Project Number 694
WCAP-14986-A, Rev. 2

OG-00-084

August 7, 2000

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Chief, Information Management Branch,
Division of Inspection and Support Programs

Subject: Westinghouse Owners Group
Transmittal of Approved Topical Report: WCAP-14986-A, Rev. 2 (Non-Proprietary), "Westinghouse Owners Group Post Accident Sampling System Requirements: A Technical Basis" (MUHP-3035)

Reference:
1) Westinghouse Owners Group Letter, OG-98-108, L.F. Liberatori to Document Control Desk, "Transmittal of Reports: WCAP-14986-P, Rev. 1 (Proprietary) and WCAP-14987-NP, Rev. 1 (Non-Proprietary), Entitled 'Westinghouse Owners Group Post Accident Sampling System Requirements: A Technical Basis'," October 26, 1998.

2) Westinghouse Owners Group Letter, OG-00-025, K. Jacobs to Document Control Desk, "Transmittal of Revised Pages for WCAP-14986, Rev. 1, 'Westinghouse Owners Group Post Accident Sampling System Requirements: A Technical Basis'," April 10, 2000.

3) NRC Letter, S.A. Richards to K. Jacobs, "Safety Evaluation Related to Topical Report WCAP-14986, Rev. 1, 'Westinghouse Owners Group Post Accident Sampling System Requirements (TAC No. MA4176), June 14, 2000.

This letter transmits twelve (12) copies of WCAP-14986-A, Rev. 2 (Non-Proprietary), "Westinghouse Owners Group Post Accident Sampling System Requirements: A Technical Basis." There is no proprietary version. This approved version, as signified by the '-A,' designation is being transmitted in accordance with the procedures established in NUREG-0390. As such, WCAP-14986-A, Rev. 2 incorporates the NRC Safety Evaluation Report (SER) and transmittal letter (Reference 3), which accepted this topical report for eliminating PASS from the licensing basis for Westinghouse nuclear power plants.

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If you require further information, feel free to contact Mr. Ken Vavrek in the Westinghouse Owners Group Project Office at 412-374-4302.

Very truly yours,

Karl Jacobs, Chairman
Westinghouse Owners Group

enclosures

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Westinghouse Non-Proprietary Class 3

WCAP-14986-A
Revision 2

Westinghouse Owners Group Post Accident Sampling System Requirements: A Technical Basis

Westinghouse Electric Company LLC

WCAP-14986-A
Revision 2

Westinghouse Owners Group Post Accident Sampling System Requirements: A Technical Basis

Robert J. Lutz, Jr.

July 2000

Prepared by Westinghouse Electric Company LLC for use by members of the Westinghouse Owners Group. Work performed under Project numbers MUHP-3033 and MUHP-3035.

Westinghouse Electric Company LLC
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 14, 2000

Mr. Karl Jacobs, Chairman
Westinghouse Owners Group
Indian Point Unit 2
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Buchanan, NY 10511

SUBJECT: SAFETY EVALUATION RELATED TO TOPICAL REPORT WCAP-14986, REVISION 1, "WESTINGHOUSE OWNERS GROUP POST ACCIDENT SAMPLING SYSTEM REQUIREMENTS" (TAC NO. MA4176)

Dear Mr. Jacobs:

By letter dated October 27, 1998 (OG-98-108), the Westinghouse Owners Group (WOG) submitted Topical Report WCAP-14986-P, Revision 1, "Post Accident Sampling System Requirements: A Technical Basis," for NRC staff's review to eliminate requirements on the post accident sampling system (PASS) for Westinghouse nuclear power plants (NPP). The WOG supplemented its application with letters dated April 28, 1999 (OG-99-041), and April 10, 2000 (OG-00-025), that (1) provided responses to a request for additional information, and (2) revised the topical report, respectively. The proprietary information designation was removed from the topical report in the Westinghouse letter dated May 22, 2000 (NSBU-NRC-00-5971).

The enclosed safety evaluation addresses the staff's review of WCAP-14986, Revision 1, for Westinghouse NPP. The staff concluded that the topical report provided a basis to eliminate the PASS as a required system for sampling the 15 parameters that are listed in Section 4 of the safety evaluation. In doing this, the staff also identified four licensee required actions (LRAs), in Section 4.1 of the safety evaluation, that must be fulfilled by a licensee of a Westinghouse NPP that would eliminate PASS in accordance with WCAP-14986 and the safety evaluation. In eliminating PASS, licensees do not have to incorporate the core damage assessment methodology (CDAM) in WCAP-14696, "Westinghouse Owners Group Core Damage Assessment Guidance," into their procedures, but they would need to assess the impact of elimination of PASS on their existing CDAM. This WCAP was approved in our letter of September 2, 1999, to the WOG. See Section 5.0 of the enclosed Safety Evaluation.

Because some licensees have the PASS in their emergency plans (EP) and may want to remove the system from the plan, the third LRA concerns the licensee's determination of the effect of eliminating PASS on the effectiveness of the EP. Based on the enclosed safety evaluation, the staff concludes that eliminating the PASS for sampling the 15 parameters listed in the safety evaluation should not decrease the effectiveness of the EP; however, the licensee must make its own independent determination as to the effect of eliminating the PASS on the effectiveness of its plant-specific EP before the system may be removed from the plan. If a licensee should determine that the effectiveness of the EP is not decreased, then the removal of the PASS would not require staff approval in accordance with 10 CFR 50.54(q).

As stated in the safety evaluation, the staff concludes, based upon the justification provided in WCAP-14986, that there is reasonable assurance that the health and safety of the public will

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WOG PROJECT OFFICE

Mr. Karl Jacobs - 2 - June 14, 2000

not be endangered by operation of Westinghouse NPP without PASS. Therefore, it is acceptable to eliminate PASS from the licensing basis for the Westinghouse NPP.

The NRC requests that the WOG publish an accepted version of the revised WCAP-14986 within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed safety evaluation between the title page and the abstract, remove all proprietary designations from the topical report, and add an -A (designating accepted) following the report identification number (i.e., WCAP-14986-A). The accepted version shall also incorporate the expanded paragraph on containment sump pH in Section 3.13 of the enclosed safety evaluation. Our approval of the topical report is contingent on the removal of all proprietary designations from the topical report.

If the NRC's criteria or regulations change so that its conclusion in this letter, that the topical report is acceptable, is invalidated, WOG and/or the applicant referencing the topical report will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

Sincerely,

Stuart A. Richards, Director
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Safety Evaluation

cc w/encl: See next page

Westinghouse Owners Group

Project No. 694

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO WCAP-14986, "WESTINGHOUSE OWNERS GROUP

POST ACCIDENT SAMPLING SYSTEM REQUIREMENTS"

WESTINGHOUSE OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION

In its letter dated October 27, 1998, the Westinghouse Owners Group (WOG) submitted Topical Report WCAP-14986-P, Revision 1, "Post Accident Sampling System Requirements: A Technical Basis," to be reviewed by the staff for eliminating PASS requirements from Westinghouse pressurized water reactor nuclear power plants (NPP). The "-P" designates that the topical report contains proprietary information. The topical report was revised in the letter dated April 10, 2000, and the proprietary information designation was removed in the letter of May 22, 2000. Therefore, the topical report is now WCAP-14986, Revision 1 (i.e., WCAP-14986, or the topical report). The WOG also responded to a request for additional information in its letter of April 28, 1999.

WCAP-14986 evaluated the post accident sampling system (PASS) requirements to determine their contribution to plant safety and accident recovery. The topical report considered the progression and consequences of core damage accidents and assessed the accident progression with respect to plant abnormal and emergency operating procedures, severe accident management guidance, and emergency plans. WCAP-14986 concluded that many of the current PASS samples specified in NUREG-0737, "Clarification of TMI Action Plan Requirements," may be eliminated (i.e., remove the requirements to perform the sampling from the licensing basis), or the time for taking and analyzing the sample may be changed. For some sample types, the WOG recommended that the capability be maintained for long term recovery purposes, but with the PASS not being required within the licensing basis of the Westinghouse NPP. With PASS outside the licensing basis, there would be no requirements on the licensees to maintain and use the PASS; however, the licensee may elect to keep the PASS in the plant and use the system as long as it does not adversely affect safety-related systems.

Specifically, the WOG recommended in WCAP-14986 the following:

- Eliminate PASS sampling of reactor coolant system (RCS) dissolved gases.
- Eliminate PASS sampling of RCS hydrogen.
- Eliminate PASS sampling of RCS oxygen.

