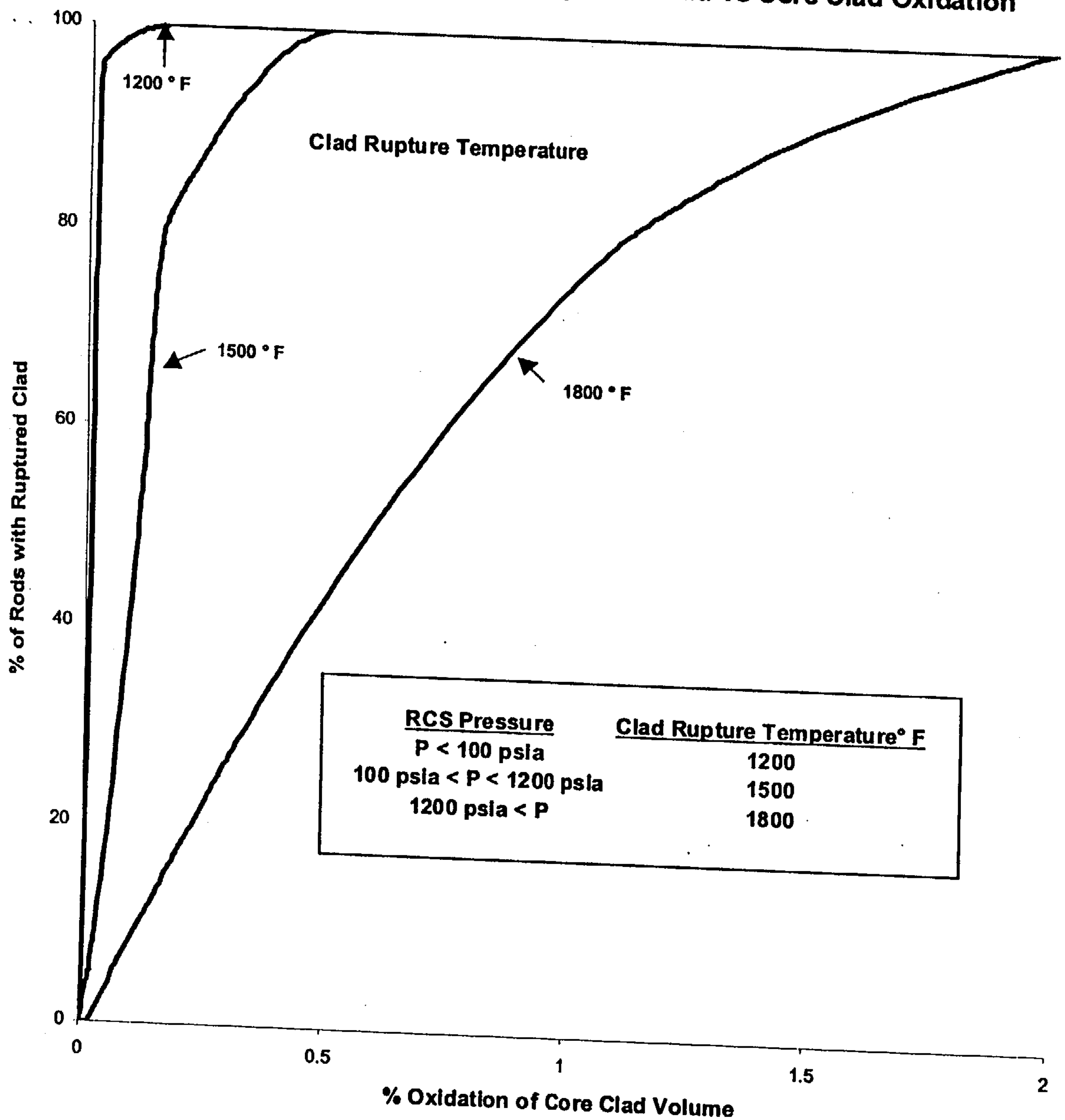


FIGURE 5-2b
Percent of Fuel Rods with Ruptured Clad vs. Maximum Core Exit Thermocouple Temperature

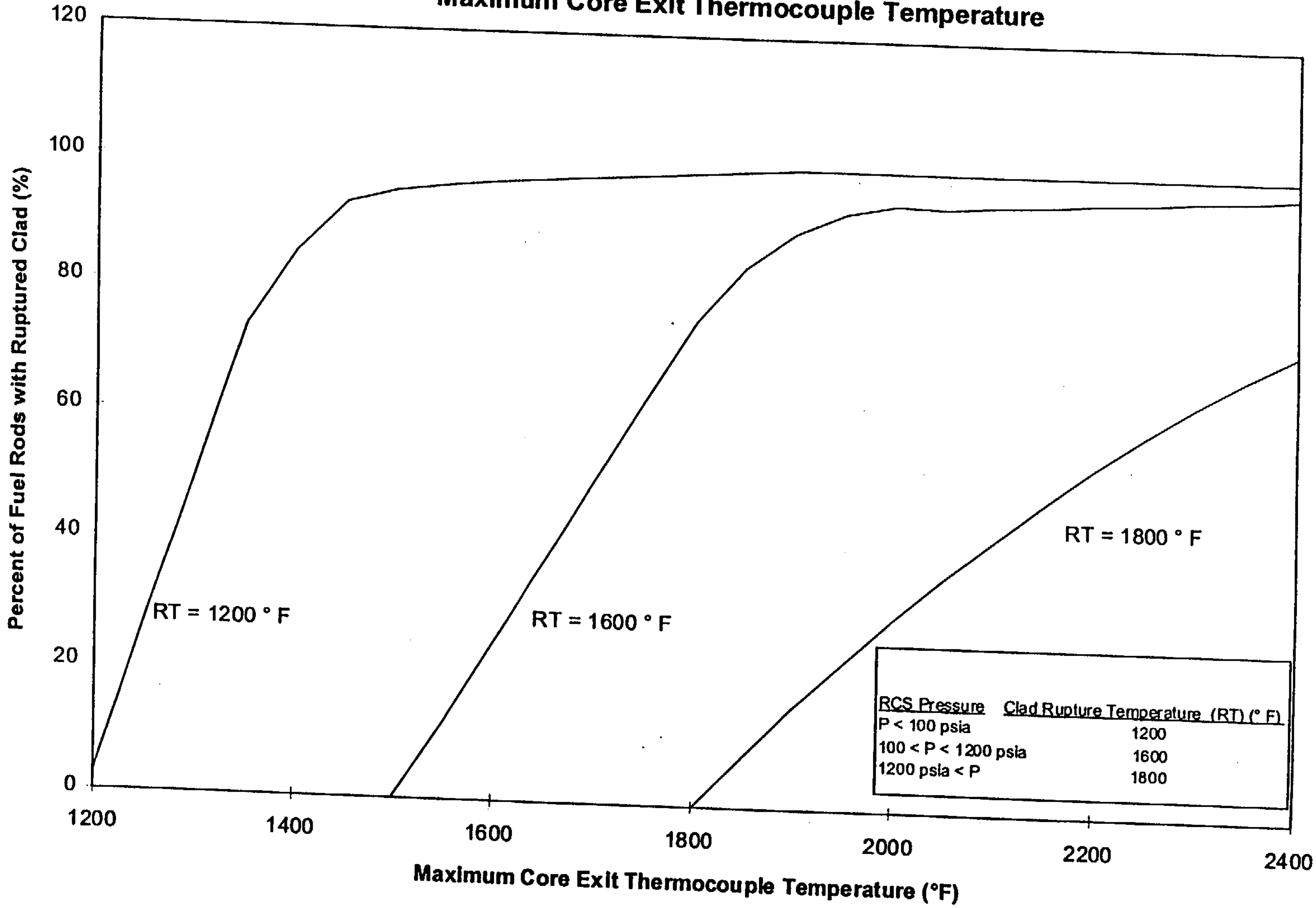
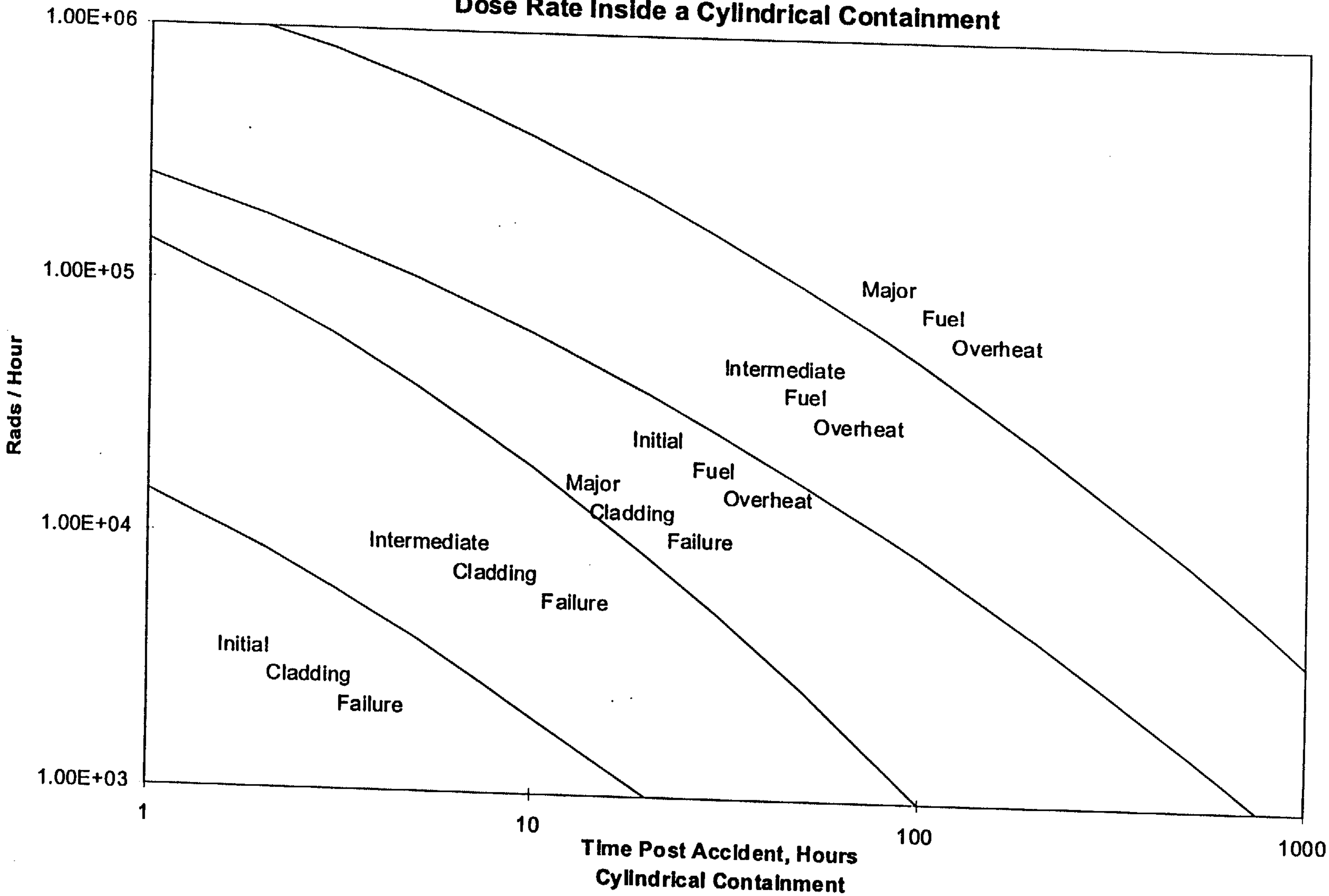


FIGURE 5-3
Typical Analysis for Post Accident Dose Rate Inside a Cylindrical Containment



of core damage. The current CEOG methodology for CDA involves four separate overlapping calculations. These methods were provided in CEOG task 467 as reported in CE-NPSD-241 and were later adopted by most CEOG member utilities into their Emergency Plan Implementing Procedures. Some of these methods use PASS samples, others use information available to the operator from the Reactor Vessel Level Monitoring System (RVLMS) and Core Exit Thermocouples (CETs). Generic licensing basis accidents will be reviewed and the appropriate changes to the CDA methodology will be recommended to support the elimination of PASS.