- Eliminate PASS sampling of RCS pH.
- Eliminate PASS sampling of RCS chlorides.
- Change the time required for obtaining and analyzing RCS boron from 3 hours to 8 hours. Change the accuracy criteria to: (1) 10% at a 1 sigma uncertainty for values above 1500 ppm, and (2) 20% of 1500 ppm, or 300 ppm, for values below 1500 ppm.
- Eliminate PASS sampling of RCS conductivity.
- Eliminate PASS sampling of radionuclides in the RCS.
- Eliminate PASS sampling of containment hydrogen.
- Eliminate PASS sampling of containment oxygen.
- Eliminate PASS sampling of radionuclides in the containment atmosphere.
- Eliminate PASS sampling of containment sump pH for plants that do not use brackish or salt water for the ultimate heat sink or have more than a single barrier between the cooling water and the containment or which have passive pH control.
- Eliminate PASS sampling of chlorides in the containment sump.
- Eliminate PASS sampling of boron in the containment sump.
- Eliminate PASS sampling of radionuclides in the containment sump.

2.0 BACKGROUND

The need for a PASS was one of the findings endorsed by the NRC following the accident at the Three Mile Island (TMI) plant. The NRC specified that all licensed plants have the capability of obtaining and analyzing post-accident samples of the reactor coolant and containment atmosphere within specified times, without causing a radiation exposure to any individual that exceeds 5 rem to the whole body or 75 rem to the extremities. Detailed criteria for the PASS are specified in Section II.B.3 of NUREG-0737 including the following:

> The licensee and applicant shall establish an onsite radiological and chemical analysis capability to provide, within a three-hour time frame, quantification of the following:
>
> a) Certain radionuclides in the reactor coolant and containment atmosphere
>
> b) Hydrogen levels in the containment atmosphere
>
> c) Dissolved gases (e.g., hydrogen), chloride, and boron concentration of liquids

The TMI-related recommendations specified in NUREG-0737 were subsequently incorporated into 10 CFR 50.34(f)(2)(viii). However, this rule applied only to applications pending at that time (i.e., Perkins Nuclear Station, Units 1, 2, and 3; Allens Creek Nuclear Generating Station, Unit 1; Pebble Springs Nuclear Plant, Units 1 and 2; Black Fox Station, Units 1 and 2; Skagit/Hanford Nuclear Power Project, Units 1 and 2; and Offshore Power Systems).

On March 17, 1982, the NRC issued Generic Letter (GL) 82-05, "Post-TMI Requirements," in which the NRC requested that licensees establish a firm schedule for implementing post-accident sampling. On November 1, 1983, the NRC issued GL 83-36 and GL 83-37, "Technical Specifications," which provided guidance on how to address post-accident sampling in the technical specifications for boiling-water reactors (BWRs) and pressurized-water reactors

(PWRs), respectively. In GL 83-36 and GL 83-37, the NRC indicated that all licensees should establish, implement, and maintain an administrative program that would include training of personnel, procedures for sampling and analyses, and provisions for sampling and analysis equipment. The licensees could elect to reference this program in the administrative controls section of the technical specifications and include its detailed description in the plant operation manuals. However, the recommendations described in Section II.B.3 of NUREG-0737 were imposed as requirements for the majority of operating plants through license conditions or by orders.

Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" (Revision 3, 1983), described acceptable means for licensees to comply with the Commission's regulations (Criteria 13, 19, and 64 of Appendix A to 10 CFR Part 50) to provide instrumentation to monitor plant variables and systems during and following an accident. Regulatory Guide 1.97 included a list of variables to be monitored which included the samples specified in NUREG-0737 and the following additional samples:

- pH in the RCS
- Boron, pH, chlorides, and radionuclides in the containment sump

Since these criteria for PASS have been issued, the NRC has performed three generic evaluations pertinent to the staff's evaluation of WCAP-14986, which are discussed below.

In the mid 1980s, the staff had a contractor review regulatory requirements that may have marginal importance to risk. One of the issues reviewed was the NUREG-0737 criteria for PASS. The conclusion reported in NUREG/CR-4330, "Review of Light Water Reactor Regulatory Requirements" (dated May 1987), was that several of the PASS criteria could be relaxed without impacting safety; however, the staff did not take action to modify the PASS criteria based upon the contractor's conclusions.

In 1993, during its review of licensing issues pertaining to evolutionary and advanced light water reactors, the staff evaluated requirements for PASS specified in 10 CFR 50.34(f)(2)(viii). The staff recommended to the Commission in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-water Reactor (AWLR) Designs," (dated April 2, 1993), that: (1) elimination of hydrogen analysis of containment atmosphere samples is appropriate, given that safety-grade hydrogen monitoring instrumentation will be installed; (2) relaxation of dissolved gas (including dissolved hydrogen) sampling time to 24 hours is appropriate; (3) elimination of the mandatory requirement for chloride samples is appropriate; (4) relaxation of the boron sampling time to 8 hours after an accident is appropriate; and (5) relaxation of the sampling time for radionuclides (used to determine the degree of core damage) to 24 hours is appropriate.

In addition, in 1993, the staff evaluated the Combustion Engineering Owners Group Topical Report CEN-415, "Modifications of Post Accident Sampling System Requirements," (Revision 1, December 1991). In a letter dated April 12, 1993, the NRC approved: (1) deletion of pH measurement in the containment sump, (2) deletion of hydrogen sampling of the containment atmosphere, (3) deletion of sampling for iodine (if core damage assessment procedures are

based on samples of xenon or krypton activities), and (4) deletion of oxygen analysis of reactor coolant.

Finally, in parallel with review of WCAP-14986, the staff also reviewed a Combustion Engineering Owners Group Topical Report (CE NPSD-1157, "Technical Justification for the Elimination of the Post-Accident Sampling System from the Plant Design and Licensing Bases for CEOG Utilities") which requested similar changes to PASS requirements for Combustion Engineering pressurized water reactors.

The staff considered the conclusions (and the basis for the conclusions) from these generic evaluations as part of its review of WCAP-14986.

3.0 EVALUATION

The NRC staff's review of the technical basis for each of the changes to PASS proposed in WCAP-14986 is discussed below.

3.1 Eliminate Pass Sampling of RCS Dissolved Gases

Dissolved gas sampling is specified in NUREG-0737 and Regulatory Guide 1.97; however, NUREG/CR-4330 suggests that it could be eliminated provided that vessel head gas vents and a reactor vessel level instrumentation system (RVLIS) are installed.

The main purpose of sampling for dissolved gases is to identify the potential of void formation in the vessel dome (and at the top of the steam generator U tubes) from dissolved gases when depressurizing, or even uncovering the core in case natural circulation needs to be used for decay heat removal.

Because RVLIS provides an indication of water level and the vessel head vent (which is safety grade) can easily vent non-condensable gases, both diagnosis and remediation is available. In addition, for plants not equipped with automated gas sampling systems, the delay between sampling and the availability of the results is long and of no practical significance in accident management.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of RCS dissolved gases is acceptable.

3.2 Eliminate PASS Sampling of RCS Hydrogen

PASS sampling of the reactor coolant for measurement of dissolved hydrogen is specified in NUREG-0737 and Regulatory Guide 1.97.

The main purpose of hydrogen sampling is to identify the potential of void formation in the vessel dome and the top of the U tubes in the steam generators or even uncovering the core when depressurizing. In addition, the amount of the dissolved hydrogen could act as a surrogate indicator for dissolved fission product and non-condensable gases. As in the case of dissolved gases, the vessel head vent and the RVLIS system can be used to both identify and

vent non-condensable gases from the RCS when depressurizing in order to establish natural circulation in the RCS.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of RCS hydrogen is acceptable.

3.3 Eliminate PASS Sampling of RCS Oxygen

PASS sampling of the reactor coolant for measurement of oxygen is only recommended in NUREG-0737, but is specified in Regulatory Guide 1.97, whenever the RCS concentration of chlorides exceeds 1.5 ppm.

High concentrations of oxygen in the RCS can enhance stress corrosion cracking of stainless steel components caused by the presence of chlorides. However, the pH of the reactor coolant is usually adjusted by the automatic addition of a buffering solution to where stress corrosion cracking cannot occur, even with the dissolved oxygen present. The buffering is done by the addition of pH control through containment spray or by the addition of trisodium phosphate in the containment sump and the recirculation of water from the containment into the reactor coolant.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of RCS oxygen is acceptable.

3.4 Eliminate PASS Sampling of RCS Chlorides

PASS sampling of chlorides in the RCS is specified in NUREG-0737 and Regulatory Guide 1.97.

High concentrations of chlorides in the reactor coolant can cause stress corrosion cracking of stainless steel components in contact with the coolant. Chlorides are introduced into the RCS by the incoming water from external sources containing chlorides. For plants which use cooling water containing chlorides, the operators are aware when the ingress of contaminated water occurs and can take appropriate corrective actions to prevent corrosion damage.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of RCS chlorides is acceptable.

3.5 Eliminate PASS Sampling of RCS pH

PASS measurement of the reactor coolant pH is specified in Regulatory Guide 1.97.

Reactor coolant pH control is important for controlling stress corrosion cracking of stainless steel components and for iodine retention. However, PASS sampling of RCS pH is not needed since in the post-accident environment of Westinghouse NPP, the pH of the reactor coolant and the containment sump are usually adjusted by the automatic addition of a buffering solution via the containment spray system. For ice condenser plants, the ice baskets contain a buffering solution that is released as the ice melts. Other plants use passive means such as baskets of trisodium phosphate in the containment sump to ensure that the pH of the containment sump

water following a design basis loss-of-coolant accident (LOCA) is within specified limits. Also, RCS pH can be satisfactorily estimated by calculations.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of RCS pH is acceptable.

3.6 Change Time Requirement for PASS Sampling RCS Boron from 3 Hours (after decision to do so) to 8 Hours (after plant reaches a stable state) and Relax Accuracy Criteria

PASS sampling of the reactor coolant for measurement of boron is specified in NUREG-0737 and Regulatory Guide 1.97. In addition, the staff recommended in SECY 93-087 that the capability to obtain PASS samples of RCS boron within 8 hours of accident initiation (after plant reaches a stable state) be maintained for advanced light water reactors.

The topical report states that knowledge of boron concentration is required to achieve cold shutdown and requests that boron be measured eight hours after the plant has been placed in a safe and stable state. WOG proposed to rely on emergency operating procedures (EOPs) to achieve such a shutdown. Although WOG recommended in WCAP-14986 that the capability to obtain RCS samples from the PASS system be maintained, WOG stated in a telephone call held August 20, 1999, that there were adequate indications and procedures available to mitigate an accident without obtaining a PASS sample for RCS boron. In its letter of April 10, 2000, WOG revised the topical report to recommend that boron sampling of RCS also be eliminated. The staff finds that RCS boron concentration is an essential parameter for the accident management and achieving a cold shutdown state; however, the plant EOPs provide for adequate boration through the transient and recovery stage such that the PASS measurement is not required.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling for RCS boron is acceptable and, therefore, the accuracy criteria for this measurement is no longer relevant.

3.7 Eliminate PASS Sampling of RCS Conductivity

The PASS sampling of the reactor coolant for measuring conductivity of the coolant is not specified in NUREG-0737, nor Regulatory Guide 1.97.

The measurement of reactor coolant conductivity is only for verifying pH measurements and it was never required by the NRC. Therefore, the staff concludes that the proposal to eliminate PASS sampling for RCS conductivity is acceptable.

3.8 Eliminate PASS Sampling of RCS Radionuclides

For the purposes of this discussion, reactor coolant sump sample analysis capabilities is also applicable for the containment sump sample. PASS sampling of the reactor coolant for measurement of radionuclides is specified in NUREG-0737 and Regulatory Guide 1.97. NUREG-0737 specifies that the PASS have the capability to promptly (i.e., within 3 hours) quantify certain radionuclides that are indicators of the degree of core damage. Furthermore,

Regulatory Guide 1.97 specifies that the isotopic analysis serves the purpose of accident release assessment.

The topical report states that post accident measurement of RCS radionuclides is currently used to perform core damage assessment and to classify fuel damage events at the Alert level for emergency preparedness. In regards to core damage assessment, the topical report states that measurement of radionuclides with PASS is not needed because there are four independent overlapping procedures for estimating core damage; the first three of which do not utilize RCS radionuclide information (and are simpler to perform). The fourth procedure is intended to be a detailed precise methodology for quantifying core damage based upon RCS radionuclide information. The topical report states that there is little expectation that the RCS sample will provide sufficiently accurate information to improve upon assessments performed by the simpler procedures. The topical report states that the core damage assessment procedure should be changed to eliminate the procedure involving radionuclide measurement.

In regards to the use of radionuclide sample information for classifying events involving failed fuel, the topical report states that the event can be classified based upon the recognition of the initiating condition which caused the fuel failure rather than measurement of the degree of fuel failure. Furthermore, the topical report states that other indications of failed fuel, such as letdown radiation monitors, can be correlated to the degree of failed fuel.

The staff considers radionuclide sampling information to be useful in estimating the degree of core damage, but recognizes that there are limitations associated with its use, in particular regarding the time needed to obtain the sample. Therefore, the staff considers it more appropriate for emergency response purposes to estimate the degree of core damage based upon real-time indications.

In addition, the staff considers radionuclide sampling information to be useful in classifying certain type of events (such as reactivity excursion or mechanical damage) which could cause fuel damage without having an indication of overheating on core exit thermocouples. However, the staff agrees with the topical report contention that other indicators of failed fuel, such as letdown radiation monitors (or normal sampling system), can be correlated to the degree of failed fuel. (See Section 4.1, Licensee Required Actions, Items 1 and 2).

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of RCS radionuclides is acceptable.

3.9 Eliminate PASS Sampling of Containment Atmosphere Hydrogen Concentration

PASS sampling of the containment atmosphere for hydrogen measurement is specified in NUREG-0737 and Regulatory Guide 1.97.

WCAP-14986 states that at least one means of obtaining a measurement of the containment hydrogen concentration is required, and that either sampling and analysis of hydrogen using PASS or use of the safety-grade containment on-line hydrogen monitor would be acceptable provided appropriate timing and accuracy needs can be met. The capability to obtain an initial measurement within about 30 minutes of the onset of core damage, and at 10 to 15 minute

intervals thereafter, with an accuracy of plus or minus one volume percent hydrogen concentration, are specified.

The redundant, safety-grade, containment hydrogen concentration monitors are required by 10 CFR 50.44(b)(1) and NUREG-0737 Item II.F.1, and are relied upon to meet the data reporting requirements of 10 CFR Part 50, Appendix E, Section VI.2.a.(i)(4). NUREG-0737 Item II.F.1 specifies that the monitors are to be functional within 30 minutes of the initiation of safety injection. As such, the monitors are expected to be functional prior to generation and release of hydrogen. Regulatory Guide 1.97 specifies that the monitors have a range of 0 to 10 volume percent. The quantity of hydrogen released to containment in most severe accidents would result in concentrations within this range. However, in the event that random or spontaneous ignition does not occur, continued hydrogen production from such mechanisms as core concrete interactions and radiolysis of reactor coolant could result in the concentration exceeding the range of the monitors late in an event. Hydrogen concentration measurements for concentrations greater than 10 volume percent are necessary to support the assessment of the hydrogen combustion threat to containment in the WOG severe accident management guidelines (SAMG). In the absence of this information, licensee severe accident management decision-making would rely on default hydrogen production assumptions contained in the SAMG. Since grab sample analysis provides the only viable means of determining the actual hydrogen concentration once the hydrogen concentration exceeds the range of the monitors, there is value to retaining the capability for long term hydrogen concentration analysis of containment atmosphere grab samples.

The staff concludes that during the early phases of an accident, the safety-grade hydrogen monitors provide an adequate capability for monitoring containment hydrogen concentration and are an acceptable alternative to maintaining the capability to obtain and analyze containment atmosphere samples for hydrogen within 3 hours. Approval of the change regarding PASS sample analysis does not change the requirements contained in 10 CFR 50.44(b)(1), and criteria in NUREG-0737 Item II.F.1, and Regulatory Guide 1.97 regarding the need to establish containment hydrogen concentration monitoring within 30 minutes of the initiation of safety injection. The staff notes that the NRC recently issued a confirmatory order for Arkansas Nuclear One that replaced the requirement to establish hydrogen monitoring within 30 minutes of the initiation of safety injection with a functional requirement that allows the licensee the flexibility to determine the appropriate time limit for providing indication of hydrogen concentration in containment. This same mechanism is available to other licensees who were issued orders in the 1983 time-frame confirming their requirements made in response to NUREG-0737 Item II.F.1. The information provided in Section 5.9 of WCAP-14986 with regard to time requirements, together with consideration of plant-specific emergency action levels, EOPs, and SAMG, can be used by those licensees in establishing the plant-specific time limit. For licensees that were not issued orders confirming their requirements regarding NUREG-0737 Item II.F.1, a different action (other than a confirmatory order) may be appropriate for relief from the timing requirement for establishing post-accident hydrogen monitoring.