The emergency classes identified by the NRC in Reference 17:

Notification of an Unusual Event (NOUE):
These events are characterized by high reactor coolant activity exceeding technical specifications indicating fuel damage.

Alert:
Alerts are declared for events that involve potentially substantial degradation of the level of safety of the plant. Events leading to Alert designations are expected to result in releases which are limited to small fractions of the Environmental Protection Agency (EPA) Protective Action Guideline (PAG) exposure. Alerts include events that are characterized by high reactor coolant sample activity (> 300 μCi/cc dose equivalent of I^{131}), indicating loss of fuel cladding in 2-5% of the fuel rods.

Site Area Emergency (SAE):
SAE is declared for events that involve or are expected to involve major failures of plant functions needed for protection of the public. Releases are not expected to exceed EPA PAG exposure levels beyond the site boundary.

General Emergency:
These events involve substantial core degradation or melting with a potential for loss of containment integrity. Releases are expected to exceed EPA PAG exposure levels.

According to the NRC Response Technical Manual (RTM-96, Reference 20) the rationale for NOUE and Alert classifications is to provide early and prompt notification of minor events which could lead to more serious consequences given operator error or equipment failure. Appendix 1 of NUREG-0654 indicates 17 example initiating conditions for an NOUE and 20 conditions for an Alert. The Alert designation is intended to capture such events as SGTRs with concurrent loss of offsite power, steam line breaks with significant (> 10 gpm) primary to secondary side leakage, and reactor coolant pump seizure. Conservative design basis assumptions result in estimates of clad damage associated with these events to be in the 2-5% range. Best estimate evaluation of these transients suggest that the expected levels of clad damage will be less than 1% and consequent RCS coolant activity will be less than 175 μCi/gm I^{131} for a typical 2700 MWt PWR.

SAEs encompass accidents which involve fuel overheating with significant off site exposure. According to Reference 21 these events are expected to involve core damage with 20% of the fuel pins failed. Typical determinants for SAEs include potential loss of coolable geometry as indicated by inadequate core cooling. These events will be characterized by an extended period of high CET temperatures. The GE classification involves actual or imminent core melting with a potential loss of containment integrity. GEs may be declared based on excessive off-site doses, loss of two of three fission product barriers with a potential loss of the third, and core damage sequences noted by the Probabilistic Safety Assessment (PSA) to potentially result in both core damage and containment failure.

5.3.1.2 Evaluation of CEOG member EAL classifications

For the purpose of declaring emergency classes, CEOG plants typically rely on CETs, ARMs and containment hydrogen data, site radiation levels and/or event specific information. Of the many emergency classifications performed at the plant, the lower tier emergency classes (NOUE and Alert) are the most likely to involve sampling of the reactor coolant. A summary of the PASS-related aspects of CEOG member EAL criteria are presented in Appendix C. A review of the table indicates that NOUEs typically require a sampling of the I^{131} content in the reactor coolant to ensure the levels are low. For most CE PWRs I^{131} equivalent concentrations below 60 μCi/gm may be assessed by use of the Normal Sample System. A limited number of Alert indications are based on the NRC 5% total clad failure iodine equivalent value of 300 μCi/gm. These fuel failure levels are representative of the high end of clad damage associated with departure from nucleate boiling induced fuel damage transients. Due to delays in collecting this sample (typically > 1 hr) even for automated PASS systems, the Alert designation will likely be made based on the event specific diagnostics (symptoms) and not the clad failure limit.

A sampling of CE PWRs confirms the earlier NRC understanding of PASS usage. Emergency planning accepts PASS input primarily for purposes of confirming decisions and information associated with offsite radiological releases for a spectrum of in-containment severe accidents. This structure has emerged as an outgrowth of the industry response to post TMI-2 NRC requirements, rather than based on a technical need for the information. Specifically, necessary emergency class declarations can be readily defined without resort to use of PASS. As discussed above, the only potential EAL where a PASS is sometimes presently used involves the use of I^{131} RCS coolant samples at low fuel damage levels to differentiate between an NOUE or Alert. As discussed above, the PASS usage in these categories arises from early NRC recommendations to declare an Alert at 300 μCi/gm I^{131} equivalent (about 2-5% failed fuel). Alternatively, a severe loss of fuel cladding may also be assessed via use of failed fuel monitors (See RTM-96). Another alternative effective approach to declaring an Alert would be to establish dose based on the distance from a sample source or sample piping. For example in the Millstone Unit 2 EAL classification scheme, the Alert level is established at a dose rate ≥14.6 mr/hr/ml one foot from an unpressurized RCS sample. In any event, there is low likelihood that PASS samples would be used for declaring an emergency class due to timing issues associated with taking a sample (issues associated with sample line flushing) and the fact that alternative information for use in declaring an Alert is more readily available.

Other indications, not requiring use of PASS information would be available for declarations of SAE and GEs. These plant conditions may be established via the same level of information that is available in the SAMGs. For example the same ICC indicators that would lead one to initially classify an event as BD/CC would also result in an SAE classification. Similarly, containment matrix locations CH, I or B indicate the existing or potential loss of the containment and consequently the event would likely be escalated from an SAE to a GE.

5.3.2 Development of Protective Action Recommendations

5.3.2.1 Basis for Offsite Protective Action Recommendations

NRC guidance on offsite PARs may be found in Supplement 3 to NUREG-0654. Most CEOG PAR programs closely follow the NRC guidance. PARs are required to be prepared upon the declaration of a General Emergency. In establishing early phase PARs the plant staff must primarily be aware of the present and projected status of the core, containment, and the site meteorology. Typically the PARs are based on:

- Degree of core damage.
- Level of radioactivity released into the containment.
- Timing and extent of containment failure or bypass.
- Stack and/or off site doses (radiation surveys/projections).

The degree of core damage is indicative of the extent of volatile fission products released from the fuel. Typically a fully overheated core (just prior to core melt) will release all the noble gases contained in the fuel rod and fuel pellets along with 25% or more of the inventory of iodines and cesiums (See Reference 28). These levels of core damage may be readily established via in-plant ICC instrumentation (CETs). Extended core uncovery will degrade an overheated core to a "molten" core in a matter of minutes. As the fuel matrix temperature exceeds ~ 3000 ^{o}F fuel pellet releases of iodine and cesium rapidly accelerate. Within an uncovery period of 2 to 5 minutes I and Cs releases will rise to >80% of initial inventory. For in containment release scenarios this information may be confirmed via ARM readings and hydrogen concentrations. For bypass conditions, monitors at plant vents and field measurements will rapidly indicate high radiation levels.

Protective Action Assessments

PARs are divided into two categories (See for example RTM-96): initial (early phase) PARs and intermediate phase PARs. The early phase protective action assessment will take from several hours after the declaration of a General Emergency to up to about 4 days. It is the objective of the early phase PARs to:

- Prevent early health effects.
- Reduce the risk of delayed health effects.

Intermediate phase PARs initiate in the period beginning after the accident releases have been brought under control and reliable environmental measurements are available, until the protective actions are terminated. This phase may last from many weeks to months. Exposure during the intermediate phase is due to exposure to deposited material and ingestion. Intermediate PARs are established based primarily on the stability of the plant conditions and field measurements. Initial and intermediate phase PARs may be established without direct reliance on PASS.

5.3.2.2 Evaluation of the Impact of PASS Elimination on PAR Determination

A review of CEOG member PAR programs indicate that PARs are established via a combination of plant conditions (as determined via the SAMG diagnosis process), and dose projections and/or field measurements. The PASS was generally not strongly integrated into the PAR decision process since in the early time frame PASS samples would typically be unavailable and unreliable (due to the transient nature of the event). The plant specific integration of the PASS into the post accident dose assessment methodology is summarized in Appendix D.

As the event progresses, the containment and RCS parameters will stabilize. In this time frame additional actions are taken based on actual and projected plant radiological releases. In all instances CEOG PARs are established based on (1) extrapolation of ARM results, or (2) use of site radiological surveys. Under ideal circumstances PASS containment samples (when available) may be used to help confirm PARs for conditions when the containment is closed and cooled with a small leak. These conditions are typically of low consequence. In practice PASS sampling could confirm noble gas releases; however, issues with plateout, sensor location, and timing make them difficult for use in estimating airborne iodine concentrations. This was the basis for eliminating heat tracing containment air sample lines. However, containment ARMs perform a similar function. Also, PASS samples **will not** provide useful guidance to the plant if (1) containment integrity is threatened (since containment failure would revolitalize radionuclides in the sump), (2) the event is a bypass or SGTR or (3) the event involves a Station Blackout.

Following a severe accident the plant staff is required to provide PARs to the State and local officials. While there are provisions for the use of PASS information in this process, as a rule, CE plants use PASS-independent schemes for establishing both initial and follow-up PARs. Since PARs must be issued within 15 minutes of the declaration of a General Emergency, PASS information will not be considered in the formulation of the initial and early PARs. Current emergency plans allow provisions for use of PASS in longer term PAR development; however, the PASS input is primarily confirmatory and is primarily focused on the projected dose assessment. For many plants, projected dose assessments are also made based on knowledge of the containment integrity, availability of plant fission product removal systems and confirmatory ARM or site area radiation surveys. Both NUREG-0737 and Reg. Guide 1.97 requires having monitoring and sampling capability for all identified plant release points. Additionally, for some plants (such as ANO-2) the site utilizes sophisticated Super Particulate-Iodine-Noble Gas (SPING) monitors to assess actual releases of noble gases, iodines and particulates to the environment. This information, along with meteorological data, is factored into the development of long term PARs. It should be further noted that while the PARs are provided to the State agencies, the actual actions associated with long term emergency plans, evacuation and sheltering may be adjusted/modified based on factors other than the NRC dose criteria. Similarly the decision to distribute potassium iodide pills to the population is not based on PASS readings but rather a full assessment of the event coupled with government policy issues. In all instances, sufficient information is available via existing instrumentation for the development of PARs in the absence of PASS.