In view of the value of sampling the containment atmosphere for hydrogen to complement the information from the hydrogen monitors in the long term (i.e., by confirming the indications from the monitors and providing hydrogen measurements for concentrations outside the range of the monitors), the staff requires that licensees retain a capability for sampling the containment

atmosphere during the later stages of accident response (see Section 4.1, Licensee Required Actions, Item 2), and recommends the analyzing of such samples for hydrogen .

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of containment atmosphere hydrogen concentration is acceptable.

3.10 Eliminate PASS Sampling of Containment Oxygen

PASS sampling of the containment atmosphere for oxygen measurement is specified in Regulatory Guide 1.97.

Containment oxygen measurement serves to ensure that the oxygen level does not reach the limit of deflagration or detonation with the generated hydrogen. Since in the post-accident environment the only source of oxygen is radiolysis of sump water, it is not expected that this source will cause a significant increase of oxygen above the initially existing concentration in the containment atmosphere.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of containment oxygen is acceptable.

3.11 Eliminate PASS Sampling of Radionuclides in the Containment Atmosphere.

PASS sampling of the containment atmosphere for radionuclide measurement is specified in NUREG-0737 and Regulatory Guide 1.97. NUREG-0737 specifies that the PASS have the capability to promptly quantify certain radionuclides that are indicators of the degree of core damage. Furthermore, Regulatory Guide 1.97 specifies that the isotopic analysis serves the purpose of accident release assessment.

PASS measurements of the containment atmosphere radionuclide concentration are used to estimate the degree of core damage and to refine the source term used in dose assessments. In turn, core damage estimates and dose assessments are used in evaluating the type and extent of public protective actions which may be warranted. The topical report states that PASS sampling of containment atmosphere radionuclides can be eliminated because these samples are not representative of the concentration of radionuclides which may be released to the environment. The basis for this conclusion is that the concentration of the radionuclides at the sample point may not be representative of the concentration in containment, the potential for revolitization of fission products upon containment depressurization, plate out of aerosols (e.g., cesium iodide or CsI) in the sample lines, and time delays associated with obtaining, processing and interpreting the sample during non-stable phases of the accident. In addition, the topical report stated that samples of the containment atmosphere could be obtained and analyzed without reliance on the PASS.

The staff recognizes that, as described in Supplement 3 to NUREG-0654, initial protection action recommendations (PARs) should be based upon plant indications of actual or projected core damage. Following this initial PAR, the licensee should continue assessment of the accident to determine whether the PAR should be modified (relaxation of the PAR should not occur until the source of the threat is clearly under control). In NUREG-0654, the NRC indicated that licensees' capability to perform this assessment should include the post accident

sampling capability. Therefore, the staff's evaluation of the topical report's recommendation for elimination of sampling the containment atmosphere for radionuclides focused on the need for this information to support whether initial PARs should be modified.

The staff agrees with the topical report's assessment regarding the limitations associated with obtaining representative samples of the containment atmosphere. The staff considers that these limitations should be taken into account when determining how to utilize the containment atmosphere sample information during an event. However, the staff position is that, due to these limitations, information obtained from PASS samples would not be a primary factor in licensee and offsite emergency response decision making regarding PARs during the early phases of an accident. The public comments received (discussed in the appendix attached to this safety evaluation) on the proposed staff action to eliminate PASS support this position. However, the staff considers that containment atmosphere sample information would provide the public additional confidence that the licensee understood the magnitude of any remaining threat that the accident may pose after plant conditions in the accident have stabilized. Therefore, the staff also concludes that a plan should be developed for sampling the containment atmosphere; however, the staff does not consider it necessary to have dedicated equipment to obtain this sample in a prompt manner. These plans should detail the plant's existing sampling capabilities and what actions (e.g., assembling temporary shielding) may be necessary to obtain and analyze highly radioactive samples. (See Section 4.1, Licensee Required Actions, Items 2 and 4).

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of containment atmosphere radionuclides is acceptable.

3.12 Eliminate PASS Sampling of Containment Sump Radionuclides.

Containment sump sampling is discussed in Section 3.8.

3.13 Eliminate Pass Sampling of Containment Sump pH for Plants that do not use Brackish or Salt Water for the Ultimate Heat Sink or have more than a Single Barrier Between the Cooling Water and the Containment or which have a Passive pH Control

PASS sampling of the containment sump for measurement of pH is specified in Regulatory Guide 1.97.

The value of containment sump pH plays an important role in controlling the post-accident chemistry of the containment sump water. If it becomes acidic, it can significantly affect chloride induced stress corrosion cracking of stainless steel components and retention of iodine in sump water. In most cases, the post-accident sump pH is maintained in an alkaline range either by passive pH control or by spray additives. However, there may be some accident sequences when the containment spray is not activated, and sump pH may then become acidic. In these cases, however, its value can be estimated with a sufficient degree of accuracy from the volumes and chemistries of the water incoming from different external sources that represent the major sources of acid. WOG recommended in WCAP-14986 that plants which (1) use brackish or salt water for the ultimate heat sink, (2) do not have more than a single barrier between the cooling water and the containment, and (3) do not have a passive pH control, should maintain pH sample capability. However WOG stated, in a telephone call held

August 20, 1999, that the sump pH can be estimated without obtaining a PASS sample for these plants.

In its letter dated April 10, 2000, WOG revised its topical report to eliminate the requirements for containment sump pH sample capability from PASS. In the telephone call of May 31, 2000, the WOG explained that plants even with brackish or salt water could eliminate PASS sampling for containment sump pH because of the passive or active pH control. Passive or active pH control is trisodium phosphate in the containment sump or sodium hydroxide as a containment spray additive, and all Westinghouse plants have one or the other pH control.

The WOG sent the following paragraph by telecopy to the staff. The following paragraph expands on Section 5.13 on the containment sump pH:

> For plants with passive containment sump pH control, the containment sump pH will be within the acceptable range for iodine retention and for chloride induced stress corrosion cracking, unless additional water (e.g., water addition in SAMG SAG-4 from the demineralized water storage tank) has been added to the containment sump. For plants with active containment sump pH control (typically via the containment spray additive tank), the containment sump pH will be within the acceptable range for iodine retention and for chloride induced stress corrosion cracking if the pH control is activated and no additional water (e.g., water addition in SAMG SAG-4 from the demineralized water storage tank) has been added to the containment sump. For the case where active containment sump pH control is not automatically actuated (e.g., automatic actuation of containment spray for small LOCA events), guidance is available for the plant engineering staff (see Section A.1.7 of Appendix A) to determine the need for pH adjustment via other means (e.g., manual actuation of containment spray). For the case of water addition to the containment sump, the plant engineering staff guidance described in Section A.1.7 of Appendix A recommends that the sump pH can be approximated from calculations of the containment sump level indication and the sources of water in the containment sump and the chemical composition of the water.

The WOG stated that it will add this paragraph to WCAP-14986 to expand its justification for eliminating the PASS sampling of containment sump pH. The staff considers containment sump sampling to be useful for confirming sump pH calculations and that unaccounted for acid sources have been sufficiently neutralized and, therefore, the staff requires that licensees maintain the capability to sample the containment sump (see Section 4.1, Licensee Required Actions, Item 2) and recommends that the licensees also maintain the capability to analyze the sample for pH.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of containment sump pH is acceptable.

3.14 Eliminate PASS Sampling of Containment Sump Chlorides

PASS sampling and measurement of the containment sump for chlorides is specified in Regulatory Guide 1.97.

High concentration of chlorides in the containment sump can cause stress corrosion cracking of stainless steel components and affect retention of iodine in containment sump water. For plants with fresh water cooling systems, the problem is minimal; but for the plants with brackish water (with a single barrier between the cooling water and the containment and without pH control) it may be a significant issue. However, the volumes and chloride concentrations of the incoming water from different sources are known and the resulting concentration of chlorides in the sump water can be estimated with a sufficient degree of accuracy.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of containment sump chlorides is acceptable.

3.15 Eliminate Pass Sampling of Containment Sump Boron

Sump boron concentration sampling and measurement is specified in Regulatory Guide 1.97. This sampling was not addressed in SECY 93-087.

The purpose of measuring boron concentration in the containment sump is to assure reactor subcriticality should sump water be used in the recirculation mode to cool the core. The refueling water storage tank (RWST) and the accumulator water have sufficient boron concentration to assure subcriticality at any time in the fuel cycle. For ice condenser containment plants, there is sufficient boron added to the ice that the melt has the concentration of the RWST. However, in instances where unborated water is introduced in the containment for emergency core cooling, the sump boron density will be lower. However, the sump level (and the corresponding amount of water) is known. Therefore, knowing the source of the added water will allow the boron concentration to be estimated. Therefore, the staff concludes that elimination of boron sampling of the containment sump is acceptable.