Comments Regarding Distribution of Stable Iodine

Severe accidents with considerable core damage (core melting) and either a failed, unisolated or bypassed containment may release potentially large quantities of iodines into the environment. RTM-96 notes that distribution of stable iodines should be considered at a projected iodine inhaled dose of 25 rem. Based on a plant survey, distribution of stable iodine varies. On one site, staff would be administrated stable iodine if an exposure of 10 rem or more is expected based on air samples. Stable iodines include salts such as potassium iodide (KI). Ingestion of KI tablets would result in a stable form of iodine being absorbed by the thyroid. The net consequence of this action is to significantly reduce the absorbed radioactive iodine dose. Since precise projections of inhaled iodine doses are not possible, estimates based on core damage state, containment state vent releases, site radiation levels and/or offsite radiation surveys will integrate into the decision process as will State policy. As discussed previously, even though PASS information is intended to provide an assessment of airborne iodine releases the data is very limited in that (1) the most severe potential release pathways are not monitored within the PASS (i.e. bypass events), (2) flashing during the containment failure process may re-suspend iodine that has already been washed into the sump water, and (3) limitations due to sampling location and iodine plateout in sampling lines may depress measured levels. Additional discussion on PASS limitations is presented in the following section.

Limitations on the use of PASS in Monitoring Airborne Radionuclides

The role of the PASS defined above has as its premise that the PASS inputs will be of a better quality than other observations; therefore, its input is useful in refining a core damage assessment. This is not entirely true, and in some instances over-reliance on PASS information could result in inappropriate PARs. The limitations of the PASS arise as a direct consequence of the fact that the PASS system provides remote, point specimen samples of reactor coolant and containment atmosphere. The remoteness of the sample and the selected sample point may clearly impact the measurement. For example, it is alleged that use of the PASS will provide a good indication of the airborne iodine content in the containment. Based on the current knowledge at the time, the PASS sample lines were designed to transport elemental iodine (gaseous). However, recent research (Reference 29) indicates that iodine in the containment will be primarily in the form of cesium iodide (CsI) particulates. Transport of this material in the sampling lines will likely result in considerable iodine plateout/settling prior to reaching the sensor. Since CsI has a high revolitialization temperature (~ 800 ^{o}F - 1200 ^{o}F), revaporization of CsI is unlikely regardless of sample line heat tracing. Thus, the PASS may significantly underestimate the iodine airborne content. Another example arises when one samples the containment atmosphere just prior to a depressurization. The static PASS sample may suggest low iodine content in the containment. However, taken out of context, in a situation where a containment failure will occur, the iodine airborne sample (even if accurate) would be misleading as iodine trapped in the containment sump will revoltialize.

Finally, it should be pointed out that many very serious events have no potential for PASS capability, e.g. station blackouts, SGTRs and Interfacing System LOCAs (ISLOCAs). Among these events are the ones that have the greatest public risk (ISLOCAs and SGTRs with stuck open secondary valves). Therefore, PARs must be developed for these sequences as well. In these instances, PARs are primarily focused on the use of non-PASS information and knowledge of accident progression. Consequently, removal of the PASS from the emergency plan implementing procedures would have a neutral impact on emergency preparedness.

5.3.3 Core Damage Assessment (CDA)

The CDA is generally integrated within the PAR decision process, and in some plants, the EALs. The CDA is an outgrowth of the accident at TMI-2 and is focused at quantifying the extent of core damage following a serious accident at a commercial nuclear power plant. This section discusses the role of the CDA in the various aspects of accident management and emergency responses. Prior to discussing the CDA, a background discussion of the characterization of core damage events is presented.

5.3.3.1 Characterization of Core Damage Events

This section focuses on accidents in which some level of damage has occurred to the reactor fuel. This damage may vary from localized failure of fuel cladding and progress upwards to gross core melt and core relocation.

Specifically, this section of the report summarizes the current understanding of core damage accidents as it relates to the potential uses for post accident sampling of the RCS coolant and containment atmosphere. The overall core behavior during a core damage accident is briefly discussed, followed by a summary of the post accident fission product behavior. It is the objective of this section to define the underlying process associated with the progression of core damage accidents in order to clarify the roles of "in-plant" instruments in trending and mitigating the event. The following information represents the current knowledge and understanding of the core, important chemical species and fission product behavior during and following a core damage accident.

5.3.3.2 Core Behavior during an Accident

Fuel rods in a typical PWR are constructed of a zircaloy tube (called fuel rod cladding) with caps on each end. The fuel rods contain uranium dioxide fuel pellets. The gas space inside the fuel rod cladding is initially pressurized. During normal operation, fissioning of the uranium results in the migration of some of the fission products, notably xenon, krypton and to a lesser extent iodine and cesium, into the gap space. The fuel rod cladding serves to contain radioactive fission products, thereby preventing their release into plant systems or the atmosphere.

There are three broad accident classes that can occur in a PWR, these are:

1. Accidents without cladding damage but with elevated coolant activity levels resulting from iodine spiking.

2. Accidents with a limited fuel rod cladding rupture but without high indicated fuel temperatures, and

3. Accidents with a significant loss of fuel rod integrity.

Characteristics of these categories of accidents are described below from the perspective of the impact of the event on the plant and the ability of typical in-plant system instrumentation and procedures (non-PASS) to diagnose, categorize and respond to the event.

5.3.3.2.1 Accidents without Cladding Damage

This accident class describes the core state following an accident with the fuel rod cladding intact but with minor defects that can lead to higher than normal reactor coolant activity, especially due to an iodine spike at reactor trip (Reference 25). In this case, the in-plant emergency response would be according to the plant Emergency Operating Procedures (EOPs). Additionally, the EAL classification scheme may indicate a Notification of Unusual Event. Since there is no potential loss of the fuel clad barrier there is no reason to escalate the offsite response level based on the core state. These accidents can be discriminated from fuel failure events using the NNSS to measure the low level of radioactivity in the coolant via direct measurement of I^{131} or via indirect dose measurements of sample activity.

If reactor coolant has been released to the containment, the containment atmosphere may contain some airborne fission products that were present in the reactor coolant system. Typical readings experienced by containment ARMs during these events could range from a containment radiation increases of <1 R/hr to ~ 10 R/hr (See for example Reference 11).

5.3.3.2.2 Accidents with limited Fuel Rod Cladding Rupture

The second class of accident describes a core state following an accident in which fuel rod cladding may be damaged, but the reactor core remains covered with water. Events of this nature may include RCP seized rotor transients and reactivity excursion events such as a main steamline break and CEA ejection. In this case there would be a release of fission products from the fuel rods to the reactor coolant system. The in-plant emergency response would be according to the plant EOPs. Best estimate analyses of these design basis events typically predicts very limited failures of fuel cladding (<1% of fuel pins). For these events the potential coolant activity would be in the vicinity of 100-400 μCi/gm. In consort with "in containment" environmental sensors (pressure, temperature, hydrogen and radiation) and offsite radiation surveys, the necessary information exists with the operating staff to respond to and classify the severity of the event. Events in this category may be classified as an NOUE, or if the cladding failures are believed to have exceeded several percent fuel cladding failures, an Alert. If reactor coolant has been released to the containment, the containment atmosphere may contain airborne fission products that have been released from the fuel rods when they were damaged. Typical radiation levels would range from 50 R/hr to about 200 R/hr.

5.3.3.2.3 Accidents with Significant Loss of Fuel Rod Integrity

The last, and most severe, accident class describes a core state following an accident in which there are indications that the core temperature was in the range where inadequate core cooling could be diagnosed. The status of the core in this accident class has greater variability than in the previous cases. Initial in-plant emergency response to these events is based on plant EOPs. However, when prolonged high core temperatures are anticipated, the plant operating staff would also begin to implement the SAMGs.

During these events the plant operating staff relies on the ICC "in plant" instrument suite including CET, RVLMS (and its associated HJTCs), hotleg resistance temperature detectors and pressurizer pressure sensors to trend the event. Indications of ICC is considered in the EAL classification scheme as a potential loss of the fuel clad barrier; therefore, there is reason to escalate the level of offsite response based on the core state. ICC indications will occur early in events leading to significant core damage. In addition, there would be an urgent need for information for supporting the offsite emergency response activities according to the plant emergency plan. The activities that directly support the offsite emergency response would be the core damage assessment, the EAL classification, and the offsite dose projection activities. Compared to the preceding cases, there is a much higher urgency for information concerning the plant status since the gross fission product levels in plant systems are much higher than anticipated. It should be noted that during these events, plant actions are strongly influenced by SAMGs, which in turn are guided by available in-plant instrumentation.

The actual state of the core following an event characterized by ICC, can vary from a core with an essentially intact geometry, to one which has undergone massive disruption and relocation (including transport of corium material ex-vessel). The severity of the core degradation is largely based on the duration and depth of the core uncovery. Deep core uncoveries for a time period of ~ 5 minutes will likely lead to a core melt condition. Plant recovery for these various modes will also be different. In this third accident class, the overall expected core behavior during the accident is dictated by the fuel rod cladding temperature and fuel pellet temperatures of individual fuel rods in the core. A summary of the fuel degradation behavior, taken from NUREG-1228 (See Reference 30), is provided in Table 5-8.

Core Geometry Intact

In this condition, the fuel rods have experienced overheating and possibly some localized melting but the pre-accident fuel rod geometry remains. The extent of clad damage can vary dependent on the accident scenarios. Events that result in rapid reductions in RCS pressure and exceed cladding temperatures of 1200 ^{o}F, have the potential for cladding failure on some rods. The extent of clad failure increases as cladding temperatures increase. Once clad temperatures exceed 1800 ^{o}F, global cladding failure is expected for most transients. The progression of these events may be trended via CETs. Examples of events that could cause localized cladding failures include small LOCAs or a large LOCA, where one or more trains of the emergency core cooling system operates (i.e. up to a design basis LOCA). These events can result in temperatures in the core which are only above the normal operating temperatures for a short period of time. These events result in the heatup of the fuel pellets as heat removal is severely limited when the upper portion of the fuel rods are uncovered, followed by a cooldown as the cold safety injection water recovers the fuel rods. However, the combination of these temperatures, which also results in an increased pressure differential across the fuel rod cladding, and mechanical stresses in the fuel rod can result in localized failures of the fuel rod cladding. Events in this category will release much of the fission products contained in the fuel rod gaps and plena. However, releases from the fuel pellet matrix will be very low.