Based on the above, the staff concludes that the proposal to eliminate PASS sampling of containment sump boron is acceptable.

4.0 <u>SUMMARY</u>

The staff concludes that WCAP-14986 provides a sufficient technical basis to eliminate the following PASS criteria specified in NUREG-0737 and Regulatory Guide 1.97:

1. RCS dissolved gases
2. RCS hydrogen
3. RCS oxygen
4. RCS chlorides
5. RCS pH
6. RCS boron
7. RCS conductivity
8. RCS radionuclides
9. Containment atmosphere hydrogen
10. Containment atmosphere oxygen
11. Containment atmosphere radionuclides
12. Containment sump radionuclides
13. Containment sump pH

- 13 -

14. Containment sump chlorides
15. Containment sump boron

4.1 Licensee Required Actions

The staff has identified the following licensee required actions (as discussed in the above sections) that must be fulfilled by a licensee that would eliminate PASS for sampling the above 15 parameters in accordance with WCAP-14986 and the safety evaluation:

1. Establish a capability for classifying fuel damage events at the Alert level threshold (typically this is 300 microcuries per ml dose equivalent iodine). This capability may utilize the normal sampling system or correlations of sampling or letdown line dose rates to coolant concentrations.

2. Develop contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. These plans should detail the plant's existing sampling capabilities and what actions (e.g., assembling temporary shielding) may be necessary to obtain and analyze highly radioactive samples. The contingency plans do not have to be demonstrated. Because these are contingency plans, the staff concludes that, in accordance with 10 CFR 50.47 and Appendix E to 10 CFR Part 50 for emergency plans, these contingency plans must be available to be used by the licensees during an accident; however, these contingency plans do not have to be carried out in emergency plan drills or exercises.

3. The staff does not consider that changes as discussed in this topical report will result in a decrease in the effectiveness of the emergency plan, however the licensee must determine for its own plant(s) that no decrease in the effectiveness of the emergency plans will result from the removal/downgrade of the PASS.

4. Licensees will maintain offsite capability to monitor radioactive iodines.

For containment hydrogen concentrations, containment hydrogen monitors required by 10 CFR 50.44(b)(1) may not be eliminated because they are required by the regulations. Although no longer a requirement, the staff recommends that licensees maintain the capability to analyze a containment atmosphere sample for hydrogen during the later stages of accident response in order to support SAMG. For containment sump pH, the staff also recommends that the licensees maintain the capability to analyze the sump water for pH. The licensees maintaining the capability to take a sample from the containment atmosphere and sump is LRA 2 above.

Because some licensees have the PASS in their emergency plans (EP) and may want to remove the system from the plan, the third licensee required action above concerns the effect of eliminating PASS on the effectiveness of the EP. Based on the safety evaluation, the staff concludes that eliminating the PASS for sampling the 15 parameters listed in the safety evaluation should not decrease the effectiveness of the EP; however, the licensee must also make an independent determination on its own as to the effect of eliminating the PASS on the effectiveness of the EP before the system may be removed from the plan. If a licensee should determine that the effectiveness of the EP is not decreased, then the removal of the PASS would not require staff approval in accordance with 10 CFR 50.54(q).

Some licensees have the PASS in their Technical Specifications (TSs). Removing PASS from the TSs is a license amendment that requires staff approval in accordance with 10 CFR 50.90. In submitting a license amendment, the licensees must address LRAs 1, 2, and 4, and describe how and when they will be implemented at the plants. The description is expected to be a reference to the applicable SAMG for the plant(s). The details may be reviewed by the staff in an inspection. The time to complete these LRAs would be included in (1) the time to implement the approved amendment with the implementation date specified in the license amendment or (2) regulatory commitments specifying the LRA implementation dates, in accordance with Nuclear Energy Institute (NEI), "Guidelines for Managing NRC Commitments," dated June 9, 1995, in which safety significant changes to such commitments to NRC are discussed with NRC before the change is made. (See the amendments for the application dated July 14, 1999, for Arkansas Nuclear One, Units 1 and 2, TAC Nos. MA6062 and MA6063, respectively, after it is issued.)

With licensees implementing the above LRAs, the staff concludes, based upon the justification provided in WCAP-14986, that there is reasonable assurance that the health and safety of the public will not be endangered by operation of Westinghouse NPP without PASS.

5.0 CORE DAMAGE ASSESSMENT METHODOLOGY

In the letter of November 22, 1996, the WOG submitted Topical Report WCAP-14696, "Westinghouse Owners Group Core Damage Assessment Guidance," for NRC review. In the topical report, a revised methodology was described that would be used by licensee emergency response organization staff for estimating the extent of core damage that may have occurred during an accident at a Westinghouse nuclear power plant. The revised methodology is a revised calculational technique for estimating core damage which relies on real-time plant indications rather than samples of plant fluids. The revised post-accident core damage assessment methodology (CDAM) in WCAP-14696 replaces the methodology approved by the staff in 1984. The 1984 methodology was revised for two major reasons: (1) the current methodology relies on radionuclide samples and does not effectively support emergency response decisionmaking due to the significant time delay in obtaining and analyzing these samples using the post-accident sampling system (PASS), and (2) the methodology does not reflect the latest understanding of fission product behavior, particularly the sequence-specific nature of fission product retention and hydrogen holdup in the RCS, and fission product deposition in the containment and sample lines.

In the staff's letter of September 2, 1999, the staff approved WCAP-14696 for use by Westinghouse plants for core damage assessment. Because the staff concludes above, based upon the justification provided in WCAP-14986, that there is reasonable assurance that the health and safety of the public will not be endangered by operation of Westinghouse NPP without PASS without also concluding that the implementation of WCAP-14696 was necessary, the staff concludes that it is acceptable for licensees to eliminate PASS from the licensing basis for the Westinghouse NPP without incorporating the core damage assessment methodology in WCAP-14696 into its procedures; however, the licensees should assess the impact of elimination of PASS on their existing CDAM.

6.0 CONCLUSION

The staff concludes, based upon the justification provided in WCAP-14986, that there is reasonable assurance that the health and safety of the public will not be endangered by operation of Westinghouse NPP without PASS. Therefore, the staff concludes that it is acceptable for licensees to eliminate PASS from the licensing basis for the Westinghouse NPP. In eliminating PASS, the licensees do not have to incorporate the core damage assessment methodology in WCAP-14696 into its procedures, but they would need to assess the impact of elimination of PASS on their existing CDAM.

Attachment: Analysis of Public Comments

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APPENDIX

Analysis of Public Comments and Staff Response

In a notice published in the *Federal Reigster* on November 24, 1999 (64 FR 66213), the NRC requested comments on its pending action to approve two industry-developed topical reports concerning the elimination of the PASS. The NRC received 19 comment letters. Nine letters were from nuclear power plant utilities (supporting the proposed action), six letters were from State government organizations (four supporting and two opposing the proposed action), two letters were from private citizens (one supporting and one opposing the proposed action), one from an industry representative (supporting the proposed action) and one from the Federal Emergency Management Agency (supporting the proposed action). The staff grouped specific individual comments from each of the letters into a number of issue categories. These issues, the comments pertinent to the issue, and NRC response to insights provided in the comments are described below. Following the analysis of specific individual comments, a summary of all comments (general and specific) and the staff's response is provided.

1. Analysis of Specific Comments

Accuracy of PASS Results

One commenter agreed with the topical reports' contention that physical phenomena such as plateout and deposition in sample lines may cause PASS samples to underpredict the fission product inventories that are potentially available for release. A second commenter disagreed with the topical reports' contention on this issue, in particular regarding plateout of iodine in the sample lines and stated that an equilibrium will be reached (deposition equal to re-evolution) after the containment atmosphere has been circulated through the sample lines for a period of time.

The NRC considers the difficulty of obtaining representative samples to be a major shortcoming of the PASS system. The deposition of iodine is particularly problematic since iodine is the best indicator (from the PASS) for evaluating core damage and potential significance of health consequences from a release of the containment atmosphere. The amount of deposition of iodine will be a function of its chemical form. At the time that the PASS criteria were developed in NUREG-0737, the majority of iodine in the containment atmosphere from a potential severe accident was believed to be in elemental form. Since that time, severe accident research has shown that the chemical form of iodine is expected to primarily be Cesium Iodide (CsI) (as an aerosol). Collection of correct samples of the CsI aerosols poses significant problems. There will be a tendency for the particles to deposit on the cooler walls of sampling lines due to thermophoresis and Stefan flow, if steam is present. All these mechanisms will be present at all times during sampling operation and it is not expected that an equilibrium state between deposition and removal of the CsI aerosols will ever be reached.

Alternate Sampling Capability

Two commenters agreed with the topical reports' contention that samples could be obtained from non-PASS systems if an accident occurred. One commenter disagreed with the topical reports' contention that samples could be obtained from non-PASS systems if an accident

occurred. The opposing commenter stated that there would be problems related to elevated hydrogen for rigging a sampling method. Furthermore, the commenter stated that any licensee requesting PASS elimination should be required to explain how they would accomplish containment atmosphere sampling with less personnel exposure than if they had a PASS.

The NRC is basing its decision on the acceptability of the proposal to eliminate PASS on the benefit that the information obtained from PASS would provide in accident management and emergency response. If this information was considered to be necessary and, therefore, planned to be obtained shortly after a severe accident, then a PASS would be prudent to ensure that samples could be taken promptly and exposure minimized. However, as described further in the summary to this Appendix, the information is not considered to be beneficial for accident management or emergency response. Therefore, there is considered to be sufficient time to establish an alternate sampling capability if samples were considered to be beneficial in the longer term.

Boron Sampling

One commenter disagreed with the topical reports' contention that boron sampling was not needed.

The NRC considers there are sufficient sources of borated water for injection by safety systems. Unborated water sources would only be used in an extremely unlikely circumstance and the use of unborated core cooling water would be balanced with the diminished potential for recriticality (given the core configuration). Furthermore, instruments are available for monitoring any potential recriticality. Knowledge of boron concentration is not a prerequisite of performing emergency operating procedures (EOP) or severe accident management (SAM) procedures.

Core Damage Assessment

Three commenters agreed with the topical reports' contention that PASS was not needed for performing core damage assessment (CDA). Two commenters raised concerns with the elimination of the use of PASS measurements for assessing the degree of core damage. The comments in support of the topical reports stated that other indicators exist which can be used for CDA. The comments disagreeing with the topical reports described the following shortcomings associated with these alternative indications.

- Radiochemical analysis of the coolant, containment sump and containment atmosphere is the most accurate method for performing CDA.

- A limitation of performing CDA based upon containment radiation monitor indication is that it is based upon the radiation monitor response to an assumed mixture of radionuclides. Since the nuclide mix varies greatly from one accident scenario to the next, the actual monitor response may vary by orders of magnitude.

- A limitation of performing CDA based upon core exit thermocouples (CETs) is that CETs cannot be used to determine whether fuel overheat or pellet melting has occurred.
- A limitation of performing CDA based upon hydrogen monitor readings is it can only assess whether the fuel is overheating (not whether the fuel has melted) and there are uncertainties associated with the hydrogen generation rate and mixing of the hydrogen in containment.

The staff recognizes that there are limitations with the individual indications used for CDA which is why current guidance relies on a number of instrument indications to diagnose and evaluate core damage. The staff agrees that radiochemical analysis is more accurate than other available indications but it too has limitations. At the time of PASS design, the iodine chemical form was assumed to be predominantly in elemental gaseous form (91 percent). The staff's current understanding is documented in NUREG/CR-5732, which indicates that iodine entering the containment is at least 95 percent particulate CsI. Once the iodine enters containment, however, additional reactions are likely to occur. In an aqueous environment, as expected for LWRs, iodine is expected to dissolve in water pools or plateout on wet surfaces. This can bias the radionuclide samples obtained from PASS and lead to underestimates of the extent of core damage.

The staff agrees that the nuclide mix varies greatly from one accident scenario to the next which affects radiation monitor response. Revised CDA guidance relies on CETs, RCS pressure and containment spray system status to sufficiently narrow the accident scenario being assessed and the expected variation in the nuclide mix.

The approach for converting instrument readings into core damage estimates is consistent with the current understanding of clad and fuel damage characteristics, and accounts for fission product and hydrogen retention/holdup in an approximate fashion. Specifically, containment radiation monitoring readings are compared to plant-specific radiation levels for 100 percent clad damage or fuel over-temperature damage, CET readings are compared to values typically associated with clad damage and fuel over-temperature damage, and containment hydrogen concentration is compared to the amount expected in containment for 100 percent over-temperature damage. CET readings that exceed the setpoints or the operating limits of the thermocouples are interpreted as core damage in that region of the core. The core damage estimates derived separately from different indicators (containment radiation, CET, and containment hydrogen concentration readings) are compared and reconciled, thereby improving the confidence in the core damage estimate.

The staff has concluded that the revised CDA guidance, that does not rely on PASS, provides the capability to assess the degree of core damage with a sufficient level of accuracy and timeliness to support emergency response decisionmaking. The revised guideline represented an improvement over the existing methodology which relied on PASS sampling. It is both simpler and more timely, and accounts for improved understanding of fission product behavior inside containment. By making core damage information available earlier in an event, such that it can be used to refine dose assessments and confirm or extend initial protective action recommendations, implementation of the revised CDA guidance should increase the effectiveness of the emergency response organization.

Dose Assessment

Two commenters agreed with the topical reports' contention that PASS was not needed for performing dose assessment. One commenter raised concerns with the elimination of the use of PASS measurements as inputs for dose assessments. The comments in support of eliminating this PASS measurement stated that installed instrumentation, which provides real time information from diverse parameters, is much better than PASS samples and that computer models although useful need to be verified by offsite field team measurements. The commenter disagreeing with the topical reports stated that field team measurements have inaccuracies associated with atmospheric transport, field team measurements may not be timely, and that there is a large uncertainty associated with source term estimate based upon in-plant instrumentation.

The NRC expects dose assessments to be timely and accurate in order to support decisions on protective actions for the public. However, the NRC recognizes that there are limitations on the accuracy and timeliness of dose assessments. Therefore, the NRC guidance (reference NUREG-0654, Supplement 3) specifies that initial protective action recommendations should be based upon plant conditions which indicate that there is actual or projected severe core damage. This initial PAR is followed by dose assessments which may be used to expand the area covered by the initial PAR. Initial dose assessments will likely be based upon an assumed source term. This source term may be refined based upon plant indications or core damage assessments. This source term can be further refined based upon offsite field team measurements. (A benefit of using field team measurements is that the source term being estimated is that released from containment rather than the source term in containment which could be altered prior to being released from containment). PASS results are another potential input to refinements to the source term. However, there are concerns with the accuracy of source term estimates based upon PASS because of the potential for the sample not to accurately represent the source term in containment and with the time needed to obtain and analyze these samples.

The NRC believes that PASS results will not have an important role in source term refinements for use in dose assessments because indications such as core exit thermocouple and containment radiation monitor (in conjunction with correlations of these indications to core damage assessments) will be more timely for refining source term estimates and indications such as field team measurements will be more accurate in refining the source term estimates.

Event Classification

Two commenters agreed with the topical reports' contention that PASS was not needed for classifying events.

The NRC agrees that other indications are available for classifying events involving fuel damage and these other indications are available in a more timely manner than PASS.

Hydrogen Measurement

One commenter agreed with the topical reports' contention that PASS was not needed for hydrogen measurement and one commenter raised concerns with elimination of PASS hydrogen monitoring. The comment in support of eliminating hydrogen sample measurement was that the hydrogen monitoring system provided measurement of hydrogen in containment and that this measurement is much quicker than measurements using PASS. The opposing comment was that the hydrogen sample from PASS provides an independent method of determining the hydrogen concentration.

The NRC considers that hydrogen measurement utilizing the PASS system is not needed because the hydrogen monitoring system can provide the same information in a more timely manner. The hydrogen monitoring system is subject to quality assurance requirements and is a redundant system.

Plant Access/Post Accident Leakage/Personnel Exposure

Two commenters provided comments in support of the topical reports contention that elimination of PASS will prevent the potential for restriction of plant access following a PASS sample, will reduce personnel exposure, and will eliminate a potential post accident leakage path.

The NRC agrees with these comments. However, the NRC recognizes that the PASS was to be designed to prevent the potential problems and that the decision to obtain a PASS sample would take into account the benefit of the PASS sample in light of the potential for restricting plant access, exposure of personnel and leakage.

Protective Action Recommendations

Six commenters agreed with the topical reports' contention that development of protective action recommendations will not be affected by the deletion of PASS. Two commenters disagreed with the topical reports' contention on this issue.

The comments in support of eliminating PASS stated that PASS samples are not useful in protective action decisionmaking because these decisions are based on plant indications (real-time monitoring instruments and system operability) and offsite field surveys.