Core Geometry Lost but Core Material Remains In vessel

In this condition, the fuel rods have experienced significant overheating and the fuel pellets and fuel rod cladding have melted and relocated downward in the core region and eventually into the reactor vessel bottom head, but the reactor vessel is still intact (e.g. the TMI-2 accident). These events are characterized by prolonged core uncovery, CETs near or above their operational limits. The decay heat in the fuel pellets cannot be effectively removed due to the lack of coolant on the outer surface of the fuel rods. As a result, the fuel pellet and fuel rod cladding continues to heatup until the temperature of the fuel rod cladding is great enough that the zirconium cladding begins to react with the steam surrounding the outer surface of the fuel rod cladding. The reaction of the zirconium cladding and the steam in the reactor coolant system is exothermic (the reaction produces additional heat) and produces hydrogen. The exothermic nature of the reaction rapidly oxidizes residual zircaloy clad and escalates the fuel rod temperature. At 2500 ^{o}F the heat addition due to Zr-water reactions is ~ twice that of the decay heat.

If cooling is not quickly restored to the overheating core, the loss of fuel rod integrity will progress to a state in which the original core geometry is substantially changed as a result of melting and downward relocation of the core and CEA. Once the local fuel rod temperatures exceed about 2600 ^{o}F, the control rod cladding will begin to melt and relocate downward in the core. When the control rod cladding material melts or is ruptured, the control rod materials also relocate downward. The SAMGs suggest potential trending of the re-location process via observation of the performance of the ex-core detectors. Anomalous trends in the ex-cores (above that sensed during normal shutdown conditions) can provide insight into the core relocation process.

Core Ex-vessel

In this condition, most of the fuel rods in the core have melted, the reactor vessel bottom head has failed due to contact with the molten core material that relocated to the bottom head, and the molten core debris has drained from the reactor vessel into the containment. Migration of the core material potentially outside of the reactor vessel may be sensed by trends in the ex-core detectors coupled with observable thermodynamic responses in the containment (e.g. pressure spikes).

5.3.3.3 CEOG Core Damage Assessment Methodology

The present CEOG core damage assessment methodology employs four overlapping calculational procedures. These procedures are summarized in Reference 16 and included in procedure guidelines for assessment of core damage utilizing:

1. Core Exit Thermocouples,
2. Containment radiation dose rates,
3. Hydrogen and,
4. Radiological analysis of samples.

The first and second methods are based entirely on ICC plant instrumentation and have no reliance on PASS. The third method may be largely accomplished via use of Class 1E containment hydrogen monitors. While containment hydrogen is generally sufficient to establish this assessment for LOCAs with significant core damage events, the CEOG methodology also uses estimates of trapped RCS hydrogen and an integrated "fine tuning" adjustment associated with the consideration of dissolved hydrogen in the RCS coolant. Integrating dissolved hydrogen measurement into this CDA scheme complicates the assessment procedure, and slows down the assessment process with no corresponding benefit. Other non-CEOG methods have been developed (See for example Reference 11) which utilize hydrogen as part of the CDA and restrict the assessment to be based entirely on the containment hydrogen measurement. The last of the CDA modules (Radioisotopic Analysis) relies entirely on input from the PASS. This last method provides the potential for some additional resolution regarding the severity of the most serious accidents (degree of fuel melting). However, the ability to make a meaningful assessment may be compromised due to limitations inherent in the sampling process (e.g. sample point restrictions and plateout of fission products in the sample line). In any event, deletion of the last CDA Methodology module (radiological analysis of samples) will not adversely impact the ability of the plant to diagnose, respond to and mitigate the consequences of a severe accident.

It should be noted that very low core damage levels (less than a few percent clad failure) may occur without any overt indication by the CETs or the hydrogen monitors. These events typically involve transients such as CEA ejection, main steam line breaks and transients involving a seized rotor. Fuel damage may be limited and localized. The EALs would likely be established based on observed characteristics associated with the initiating plant event. Radiation levels in containment may increase due to the release of radionuclides into the reactor coolant system. This increase may be used as an estimate of core damage. Since the fuel damage is typically low for these events (between 0.1% fuel failure and 5% fuel failure) the CDA may also be accomplished using normal plant systems, for example, the normal sampling system with appropriate delays for nuclide decay and/or the plant's failed fuel or letdown radiation monitors. This is particularly true when one considers the lower potential source terms and realistic expected range of fuel failures. The current WOG document considers these failures to be sufficiently low so that quantification is unnecessary. In any event, alternatives exist to the use of the PASS for these low probability, low consequence events.

5.3.3.4 Recommended Changes to the CEOG CDAM

Based on the above discussion it is recommended that the CEOG CDAM be changed in two ways. One recommendation is that the Radionuclide CDA be eliminated. The initial intent of this assessment was to add numerical precision to the core damage assessment. As a result of large uncertainties associated with the ability to measure the particulate and heavy radioisotopes, the precision associated with the assessment is questionable. The information will be provided after the fact and have no impact on accident management or emergency actions. In addition, the remaining 3 CDAs in combination provide an adequate CDAM. Furthermore, the implementation of the radioisotope CDAM is difficult and cumbersome and places an unnecessary drain on limited plant resources during normal operation (due to training and calibration and maintenance activities associated with these monitors) as well as following an accident.

The second recommendation is that when the hydrogen CDA is employed, that CDA be modified to exclude the dissolved hydrogen measurement. While dissolved hydrogen may exist in the RCS, the amount of dissolved hydrogen is expected to be small. Furthermore, assessments of the maximum hydrogen concentration may be bounded by simple application of Henry's law. Presently, this feature is intended to fine tune hydrogen production estimates to better quantify core damage. Such estimates are not currently included in core damage assessments used by the NRC (RTM-96) and the proposed WOG CDAM. Additionally, this measurement has been made via pressurized samples for most CE PWRs. This sampling activity leads to potential leaks in the auxiliary building and therefore is a potential for consequent personnel exposure.

5.3.4 Emergency Response Data System (ERDS)

10CFR50, Appendix E, Section VI indicates the plant data required to be input to the Emergency Response Data System. All CEOG member utilities have ERDS data programs that meet the associated 10CFR50 requirements. PASS data is not included within the ERDS.

5.4 Long Term Recovery

NRC requirements for long term recovery actions are found in 10CFR50.47 (b)(13) and 10CFR50, Appendix E (Paragraph IV.H). The licensee is required to develop criteria for re-entry into the facility following an accident. In the facility re-entry phase the plant has been stabilized and containment pressure and temperature are well controlled. Therefore, the long term recovery phase was considered a logical time to use the PASS (Reference 6). Since time is not a significant factor in the long term recovery process (cleanup/containment re-entry) any required chemical analyses may be performed off-site. The only feature of the PASS that has value would be the sample dilution capability. Thus, long term recovery actions would attempt to use PASS to minimize occupational exposure of personnel. However, adequate re-entry criteria may be established using area radiation monitors, portable health physics survey monitors, and other instrumentation as appropriate.

Table 5-6:

Typical RCS Gas and Structure Temperature Response to PWR Station Blackout++

Accident Condition	Time (min)	Fuel (^oF)	Core exit gas (^oF)	Upper plenum gas (^oF)	Upper plenum structure (^oF)	Hot leg gas (^oF)
Core uncovery	t_o	750	739.1	701.3	701.3	701.3
		1400	1200+			
		1700	1500+			
Melting of fuel pellets begins	t_o + 24	3600-4800	3320.3*+	1340.3	890.3	827.3
Melt relocation	t_o + 54	>4800	3977.3*+	1700.3	1385.3	971.3

* CET becomes erratic above 2300 °F. Also note that CET will lag gas temperature.
\+ CET lag gas temperatures by about 200 °F (See for example Reference 16).
++ Data based on Reference 28.

Table 5-7:

Typical RCS Gas and Temperature Response to PWR Small LOCA+ (based on MAAP*)

Accident event	Time (min)	Core exit gas (^oF)	Upper plenum gas (^oF)	Upper plenum Structure (^oF)	Hot leg gas (^oF)
Core uncovery	t_o	496.1	497.9	517.7	519.5
Melt initiation	t_o + 231	4904.3	2078.3	1293.5	919.1
Melt relocation	t_o + 468	4004.3	2366.3	2330.3	838.1

t_o Time of core uncovery
* Modular Accident Analysis Program (Reference 36)
\+ Data based on Reference 28

Table 5-8:

Typical Expected Core Behavior during an Accident

Core Temperature (F)	Core Behavior	Fission Product Behavior
4800	Melting of fuel pellets begins	*
4200	Release of all volatile fission products from fuel begins	100% of volatiles released
3600	Possible formation of uncoolable core	Approximately 80% of volatiles released
3000	Fuel pellets dissolving in melt components begins	Approximately 60% of volatiles released from fuel matrix
2400	Zircaloy water reaction accelerates	Very rapid release of volatile fission products from fuel pellets begins
1800	Very rapid Zircaloy water reaction begins; formation of hydrogen and failure of fuel rod cladding	Toward the higher end of the band between 10 and 33% of the fuel rods will remain intact.
1200	Initiation of possible fuel cladding burst – release of fission products in gap space	Gap releases from fuel upon rupture. Approximately 95% of the NG inventory and 5% of the I and CS inventory reside in the gap/plena of fuel rods.
<1200	Little possibility of fuel cladding rupture	Nobel gases and gap iodines, cesiums fission product release via cracks in cladding

* Prior to the onset of core melt, ~ 75 to 100% of the fuel rods in the core will experience cladding failure.

6.0 PASS ELIMINATION ASSESSMENT

In the previous section the role of the PASS was discussed with respect to its impact on various accident management and emergency response programs. This section provides a discussion of each required PASS sample analysis. The analysis can be performed using in-line instrumentation or a grab sample. In the following discussion, the PASS required sample is identified along with a brief summary of the rationale for the need for the sample and its use in the CEOG accident management/recovery process. This rationale is applicable for analysis utilizing either in-line instrumentation or a grab sample. This is followed by a summary technical basis for elimination of the requirement.

The general philosophy for the elimination of PASS sampling from the plant design basis is that:

- The samples are not required in the EOP/SAMG decision-making process. When the sample is referenced, redundant qualified means are available more quickly to accomplish the same objective.
- The requirements of 10CFR50.47 (emergency preparedness) may be adequately met without use of PASS. EALs can be declared, appropriate PARs can be formulated and CDA may be adequately completed and data may be supplied to the ERDS without reliance on PASS samples.
- While the PASS results in a more direct measurement of what it samples, it does not necessarily result in improved predictive capability. Furthermore, over-reliance on PASS can result in poor emergency planning decisions and unnecessary personnel radiation exposure.
- Finally, use of PASS requires significant plant resources to maintain and apply. Thus, decisions to take PASS samples may conflict with more pressing needs associated with accident mitigation.

The following sections provide an itemized discussion of the NUREG-0737 and related PASS requirements and a technical basis for their elimination.