The comments against eliminating PASS stated that offsite officials need to know the actual volume of radioactive material inside of containment (not just the inferred source term) to make additional protective action recommendations.

The NRC considers that PASS may be useful in making subsequent protective action recommendations (or confirming the initial PAR) after the initial protective action recommendation has been made. However, the NRC considers that there is adequate information on the actual (or potential) consequences of a release of radioactive material from field team measurements and containment atmosphere radiation monitors to support assessment of protective action recommendations.

Resources

Four commenters agreed with the topical reports' contention that obtaining and analyzing PASS samples may divert resources from other important emergency response activities.

The NRC does not consider the potential for diverting resources to be a problem because the decision to obtain a PASS sample should be based upon an evaluation of what is the most important activities to perform during the accident.

Severe Accident Management Guides

One commentator disagreed with the topical reports' contention that PASS is no longer required during emergency response in part because of the implementation of Severe Accident Management Guidelines (SAMGs) at nuclear power plants.

The staff does not consider SAMGs to be a replacement for PASS. The SAMGs were intended to provide guidance to the plant operator under severely degraded accident conditions that are outside the plant design and licensing basis. Based on in-plant instrument readings, the core damage state is classified as "in-vessel" or "ex-vessel". Because of PASS limitations, the staff concludes that core damage assessment can be provided with a sufficient level of accuracy and timeliness to support emergency response decision making and SAMG implementation without the PASS. The basis for this conclusion is summarized in the above "Core Damage Assessment" section of this document.

Sump pH

Two commenters disagreed with the topical reports' contention concerning the need for pH measurement from PASS. One commenter stated that knowledge of sump pH will confirm or deny that pH is within design limits and that this information will allow emergency response staff to address pH concerns or to be free to address more pressing concerns (if pH is adequate). The second commenter stated that the NRC should be assured that there is a fool-proof way of buffering recirculation water.

The NRC considers that the chemicals added to the sump water by either trisodium phosphate stored in the sump or sodium hydroxide added to the spray water will provide sufficient buffering action to account for the effect of the major acidic chemicals present in the containment sump after an accident. It is not expected that any unaccounted for acidic substances generated in the sump will significantly lower its pH. Its reevaluation during the accident will not, therefore, be needed, especially since no additional means for pH control will then be available.

System Operation Verification

The commenter stated that PASS data can be used to verify that safety features are operating as designed.

The NRC considers that there may be some benefit to PASS in providing information on the effectiveness of system operation. However, the NRC considers that there are alternative indications which are available in a more timely manner for this purpose.

2. Summary

In addition to the specific comments extracted from the comment letters, many commenters provided a general assessment of the need for PASS. The commenters in support of elimination of PASS stated that PASS information is not used in emergency response and will not adversely affect emergency response, that resources are better used in other areas of emergency response, and that the cost of PASS does not warrant maintaining the system. Commenters opposing elimination of PASS stated that PASS will provide information useful in emergency response.

The NRC appreciates the time and effort taken by all the commenters. Input from the public stakeholders is an important part of the NRC's decision making process. The NRC concludes, as detailed in the body of this safety evaluation, that the PASS has a small benefit in emergency response. The primary benefit is in confirming other indications used to make emergency response decisions. The benefit of PASS is limited by the time needed to obtain the samples and problems with obtaining accurate samples (in particular radioisotopic samples of the containment atmosphere). The NRC concludes that elimination of PASS will not pose a significant hazard to the public and that continued imposition of NRC orders requiring PASS is not warranted.

It is expected that licensees will utilize the industry topical reports and the NRC's safety evaluation in requesting elimination of PASS at their plants. The NRC will provide the public an opportunity to comment on plant requests for elimination of PASS as part of the license amendment process. Therefore, the public will have the opportunity to raise any site-specific concerns related to elimination of PASS at that time.

Westinghouse Owners Group Post Accident Sampling System Requirements: A Technical Basis

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TABLE OF CONTENTS

TABLE OF CONTENTS (cont'd.)

LIST OF TABLES

EXECUTIVE SUMMARY

Following the accident at Three Mile Island Unit 2 (TMI or TMI-2), the Nuclear Regulatory Commission (NRC) promulgated a number of new requirements (summarized in NUREG-0737) based on the "lessons learned" from the post-accident investigations. These 1982 requirements were based on the knowledge of severe accidents in the time frame shortly after the TMI-2 event. Since that time, significant research and analysis has been completed which gives a better understanding of severe accidents. Based on today's understanding of severe accidents, some of these requirements and their bases are no longer valid.

With the development of the Westinghouse Owners Group (WOG) Severe Accident Management Guidance (SAMG) in June 1994, the need for core damage assessment (CDA) and some of the post-accident sampling system (PASS) requirements, from an in-plant recovery perspective, was eliminated. In December of 1999, the Westinghouse Owners Group issued a revised WOG Core Damage Assessment Guideline (CDAG) which eliminated the need for PASS in making core damage assessments for the purpose of providing offsite emergency response recommendations (Ref. 14). Therefore, with the elimination of the need for determining the radionuclide content of reactor coolant, containment sump and containment air samples for core damage assessment, one of the requirements for including radionuclide determinations in PASS is completely eliminated.

The remainder of the current PASS requirements were reviewed to determine their contribution to plant safety and accident recovery. The review considered the progression and consequences of core damage accidents, based on current knowledge and understanding. The review assessed the accident progression with respect to the plant Abnormal and Emergency Operating Procedures, the Severe Accident Management Guidance, and the Emergency Plan.

Based on the current understanding of core damage accidents, the current guidance for bringing the plant to a safe stable state following a core damage accident, and the basis for emergency planning decision making, it is recommended that the current PASS requirements be eliminated.

There are a number of areas where the present plant specific emergency response model should be reviewed to assure that the deletion of PASS does not compromise the ability to use the emergency response tools. This review should include plant specific guidance such as the Core Damage Assessment Guidelines, Emergency Response Guidelines, Severe Accident Management Guidelines, and Emergency Plan Implementing Procedures. In addition, to facilitate long term recovery planning, a conceptual method to obtain samples of reactor coolant liquid, containment sump liquid and containment atmosphere following a core damage accident should be identified. This does not include demonstration that the sample can be obtained.

The technical justification for deletion of PASS also leads to the conclusion that the effectiveness of the plant emergency response is not decreased as a result of PASS elimination. Therefore, the requirements of 10 CFR 50.54(q) are met.

LIST OF ACRONYMS

ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light-Water Reactor
ATWS	Anticipated Transient Without Scram
CDA	Core Damage Assessment
CDAG	Core Damage Assessment Guideline
CEOG	Combustion Engineering Owners Group
CSF	Critical Safety Function
CSFST	Critical Safety Function Status Tree
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
ERG	Emergency Response Guideline
GDC	General Design Criterion
IDCOR	Industry Degraded Core Rule-Making
IPE	Individual Plant Examination
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Program
NEI	Nuclear Energy Institute (and its predecessor, NUMARC)
NRC	Nuclear Regulatory Commission
NSAL	Nuclear Safety Advisory Letter
NSSS	Nuclear Steam Supply System
PASS	Post Accident Sampling System
PORV	Power Operated Relief Valve
PSA	Probabilistic Safety Analysis or Probabilistic Safety Assessment (synonymous with Probabilistic Risk Assessment, or PRA)
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RNO	Result Not Obtained
RTD	Resistance Temperature Detector
RWST	Refueling Water Storage Tank
SER	Safety Evaluation Report
SI	Safety Injection
SG	Steam Generator
SAMG	Severe Accident Management Guidance
TMI	Three Mile Island
TMI-2	Three Mile Island Unit 2
WOG	Westinghouse Owners Group

1 BACKGROUND

Why do we need post accident sampling? During and shortly following the accident at Three Mile Island Unit 2 (TMI or TMI-2) in 1979, the emergency response team had problems obtaining timely information for assessing the state of the plant and the core in relation to both onsite and offsite emergency response activities. There were two major factors that contributed to the situation at TMI: a) the reliance on sampling for information related to the condition of the plant, and b) the high levels of radioactivity in the samples. The design basis for post accident recovery from an accident included the analysis of samples of plant fluids using the same sample points and sample analysis techniques that were used during normal operation. Since this accident was well beyond the design basis event, the ability to draw and analyze samples of plant fluids was compromised by the high levels of radioactive materials in the samples. A number of workers received significant radiation doses while attempting to draw samples of plant fluids. Many of the samples had to be sent offsite for analysis due to the high radiation levels. Issues also arose concerning the plant condition following the accident that had not previously been addressed, such as the potential hydrogen bubble in the upper reactor vessel head.