Note that the discussion for the reactor coolant sample analysis capabilities is also applicable for containment sump sample. NUREG-0737 requires the licensee to have the capability to obtain and analyze samples of the reactor coolant. Specific sample points are not prescribed. As such, there is no specific requirement that the containment sump be a sample point. However, a sample obtained from the containment sump is representative of the reactor coolant during various severe accident scenarios, e.g. large break LOCA with recirculation available.

6.1 Reactor Coolant Dissolved Gases & Reactor Coolant Hydrogen

Purpose:
NUREG-0737 requires the determination of either total dissolved gas or dissolved hydrogen in the reactor coolant. The primary intent of this measurement was to provide guidance to the operational staff of the potential for the formation of a non-condensable gas bubble upon system depressurization. This parameter has also been utilized in the CEOG CDAM (Reference 16) to refine the core damage measurement evaluated using measured hydrogen production and thus is used in the assessment of core damage.

Recommendation: Delete Requirement

Justification:
Reactor coolant dissolved hydrogen or dissolved gas is determined by a number of possible sampling and analytical methods. Sampling typically involves capture of a pressurized liquid sample of a known

volume. The sample may then be vented to an evacuated vessel and degassed. Determination of total gas may then be done by a pressure-volume relationship or, alternatively, a sample may be withdrawn for analysis for hydrogen.

One of the principal difficulties in such a sample is that in order to minimize the potential dose to the sampling personnel, the size of the captured pressurized liquid sample is usually limited to 35 to 125 ml. This is far less than the typical 1000 ml sample used with the NSS during normal reactor operation. For events where extensive core damage has occurred, the dissolved hydrogen concentration is expected to be high and readily detectable. For cases where core damage results in hydrogen concentrations closer to the normal range of 25-50 cc(STP)/Kg-H_2O, the uncertainty of the measurement increases due to the smaller sample size. Obtaining this measurement is both time consuming and manpower intensive. Furthermore, test applications of this system have produced small leaks into the auxiliary building. During severe accident scenarios such leakage could unnecessarily expose plant personnel to high levels of radiation.

Post accident sampling releases to the environment were considered to be of such importance as to be explicitly included in one plant's Maximum Hypothetical Accident (MHA) design basis for calculating doses at the exclusion area boundary, low population zone and the control room. This MHA analysis suggests that the post accident releases can contribute ~ 2 rems to the thyroid dose.

Analysis for hydrogen may also be performed using an in-line analyzer. While this eliminates the difficulties noted above, the use of in-line instrumentation requires that the ability to obtain and analyze a grab sample as backup be retained. Consequently, use of in-line instrumentation does not relieve the licensee of the burdens and questionable accuracy associated with the grab sample.

As stated above, the need for a dissolved gas sample is based on the premise that the plant staff must know the dissolved concentration of oxygen and hydrogen in order to properly depressurize the RCS following a beyond design basis accident. The intent of this PASS parameter was to make up for lack of information regarding void formation and procedures to deal with its occurrence. However, as part of NUREG-0737 the NRC required the utilities to: (1) upgrade procedures, (2) install instrumentation to directly observe formation of a bubble in the RCS and (3) add equipment to RCS high points to vent the bubble. The instrument which monitors void formation in CEOG plants is typically the RVLMS. The RVLMS utilizes sets of heated junction thermocouples to sense the presence of a liquid vapor interface. This system is a safety-grade ICC plant component and is capable of monitoring the formation and elimination of the RCS void at typically 8 axial locations above the top of the fuel alignment plate. In the current EOPs, the void elimination procedure is written without consideration of PASS samples. Procedures indicate that monitoring of the pressurizer and reactor vessel upper head level will provide indication to the operators of incipient bubble formation. Corrective action may then be taken to control the rate of depressurization and resulting bubble formation or to vent the bubble via the pressurizer or reactor vessel head vent. Consequently, voids are appropriately dealt with based on their occurrence. In any event, knowledge of the precise RCS void content could not avoid the void formation, nor would it modify void elimination strategies.

One utility (Southern California Edison) requires that the chemistry department sample the RCS for dissolved hydrogen every 4 hours during cooldown. The purpose of this measurement is to support normal plant operational activities associated with the volume control tank purge in an attempt to reduce RCS hydrogen prior to transitioning to shutdown cooling and, if needed, may be accomplished via the NSS.

Availability of this dissolved hydrogen concentration measurement has led to use of a complex RCS dissolved gas assessment for use in the computation of core damage. Inclusion of this information in the CEOG CDAM was provided for completeness. For practical considerations, increased knowledge of the progression of a severe accident suggests that use of the dissolved gas assessment provides only a minor perturbation on the CDA except for the case where core damage is very limited. Any benefits obtained by crediting dissolved hydrogen are overshadowed by errors introduced by combining two samples which may be obtained at different times (one in the RCS and one in the containment). The possible consequence is to double count hydrogen or inaccurately combine the samples. Other CDAMs including that of the NRC, do not credit dissolved hydrogen (Reference 18).

Status:
Partial relief of this sample/analysis was granted by the NRC in CEN-415, Revision 1-A. In that report, the NRC agreed that the normal sampling system could be used for accidents where low concentrations of dissolved gas were expected. This allowance is contingent upon the ability to determine that use of the normal sampling system will not result in exceeding GDC 19 requirements for personnel exposure.

Conclusion:
As noted above, the need for a hydrogen PASS sample to guide the operator in plant cooldown is not supported by fact. Means of observation of the incipient bubble are directly available to the operator as is a means for control of the formation of non-condensable gas bubbles in the RCS. These methods do not rely on obtaining an analysis of RCS dissolved gas.

While dissolved H_2 is formally used in the existing CEOG CDAM, the use of a dissolved hydrogen measurement is unnecessary. This item is an outgrowth of an excessively analytical effort to closely quantify core damage to a high degree of precision. This precision diverts resources from more relevant plant diagnostics and may result in wrong conclusions. It is recommended that a modification to the CDAM be made to eliminate RCS H_2 measurements and replace them with a Henry's Law Calculation for dissolved H_2.

Finally, the potential for a high pressure sample line leak in the auxiliary building may unnecessarily expose the plant staff to high levels of radiation.

6.2 Reactor Coolant Oxygen

Purpose:
The purpose of sampling the RCS for dissolved oxygen is to assess the potential for chloride induced Stress Corrosion Cracking (SCC) of stainless steel piping and components. Analysis of RCS oxygen is not a requirement of NUREG-0737; however, it is included in RG 1.97.

Recommendation: Delete Requirement

Justification:
There are no accident management or emergency planning functions that require identification of the reactor coolant oxygen content. The requirement for reactor coolant oxygen sampling and analysis is tied to preventing chloride induced stress corrosion cracking of stainless steel piping and components, which ensures that continued long term cooling of the core is not compromised.

As noted in the section on reactor coolant chlorides, SCC is mitigated by the control of reactor coolant pH without regard to the oxygen concentration.

Conclusion:
Presentation of material including the above rationale have been made to the NRC and elimination of this parameter was approved by the NRC in CEN-415 Revision 1-A. Consequently, there is no need to maintain a post accident system capability to sample and analyze the RCS oxygen.

6.3 Reactor Coolant Chlorides

Purpose:
The purpose of sampling the RCS for chlorides is to assure that chloride induced SCC of stainless steel piping and components will not occur in the long term.

Recommendation: Delete Requirement

Justification:
The NRC has recognized that the potential for high concentrations of chlorides in the reactor coolant system is a strong function of the plant design and the water source for the ultimate heat sink. This is evidenced by the requirements in NUREG-0737 for the time at which the first sample for chlorides must be taken. For fresh water plants and plants with brackish or salt water with more than one barrier between the containment and the ultimate heat sink the initial chloride sample is not required for 96 hours (4 days). For brackish and salt water plants with only one barrier between the potential source of chlorides and the containment the first chloride sample is required in 24 hours.

SCC is a function of temperature, chloride and oxygen concentration and pH. There is no removal mechanism for chloride in post accident scenarios; however, SCC may be controlled via pH control of the reactor coolant.

For most severe accident scenarios, the principal means of chloride intrusion will be from uncontrolled sources such as the containment sump water. All CEOG plants with the exception of St. Lucie Unit 1 control sump pH via passive means, e.g. Trisodium Phosphate Dodecahydrate (TSP) stored in containers on the basement floor of containment. The TSP rapidly dissolves in post-LOCA flood water to neutralize acid species. For these plants, the control of reactor coolant pH is passively ensured and chloride induced SCC is mitigated.

St. Lucie Unit 1 relies on a containment spray additive, sodium hydroxide (NaOH), to control sump pH. For accident scenarios where containment spray is automatically actuated, the result is identical to those plants employing the passive control method. For the accident scenarios where containment spray is not actuated, the decision may be made by plant operators to manually initiate spray to ensure that the sump pH is controlled.

Additionally, post-LOCA sump pH is controlled in the desired range to minimize the revolatilization of radioiodines.

Conclusion:
The corrective action for a chloride intrusion is to control pH. Passive and active means exist to accomplish this for both SCC mitigation and airborne radioiodine control. As control of sump pH is a pre-analyzed parameter based upon the expected sump composition there is no additional benefit to the chloride analysis and the requirement should be deleted.

6.4 Reactor Coolant pH

Purpose:
The purpose of sampling the reactor coolant for pH is to assure that chloride induced SCC of stainless steel piping and components will not occur in the long term and to assure that radioiodine species are retained in the water.

Recommendation: Delete Requirement

Justification:
As discussed in Reference 8, NUREG-0737 does not include a specific requirement to monitor the pH of the reactor coolant. RG 1.97 does include a recommendation to monitor the pH of the liquid in the sump. In the post-LOCA environment, pH in the sump is controlled to ≥ 7 either by sodium hydroxide addition via containment spray (St. Lucie 1), or passive addition of TSP (all remaining CE Units) as previously discussed. The reliability of this control is sufficiently high to ensure that neither corrosion of the stainless steel components nor re-evolution of iodine species will occur.

Conclusion:
Deletion of the pH measurement was approved by the NRC in CEN-415 Revision 1-A. Consequently, there is no requirement. Since the results are confirmatory only, this sample and analysis should be deleted.

6.5 Reactor Coolant Boron

Purpose:
The purpose of sampling the RCS for boron is to assure that there is adequate shutdown margin of boron in the reactor coolant system to enable cold shutdown to be achieved. The post accident sampling capability to measure the reactor coolant boron is a NUREG-0737 and a Regulatory Guide 1.97 function.

Recommendation: Delete Requirement

Justification:
Verification that adequate shutdown margin exists is an immediate concern in an accident scenario. The PASS, with an approximate sampling and analysis time of three hours, does not support this need.

A review of the CEOG member utility EOPs/AOPs (See Appendix A) reveals that there are currently several general uses of boron concentration in the EOPs:

1. Ensuring that the reactor remains shut down after reactor trip and during cooldown.

2. Verification of Reactivity Control Safety Function Status Check..

3. Ensuring that the reactor remains shut down during a cooldown in Station Blackout.

4. Ensuring that the reactor remains shutdown following a SGTR event with back leakage.

5. Confirming adequate boration in the RCS prior to RCP restart.

In all cases where boron concentration is directly or indirectly referred to, there are always other corroborative indications that can be used by the operator to verify that the reactor is shutdown and will remain shutdown. For example: CEA position indication and insertion limits, reactor power lowering or stable, negative startup rate, positive indication of the flow of a high concentration of boron into the RCS. It was noted that at Palo Verde Nuclear Generating Station, preanalyzed boron addition via the RWST shows adequate shutdown margin and no sampling is required. In the case of sampling prior to RCP restart, the primary concern is sending a slug of under-borated water into the RCS. However, if under-borated water is present it would likely accumulate in the pump suction line, which is an unmonitored location. Thus, the information established via the hotleg boron sample will at best be misleading.

In light of arguments similar to the above, the NRC has granted the delay of initial PASS boron sampling from 3 hours to 8 hours. This delay was largely based on the understanding that no short term actions will occur following a severe accident which would require the monitoring of shutdown margin. Even then the basis for the 8 hours was arbitrary. Reactivity control may be provided without a detailed boron sample. Thus, sampling of the sump/RCS for confirmation may be delayed to such a time that the normal sample system is available.

From a degraded core perspective, relocation and compaction of CE designed cores will severely limit the potential for recriticality. Thus, as discussed previously, boration of a severely damaged core is not necessary (See Reference 23).

Conclusion:
Based on the above assessments, a PASS sample for reactor coolant boron does not support the immediate need to verify that adequate shutdown margin exists. Alternative methods will provide sufficient indication of shutdown margin such that this sample/analysis may be downgraded to a recommended action. This will require changes in the plants' EOPs.

6.6 Reactor Coolant Conductivity

Purpose: Conductivity measurements typically are used to confirm other analyses. For example, ionic species, e.g. boron, chloride, etc., contribute to solution conductivity. If an imbalance exists between the measured conductivity and that expected based on the concentration of the analyzed species, it indicates either an error in the analyses or the presence of additional ionic species which were not included in the analysis matrix.

Recommendation: Delete analysis

Justification:
There is no requirement in NUREG-0737 or Regulatory Guide 1.97, Revision 3 (Reference 4) to measure the conductivity of the reactor coolant. In addition, there are no accident management or emergency planning functions that require knowledge of the RCS conductivity.

Conclusion:
The determination of reactor coolant conductivity does not provide significant information and should be deleted.

6.7 Reactor Coolant Radioisotopes

Purpose:
The purpose of sampling the RCS for radioisotopes is to provide information for input to the CDA, to establish reactor coolant activity (without isotopics), and to provide a basis for one of the Alert EALs. The Alert emergency class is typically declared when I^{131} exceeds 300 μCi/gm. Reactor coolant activity measurement (without isotopics) is suggested prior to entry into SDC.

As discussed in the introductory material, reactor coolant may be sampled in the RCS or the containment sump. The following discussion applies to the use of radionuclide samples obtained from either location.

Recommendation: Delete analysis

Justification:
The current CEOG Core Damage Assessment Methodology (Reference 16) provides four independent overlapping procedures for estimating the degree of core damage. For the lesser core damage estimates, all four (4) procedures apply including estimating core damage via core exit thermocouple indications, containment hydrogen concentration, containment radiation monitor and radionuclide assessment. Use of the fourth method is intended for the most severely damaged core scenarios and its use provides no useful input for selecting SAMG CHLAs, establishing EALs or formulating PARs.

Coolant Activity Assessment

The EOPs for the LOCA contains a statement regarding RCS activity in the guidance for entry into shutdown cooling. The EPG cautions that the operators should be aware of the coolant activity prior to SDC entry because it represents a potential release path to the environment and may also result in restriction of access to portions of the auxiliary building. Specific direction to analyze for radionuclides is not given. The notation is strictly cautionary in nature and will not impact the function of safety systems. For example, in a large break LOCA, a Recirculation Actuation Signal (RAS) will occur in as little as 20 minutes after the event initiation. The RAS is an automatic function which will circulate water from the containment sump, through the high pressure safety injection and containment spray pumps, the SDC heat exchangers, and back into containment. This is a design basis event which is not impacted by the coolant activity levels. In the case of an event in which a lesser amount of core damage occurs, the SDC entry procedure will provide guidance to ensure that precautionary measures are taken prior to initiating SDC flow. These precautionary measures may be conservatively made in the absence of a coolant radioisotopic analysis.

Core Damage Assessment

RCS liquid radioisotopes samples in Core Damage Assessments are primarily and formally used in one of the four core damage assessment protocols developed in Reference 16. For badly damaged core conditions there is little value for this assessment. Since significant core uncovery has occurred the significant quantities of radioisotopes would have left the RCS or would have plated out in regions away from the sample point and would not necessarily be dissolved in the RCS sampled coolant. This level of core damage may be adequately assessed via use of the remaining three CDA protocols (CETs, hydrogen and containment radiation) along with event specific information available from plant related SAMG observations.

At the low fuel damage portion of the spectrum the PASS may be used to estimate if the I^{131} equivalent is above or below 300 μCi/gm (< 5% cladding damage). This measurement is intended to aid the declaration of an Alert. In practice, this measurement will occur well after other indicators have emerged that would trigger the Alert classification. Alternate methods to assess this level of core damage are available to the plant staff. These methods include use of the failed fuel monitor or dose estimates in the vicinity of sample bombs.

Conclusion:
Eliminating the sampling of RCS radioisotopes will not impact the ability of the plant to manage the accident or effect appropriate emergency response. With regard to the EOPs, isotopic analysis is not required prior to entering shutdown cooling. The ability to assess core damage may be accomplished via methods that do not rely on the reactor coolant radioisotopic analysis. The existence of procedures to provide CDAs which are sufficient to provide guidance to make appropriate operational decisions in the absence of such an analysis supports the deletion of this requirement. This will require additional review of the current CEOG CDAM. The decision to enter into shutdown cooling is not prohibited in the absence of coolant radioisotopic analysis. Therefore, deletion of this parameter is acceptable.

6.8 Containment Atmosphere Hydrogen

Purpose:
The purpose of sampling the containment atmosphere for hydrogen concentration is to provide a means of assessment of core damage and to monitor the potential formation of a combustible atmosphere in containment. Analysis of containment hydrogen concentration is a requirement of NUREG-0737 and Regulatory Guide 1.97, Revision 3.

Recommendation:
Maintain capability to monitor hydrogen in the containment atmosphere. Delete PASS sample requirements.

Justification:
NUREG-0737 requires the capability to quantify the hydrogen levels in containment. Further, if in-line monitoring is used, there is a requirement for the capability of obtaining a backup grab sample. The requirement for a backup grab sample provides additional confidence in the ability to obtain a sample since PASS instrumentation is not required to be safety grade.

Containment hydrogen is best determined through the use of the current in-line safety-grade hydrogen monitors installed in CEOG plants. These in-line monitors satisfy the requirements of Regulatory Guide 1.97 and provide real time data that can assist the operators in assessing core damage long before a grab sample could be obtained and analyzed. The safety-grade monitors provide indication of the potential hydrogen combustion threat which can have an impact on both SAMG actions and emergency planning PARs. The safety-grade qualification ensures the accuracy of the results and minimizes the human factors associated with grab sample collection and analysis. The redundant trains of these monitors obviates the need for a backup grab sample and the intent of the PASS requirement is met.

Conclusion:
Approval of the use of containment safety-grade hydrogen monitors was granted by the NRC in CEN-415, Revision 1-A. Therefore, removal of this requirement from PASS is justified. In the associated SER, the NRC staff concluded that because the hydrogen monitors in the containment provide adequate

capability for monitoring post-accident hydrogen, there is acceptable justification for this deletion and found the requested modification acceptable.

6.9 Containment Atmosphere Oxygen

Purpose:
The purpose of sampling the containment atmosphere for oxygen concentration is to provide an indication, along with the hydrogen analysis, of the potential for the formation of a combustible atmosphere in containment. However, there is no requirement for PASS capability to measure the containment oxygen concentration in NUREG-0737. Regulatory Guide 1.97 includes a recommendation to measure containment oxygen concentration. Currently, of the CEOG plants, only Palo Verde has this analysis within its design basis.

Recommendation: Delete requirement

Justification:
The capability to measure oxygen concentration in the containment does not support any accident management or emergency planning functions. Containment oxygen levels may be readily approximated from the initial containment conditions assuming an air atmosphere, the steam partial pressure and the hydrogen concentration from the post-LOCA containment hydrogen monitors.

Conclusion:
There is no requirement for PASS to analyze for containment oxygen concentration. Further, a means exists to approximate the concentration from existing containment conditions. This requirement may therefore be deleted.

6.10 Containment Airborne Radioisotopic Samples

Purpose:
The purpose of sampling the containment for radionuclide content is to enable offsite dose assessments to be made for both post accident containment leakage and containment failure conditions.

Recommendation: Delete requirement

Justification:
When NUREG-0737 was originally conceived in the early 1980s, it was imagined that the most accurate assessment of offsite doses would result from using the containment airborne radionuclide estimates found from the analysis of samples. However, given the time required to obtain and analyze a sample in relation to the dynamic processes that are occurring in the containment during and following an accident, the information obtained from samples of containment atmosphere would not be timely. Furthermore, considering the behavior of fission products, it is apparent that the sample results are not very accurate. In the early 1980s it was believed that the airborne iodine will be primarily volatile elemental iodine gas or organic iodine compound. Following extensive government funded research over the past two decades, the expectation of the source term constituents has changed from primarily volatile to overwhelmingly particulate. This has an important implication in the ability of a remote system to accurately assess airborne iodine concentrations. For example, for many core damage accidents a significant portion of the volatile and non-volatile fission products would be deposited on reactor coolant system internal surfaces and would not be released to the containment. Therefore, the assessment of core damage based on the containment radionuclides could be severely distorted. In addition, severe accident

analyses have found that when the containment is depressurized (as in a containment pressure boundary failure or an intentional release through a containment vent), a significant fraction of the fission products previously deposited on internal surfaces of the reactor coolant system could be released to the containment and subsequently to the atmosphere. Thus, the estimation of offsite consequences due to a release from containment following a core damage accident, based on the containment inventory of radionuclides, may significantly underestimate the actual consequences.

Since the PASS is intended to sample post accident containment conditions, it cannot provide useful data following severe accidents which bypass the containment either via a "V" sequence (inter-system LOCA) or via a SGTR with SG releases. Information provided with respect to the containment are also suspect due to issues associated with (1) obtaining data from a "truly representative" sample point, (2) plateout of cesium iodides (CsI) in the sample lines, and (3) time delays associated with obtaining, processing and interpreting the sample during non-stable phases of the accident. Item 3 refers to issues associated with transient generation of the fission products within the core and its subsequent redistribution to cooler portions of the RCS, secondary side, and containment, as well as the impact of containment conditions (spraying, operation of fan coolers, containment leaks, etc.) on the fission product inventory.

Thus, sampling the containment atmosphere to obtain a source term for offsite dose calculations is not a reliable means of predicting offsite doses and is limited to a select set of severe accidents. For containment leakage scenarios, the use of radionuclide assessments as a basis for fine tuning and establishing PARs may underestimate the risk to the population. This under-estimation is a direct result of the sampling process whereby fission products aerosols (primarily cesium iodides) that could otherwise leak to the environment may be deposited in the sampling line prior to reaching the monitor. Even with precise knowledge of the containment radionuclide airborne distribution, estimation of radioactive releases will be challenging as particulates (for example CsI) will be filtered via the leakage pathways, and potentially clog them.

In the case of potential containment failure or containment venting, the use of containment atmosphere samples would under-predict the actual releases as the containment pressure is reduced due to re-evolution of aerosol fission products from surfaces within the containment, as well as transport of fission products in the RCS. Severe accident analyses, such as those summarized in the EPRI Severe Accident Management Technical Basis Report, show that the aerosol fission product inventory in the containment increases when the containment is depressurized.

The current CEOG Core Damage Assessment Methodology provides four independent overlapping procedures for estimating the degree of core damage. For the lesser core damage estimates, all four CDA methods apply. These four methods include procedures for estimating core damage via core exit thermocouple indications, containment hydrogen concentration, containment radiation monitor and radionuclide assessment. Use of the fourth method is intended for the most severely damaged core scenarios and its use provides no useful input for selecting SAMG CHLAs or establishing EALs.

According to the EPGs or the SAMGs, after recovery from a core damage accident is completed, there may be a need to accurately determine the airborne containment fission products so that post-accident recovery actions can be planned and plant cleanup can commence. In this case, the containment would be at nearly atmospheric conditions and a sample of the containment gas space would provide an accurate assessment of the airborne noble gases and small quantities of aerosols that may have to be vented to atmosphere to gain access to the containment. Such samples may be obtained and analyzed without reliance on the PASS.

Conclusion:
Partial relief of this requirement was previously granted by the NRC in CEN-415, Revision 1-A which allows deletion of the heat tracing requirement on the sample line if radioiodines are not used for core damage assessment. Since alternative means exist for the assessment of core damage that do not rely on containment radionuclide analysis, this requirement may be deleted.

Note that while heat tracing may impact the revolatilization of the deposited elemental iodines, it will not affect the deposition of CsI particulates (which is expected to be the dominant chemical form). This compound has a vaporization temperature of approximately 1280 ^{o}F.

The "in containment" radioisotope sampling capability provided by the PASS is better performed by an iodine site survey detection capability (such as that identified in NUREG-0654). Site survey capability is applicable to all accident scenarios and release points and provides a realistic means for adjusting ARM-based dose projections. Elimination of PASS radionuclide samples and reliance on site samples is a risk neutral action.

7.0 PROPOSED PASS ELIMINATION PROGRAM

This Section summarizes the CEOG PASS Elimination Recommendations. As discussed in the preceding pages the PASS is a costly intensive system that provides minimal (if any) benefit in accident management or emergency planning. Elimination of PASS requirements will result in group savings to the CEOG in excess of $20,000,000 over the remaining life of the units. Based on a review of NUREG-0737 (and related) requirements, it is recommended that all PASS sample requirements be removed and replaced by flexible functional requirements to:

- Establish hydrogen concentration in the containment atmosphere,
- Ensure post-accident reactivity control,
- Ensure adequate pH control of RCS coolant.

All the above requirements may be met without reliance on the PASS. The detailed recommendations of this effort are contained in Table 7-1.

In implementing the above changes, the utilities will retire all automatic PASS monitors “in place”, return PASS system valves to a normal (non safety-related) maintenance program, and relocate system training within standard chemistry procedures and re-evaluate the capability of the NSS or other system to obtain and evaluate lower level fuel damage conditions. In addition, adherence to NUREG-0654 ensures offsite monitoring capability of iodines released to the environment. Implementation of the PASS removal will also result in several non-risk significant (50.59) changes to various EOPs, AOPs, SAMGs and elements of the plant emergency plan CDA. The changes to these documents can be made in accordance with 10CFR50.59 and 10CFR50.54(q) provided the NRC agrees with the technical basis contained in this report.

Table 7-1: CEOG Recommendations Regarding PASS Elimination					
No.	**Item**	**Accident Mitigation or Emergency Programs, where Sample is Referenced**	**Recommendation**	**Basis**	**Utility Action**
1	Sample of Reactor Coolant Dissolved Gases & Reactor Hydrogen.	Hydrogen CDAM[(1)]	• Delete requirement. Modify CEOG CDA to consider dissolved H_2 in assessment analytically via use of Henry's Law	• Required in NUREG-0737 and RG 1.97 • Dissolved gas measurement not used in EOPs, void removal based on use of RVLMS with RVUH or PZR head vent • Use in CDA to "fine tune" assessment not important. Alternative method can bound result. • Leakages associated with system during sampling of a post-degraded core scenarios may result in fission product releases outside of containment (Contribution of ~ 2 rems following an MHA possible to EAB, LPZ and control room). Operation of PASS may limit access to AB.	• Remove requirement from T.S. • Adjust CDAM, as appropriate
2	Reactor Coolant Oxygen	None	• Delete requirement	• Required in RG 1.97 • Passive pH control provided via use of TSP. Approved via CEN-415-A. pH control will prevent chloride induced SCC of stainless steel piping.	• Oxygen samples may be removed from procedures plants with passive pH control. Plant w/o passive pH control must ensure long term pH control for transients where additive system is not automatically actuated.[(2)]
3	Reactor Coolant Chlorides	None	• Delete requirement for fresh water plants and brackish water plants with multiple barriers	• Required in NUREG-0737 and RG 1.97 • Passive pH control provided via use of TSP. Approved via CEN-415-A. pH control will prevent chloride induced SCC of stainless steel piping.	• Plant w/o passive pH control must ensure long term pH control for transients where additive

[(1)] RCS hydrogen assessment not included in ANO-2 CDAM.

[(2)] All CE PWRs with the exception of St. Lucie Unit 1 have implemented passive pH control measures. It should be noted that spray actuation (which provides NaOH to the containment) is expected to occur following accidents requiring sump recirculation (large and medium LOCAs). Long term pH control can be assumed in plants without passive control by operator action to inject NaOH manually.

Table 7-1: CEOG Recommendations Regarding PASS Elimination					
No.	**Item**	**Accident Mitigation or Emergency Programs, where Sample is Referenced**	**Recommendation**	**Basis**	**Utility Action**
			between the containment and ultimate heat sink.	• Chloride sources are known and controlled. Unintentional and unknown addition of chlorides into RCS is considered remote.	system is not automatically actuated.[(2)]
4	Reactor coolant pH	None	• Delete requirement	• Required in RG 1.97 • Passive pH control provided via use of TSP. Approved via CEN-415-A. pH control will prevent chloride induced SCC of stainless steel piping and control re-evolution of iodine gases from the sump.	• Plant w/o passive pH control must ensure long term pH control for transients where additive system is not automatically actuated.
5	Reactor Coolant Boron	EOPs, Supporting procedures & SAMGs	• Delete PASS requirement • Where requirements exist, delete requirement for sample and replace with <u>optional</u> action to assess shutdown margin via sampling of RCS boric acid concentration using the NSS.	• Requirement of NUREG-0737 and RG 1.97 • Success of the reactivity safety function may be established via several means including - NSS samples [(3)] - Startup detectors - Analytical balances - Indications of injected borated flow • Ample approaches are available to ensure post-accident reactivity control post-accident. Thus any sample could be delayed to the point a sample may be taken via existing non-PASS system. • No short term need exists to sample boron • Confirmatory following a degraded core accident. • Not needed following core relocation.	• Remove required actions to sample boron and replace as recommendation based on specific concerns regarding inadequate boration. Expectation is to use NSS within EOPs. Alternative measure to assess boron will be identified and implemented.
6	Reactor Coolant Conductivity	None	• Delete requirement	• No regulatory requirements	• Conductivity measurement may be deleted from any

[(3)] Capabilities of NSS to monitor boron concentration varies among the CE units (See Section 5.1.3).

Table 7-1: CEOG Recommendations Regarding PASS Elimination

No.	Item	Accident Mitigation or Emergency Programs, where Sample is Referenced	Recommendation	Basis	Utility Action
					accident management or emergency planning program.
7	Reactor Coolant Radioisotopes	Emergency Planning: used in CDA, and EAL (ALERT declaration)	• Delete Requirement	• Requirement of NUREG 0737 and RG 1.97 • Included in CDA to define low end of a damaged core (< 5% fuel failure) • Low level CDA assessments have minimal impact on public risk. • Emergency class declaration may be established via alternatives - Event based declarations - Use of failed fuel monitors, NSS - ICC monitors, alternate dose assessment • CDA may be established via failed fuel monitors, letdown radiation monitor or NSS.	• Establish non-PASS method for addressing EAL associated with DNB/reactivity addition transients. • CDA procedures should allow bounding estimates until a clad failure assessment may be made using existing non-PASS systems. • Alternative methods may also be used (See for example 5.1.4). • For low level sample capability of NSS; capability will be defined and may be extended.
8	Containment Hydrogen	EOPs – H_2 recombiner startup Containment Purge SAMGs, ERDS	• Delete from PASS, but retain functional requirement for use during severe accidents (SAMGs)	• Containment hydrogen concentration containment supports - CDA (for most utilities) - Containment challenge assessment and, - Limited CHLA strategies • Relief granted via CEN-415-P	• None for utilities that use Class 1E H_2 monitors to fulfill this function. • For other utilities, when needed, H_2 monitoring may be performed by Class 1E monitors.
9	Containment Oxygen Sample	None	• Delete requirement	• Measurement does not support any accident management or emergency planning function.	• Applies only to PVNGS, delete upon approval of request.

Table 7-1: CEOG Recommendations Regarding PASS Elimination					
No.	**Item**	**Accident Mitigation or Emergency Programs, where Sample is Referenced**	**Recommendation**	**Basis**	**Utility Action**
10	Containment Airborne Radioisotopic Samples	Indirectly support Emergency Planning via EALs, PARs and trending of doses and CDA.	• Delete requirement	• Sufficient information exists to categorize core damage from an EP perspective w/o a radioisotopic-based CDA • Radioisotopic CDA may be depressed due to plateout and related sampling issues. • Site doses do not directly rely on PASS. PASS information if available may be used to only support dose projections for a limited number of scenarios. • EALs may be established w/o detailed radioisotopes. • PARs may be formulated via use of ARMs and CETs. • Dose projections may be established via analytical adjustment to ARM readings, site surveys and vent releases.	• Remove radioisotopic CDA for plant CDAM. • Remove reference to PASS radioisotopic assessment as confirmatory backup for dose calculations. • Maintain offsite capability to monitor iodines

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APPENDIX A

CURRENT USE OF PASS IN PLANT EOPs AND AOPs

Assessment of use in PASS Emergency Operating Procedures (EOPs) and Abnormal Operating Procedures (AOPs)									
	FCS	**SL1**	**SL2**	**WSES**	**SONGS**	**PVNGS**[1]	**ANO-2**[8]	**MP-2**	**PALISADES**
1. Sampling of Boron in Reactor Coolant									
• Post trip verification (optimal) and cooldown	NSR	NSR	NSR	Note 4	NSR	NSR	NSR	NSR	NSR
• Prior to RCP restart (See Note 6)	NSR	NSR	NSR	NSR	NSR	NSR	NSR	NSR	**(R)**
• Monitor dilution due to backflow of secondary liquid into RCS following SGTR	Note 7	NSR	NSR	NSR	NSR	NSR	NSR	NSR	**(R)**
• Monitor Boration during SBO cooldown	NSR	NSR	NSR	NSR	NSR	NSR	NSR	NSR	NSR
2. Sampling of Hydrogen in RCS	NSR	NSR	NSR	NSR	Note 2	NSR	NSR	Note 3	NSR
3. Sampling of Hydrogen in Containment Atmosphere	NSR	NSR	NSR	NSR	NSR	NSR	NSR	NSR	NSR
4. Sampling Reactor Coolant Activity prior to SDC entry	NSR	NSR	NSR	NSR	NSR	NSR	Note 5	NSR	NSR

(R) - required by procedure
NSR - Not Specifically Required: Action to sample may be in EOPs/AOPs however, sample may be taken via NSS. Failure to sample does not degrade effectiveness of EOPs (See Reference 14).

1. No PASS sample is required to support EOPs.
2. Samples every 4 hours to VCT purge during transition to SDC.
3. When hydrogen analyzers are not available.
4. Procedure requires boration to adequate SDM. Implies sampling of RCS is necessary to verify adequate SDM. Use of PASS is not specifically required.
5. Procedures for high activity in RCS state: when letdown monitors reach a pre-determined level, an RCS sample is required to verify indication is true. Above these levels, RCS samples are required to be taken via the PASS.
6. Applies when natural circulation within the RCS has occurred during a SGTR.
7. Secondary side sample sampling required to assess affected generator.
8. Existing procedures utilize NSS up to 0.025% FF could be used to monitor RCS liquid with low levels of core damage (See Table 5-2).

APPENDIX B

CURRENT USE OF PASS IN PLANT SAMGs

Review of Role of PASS in SAMG	
Plant	**Description**
FCS	Not relied on SAMGs. Information not required but may be considered if timely.
SL1 & SL2	SAMGs are separate guidelines that are considered at Site Area Emergency or higher classification and when EOP-15 actions fail to mitigate the emergency condition. The E Plan and SAMGs are maintained as separate programs, but SAMGs would be implemented as part of E Plan related response. PASS samples are not relied upon in SAMG implementation.
WSES	PASS samples are not relied upon in SAMG diagnosis, but are identified as one means of monitoring the physical condition of the core in some CHLAs.
SONGS	PASS samples not relied upon in SAMG implementation, however, chemistry samples (including PASS samples) are considered as supplemental information within SAMGs
PVNGS	Not relied on or used within SAMGs.
ANO-2	PASS is referenced in SAMGs, however, it is not relied upon to take actions.
MP-2	PASS samples not relied upon in SAMG implementation, however, chemistry samples (including PASS samples) are considered as supplemental information within SAMGs
PALISADES	Post-accident sampling monitoring is not required as part of SAMGs.

APPENDIX C

CURRENT USE OF PASS IN DECLARING EMERGENCY CLASSES

Role of PASS in Emergency Planning

Plant	PASS related EAL Declarations	Formulation of PARs	Input to ERDS
FCS	Used for ALERT @ 180 μCi/cc I^{131} Eq (See Note 5)	PASS not used, PARs refined by field team. Site survey includes iodine monitor.	No PASS data transmitted
SL1	Used for ALERT @ 275 μCi/cc I^{131} Eq	None, See Note 1	No PASS data transmitted
SL2	Used for ALERT @ 275 μCi/cc I^{131} Eq	None, See Note 1	No PASS data transmitted
WSES	Used for ALERT @ 275 μCi/cc I^{131} Eq	None	No PASS data transmitted
SONGS	Not used (See Note 4)	None, See Note 4	No PASS data transmitted
PVNGS	Used for ALERT @ 300 μCi/cc I^{131} Eq	None, See Note 7	No PASS data transmitted
ANO-2	Used for ALERT @ 378 μCi/gm I^{131} Eq	PAR criteria divided on fuel damage levels greater or less than 10%. (See Note 3)	No PASS data transmitted
MP-2	Not used (See Note 2)	Not available	No PASS data transmitted
PALISADES	(None, > 25 μCi/gm I^{131} used to define ALERT (See also Note 6)	None, See Note 6	No PASS data transmitted

Notes:

1. St. Lucie Units:

EAL
PASS does not enter into classification. Only low activity (less than 275 μCi/cc I^{131} DEQ) sample results are used in classification scheme. For core damage indication, other passive methods are used to determine core damage relevance and classifications (such as, CHRRM dose rate, CET temperature, dose rates in the environment, etc.) PASS is not expected to be used until long after stabilization of the plant conditions to determine after-the-fact core damage estimations.

PAR
None, PASS sample data is not directly used in PAR determination. Data to answer "severe core damage" question is assessed via use of Containment High Area Radiation Monitors (CHARM), CETs and the timing and extent of the loss of core cooling capability. All of these are precursors to actually observing severe core damage.

ERDS
No PASS data is transmitted over ERDS. However, the NRC is expected to request PASS activity, and this information would be transmitted over ENS phone and/or faxed. Any PASS relevant information would not be valuable until plant core conditions have stabilized (long after the event).

2. Millstone Unit 2:
   Alert declaration based on dose reading 1 foot from sample bottle.

3. ANO-2:
   PARs based on Failed Fuel Levels 10% CDA may be established via "in-plant" instruments. CDAM based on non-CE product.

4. SONGs:
   EAL
   PASS is used in determining core damage.

   PAR
   To determine PARs, first a dose assessment is performed as follows:
   - Monitor Releases: Process Rad Monitors
   - Unmonitored Releases: Field Team, Pressurized Ion Chambers, Dose Assessment calculation based on estimated NG/I ratio.

   Note that PASS is not directly mentioned in the procedures.
   SCE does not use hydrogen measurements in EALs, PARs or ERDS.

5. FCS:
   Normal Sampling System capable of monitoring > 7% fuel failures ("gap releases").

6. Palisades:
   Sample for NOUE and ALERT declarations based on use of NSS CDA for PARs primarily based on containment monitors.

7. Palo Verde Nuclear Generating Station (PVNGS):
   PARs are based on emergency classification (EAL), plant conditions and dose projections. PASS would only be used as confirmatory to safety grade indicators.

APPENDIX D

CURRENT USE OF PASS IN FORMULATING PROTECTIVE ACTION RECOMMENDATIONS AND ESTABLISHING DOSE PROJECTIONS

Role of PASS in Offsite Dose Projections and Dose Monitoring		
PLANT	**Methodology Used for Offsite Dose Projections**	**Offsite Dose Monitoring**
FCS	Off-site doses for Ft Calhoun Station are projected based on effluent monitor data followed by radiation monitor data. Almost all effluent monitors are operable up to the point where Site Area Emergency would be declared (100 mrem at site boundary). PASS is generally not used because of time lag. If there were a situation where a release was planned due to critical repairs it is possible that the nuclide data from the PASS could be used to refine the dose estimated.	Offsite dose monitored via site survey. Equipment designed to sense presence of I^{131} using charcoal cartridges and a portable scaler (capable of pulse height analysis) connected to a sodium iodide detector.
SL1 & SL2	Offsite dose projections are determined from one or more indications relating to a radioactive release in progress or the potential for release (from CTMT breach). Indicators used include: effluent monitors (plant vent, steam jet air ejector, other monitored release point monitors), CHRRM, grab samples, field monitoring team measurements (dose rate, air sample), or default estimates based on plant conditions. Offsite dose estimates (computer and/or hand calculations) are used to estimate dose from release source terms. All means used are capable of providing estimates within about 30 minutes or less, once initiated. PASS is not used in any offsite dose determination.	There are no special systems used to track offsite releases. Field monitoring teams physically measure offsite dose and define the physical parameters of a plume.
WSES	Projections based on rad monitors and field team measurements.	Field team establishes gross activity at various offsite locations
SONGS	Initial projections by using process monitors and by using an NG/I ratio estimates. Note that SCE does not use hydrogen measurements to determine core damage for use in developing offsite dose projections	- Offsite Dose Monitoring - Field Team data/readings (Air sample data can be read in the Emergency Operations Facility to determine I^{131} concentration). - PIC (Pressurized Ion Chamber)
PVNGS	Projections are based on radiological monitoring system consisting of specific radiation monitors throughout the plant. PASS is not used.	Field team data/readings
ANO-2	Uses radiological dose assessment computer system (RDACS) for projecting and trending offsite releases. Code considers releases from 11 vent pathway SPING monitors and the meteorological tower. If release is unmonitored or field team results are very different from projections, additional data may be sought after. Procedures allow use of source data from local grab samples at SPINGs or via the PASS.	Field monitoring teams establish gross activity at various offsite locations. Field teams can detect the presence of I^{131} with a "go/no go" assessment in the field. Air sample data can be read in the Emergency Operations Facility to determine specific I^{131} concentrations.
MP-2		
PALISADES		

APPENDIX E

CURRENT USE OF PASS IN CORE DAMAGE ASSESSMENT METHODOLOGY

Role of PASS in Core Damage Assessment									
	FCS	SL1	SL2	WSES	SONGS	PVNGS	ANO-2	MP-2	PALISADES
CEOG CDAM used	√	√	√	√	-	√	Note 1	√	√
Modified CDAM	-	-	-	-	√	-	√	-	-
CDA supports EALs	Note 2	Note 2	Note 2	Note 2	**	Note 2	*	No	No
PASS sample supports PARs	No	No	No	No	No	No	No	No	No
CDA Levels used for PAR decision criteria	Note 4	Note 3	Note 3	Note 3	Note 3	Note 3	>10% FF	Note 3	Note 5
* low fuel damage limit only									

Notes:

1. If there is a situation that is likely to result in fuel overheat then general emergency is declared a 2 mile radius and a 5 mile downwind PAR is recommended. If FF is greater than 10% with additional criteria, then a 5 mile radius and 10 mile downward PAR is recommended.
2. Breach of fission product barrier (Assessment may be made via NSS).
3. PARs not directly tied to PASS CDA.
4. PARs based on dose rate in vicinity of specified instruments.
5. PARs graded based on FF % in range of 10 to 50% FF (based on "in-plant" instruments).