One of the "lessons learned" from the TMI accident was the need to have the capability to obtain and analyze samples of plant fluids containing high levels of radioactivity. The 1980 review of the accident progression and consequences at TMI also revealed the possible need for information that was not previously addressed in dealing with the recovery from an accident (references 1 and 2).

1.1 REGULATORY BASIS

The need for post accident sampling capabilities was part of the U.S. Nuclear Regulatory Commission's (NRC) changes in licensing and operating requirements to address the TMI "lessons learned". These requirements were published as part of the NUREG 0600 (reference 3) and NUREG-0737 (references 4 and 5) changes to plant design and operation after the TMI accident. These requirements were directly tied to the understanding and knowledge of the progression and consequences of core damage accidents immediately after the TMI accident.

Specifically, NUREG-0600 contains the following requirements:

- Obtain Reactor Coolant System (RCS) and containment atmosphere sample without exposure in excess of 3 rem whole body dose,
- Radioactivity analysis in less than 2 hours to include noble gases, iodines and cesiums, and non-volatile isotopes,
- Boron sample and analysis in less than 1 hour, and
- Chloride sample and analysis within a shift

As further assessments of the need for post accident sampling were completed, the requirements specified in NUREG-0737 were expanded to include, among others, the following key points:

- Obtain RCS and containment atmosphere sample in less than 1 hour without exposure in excess of 3 rem whole body dose.
- Radioactivity analysis in less than 2 hours to include noble gases, iodines and cesiums, and non-volatile isotopes. *[The basis for this requirement, as noted in NUREG-0737 is that noble gas indicates cladding failure, iodines and cesiums indicate high fuel temperatures, and non-volatiles indicate fuel melting.]*
- Radionuclide samples must be analyzed to the levels given in Regulatory Guides 1.3, 1.4 and 1.7. Further, the capability should be in the range of 1 microcurie per gram to 10 curies per gram.
- Containment hydrogen levels in less than 3 hours combined sample and analysis.
- Dissolved RCS gases (e.g., hydrogen) .
- Boron sample analysis capability in less than 1 hour.
- Chloride sample analysis capability within a shift.
- If on-line sampling is used to meet the time limits, a grab sample capability must be maintained as a backup for both containment atmosphere and RCS samples.
- The combined time for sampling and analysis shall be equal to or less than 3 hours.
- Sampling shall not require change in isolation status of auxiliary systems.
- Measuring dissolved oxygen in the RCS sample is recommended, but not mandatory.
- Accuracy of the post accident sampling analysis shall be within a factor of 2 compared to more sophisticated sampling and measurement techniques.
- The time for chloride is dependent on whether the plant's coolant water is brackish/ seawater and whether there is only a single barrier between the containment and the cooling water. For brackish/seawater plants with only a single barrier between the containment and the cooling water, a chloride analysis within 24 hours of the sample was required; for all others, a chloride analysis within 4 days was required.

In addition to the post accident sampling requirements, a requirement for continuously measuring the containment hydrogen concentration was also included in NUREG-0737 in Section II.F.1. The requirement specifies the capability to continuously record and measure

containment hydrogen concentrations within 30 minutes of the initiation of a safety injection signal.

These NRC requirements were closely tied to other post TMI-requirements. Specifically, the NRC post-TMI action plan called for an emergency classification scheme which required the diagnosis of failure of the fission product boundaries and in particular, the fuel rod cladding.

The requirement for an emergency classification scheme is contained in 10 CFR 50.47 (b)(4):

> *"A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee ..."*

and 50.47(b)(9):

> *"Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use."*

The emergency classification scheme that was used by most plants in the 1980's and early 1990's is based on information in Appendix 1 of NUREG/CR-0654 (reference 6). The emergency classification scheme detailed in NUREG/CR-0654 depends, in part, on the diagnosis of failures of the fission product boundaries, including the fuel rod cladding. In the early 1990's the Nuclear Energy Institute (NEI) sponsored the development of an updated emergency classification scheme (reference 7). This updated emergency classification scheme also relies on the identification of a loss of the fuel rod cladding integrity as part of the classification criteria.

The NRC requirements for post accident sampling system capabilities are contained in 10 CFR Part 50.34(f)(2)(viii) for plants that did not have a construction permit at the time of the TMI-2 accident:

> *"Provide capability to promptly obtain and analyze samples from the reactor coolant system and containment ... Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines, cesiums, and nonvolatile isotopes), ..."* **[Note: although 50.34 (f)(2)(viii) was not required for any of the currently operating Westinghouse Pressurized Water Reactors (PWRs), it is included here to show NRC intent.]**

For those plants that had a construction permit or operating license at the time of the TMI-2 accident (all of the existing WOG operating plants), the applicable post accident sampling criteria are contained in NUREG-0737, Criteria II.B.3:

> *"A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (in less than 2 hours) certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicates cladding failure), iodines and cesiums (which indicate high fuel temperatures), and nonvolatile isotopes (which indicate fuel melting)".*

With the plethora of new regulatory requirements following the TMI-2 accident, many utilities took advantage of a situation in which a plant upgrade could cover more than one new requirement. Such is the case with post accident sampling. The post accident sampling of radionuclides could help the utility meet the new requirements for emergency classification, as well as the core damage assessment.

In other related regulatory documents, Regulatory Guide 1.97, Revision 3 (reference 8) contains a list of variables to be monitored during and following an accident that is based on NUREG-0737. Regulatory Guide 1.97 represents the latest NRC positions on post accident sampling. Table 1 provides a list of the variables that involve sampling and related information from Regulatory Guide 1.97. The regulatory guide also specifies that within the first 30 days of the accident, RCS oxygen analysis need not be performed until chloride analysis indicates an RCS chloride concentration greater than 0.15 ppm. Once the chloride concentration exceeds this value, oxygen should be determined within 3 hours. For this 30-day period, it is acceptable to verify that dissolved RCS oxygen is less than 0.1 ppm, if the measured dissolved RCS hydrogen residual is 10 cc (STP)/kg of coolant or less. However, consistent with minimizing personnel radiation exposures "as low as reasonably achievable", (ALARA) direct monitoring for dissolved RCS oxygen is recommended.

The regulatory guide also refers to General Design Criterion (GDC) 64, "Monitoring Radioactivity Releases" (reference 9) that includes a requirement that means be provided to monitor the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluid, effluent discharge paths, and the plant environs for radioactivity that may be released from postulated accidents. While GDC 64 refers to monitoring, sampling has always been an acceptable method to monitor the chemical or radioactive content of plant fluids. Thus, the GDC 64 criteria have applicability to post accident sampling.

1.2 EXISTING POST ACCIDENT SAMPLING SYSTEMS

In March of 1995, Westinghouse initiated a survey of the Post Accident Sampling System (PASS) capabilities at the nuclear plants employing a Westinghouse Nuclear Steam Supply System (NSSS). The purpose of the survey was to gather information concerning the potentially diverse PASS capabilities and actual problems with PASS operation, testing and maintenance. This information was subsequently used to define a generic Westinghouse Owners Group program for modifying the PASS requirements at Westinghouse NSSS plants, based on the current knowledge of the diagnosis, progression and consequences of core damage accidents. Those modifications are the subject of this report.

Table 1
Regulatory Guide 1.97 Requirements

Parameter	Range	Category[1]	Purpose
RCS Soluble Boron (Grab Sample)	0 to 1000 ppm	3	Verification of reactivity control
Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	½ Tech Spec Limit to 100 times Tech Spec Limit	1	Detection of fuel cladding breach
Analysis of Primary Coolant (Gamma Spectrum)	10 μCi/ml to 10 Ci/ml or TID-14844 source term in coolant volume	3	Detailed analysis of fuel cladding breach; accomplishment of mitigation; verification; long term surveillance
Containment Hydrogen Concentration	0 to 30 vol-% with capability of operating from -5 psig to design pressure	1	Detection of potential for breach; accomplishment of mitigation
Primary Coolant & Sump Grab Sample[2]		3	Release assessment; verification; analysis
Gross Activity	1 μCi/ml to 1 Ci/ml		
Gamma Spectrum	Isotopic Analysis		
Boron Content	0 to 6000 ppm		
Chloride Content	0 to 20 ppm		
Dissolved Hydrogen or Total Gas	0 to 2000 cc (STP)/kg		
Dissolved Oxygen	0 to 20 ppm		
pH	1 to 13		
Containment Air Grab Sample		3	Release assessment; verification; analysis
Hydrogen Content	0 to 10 vol-%		
Oxygen Content	0 to 30 vol-%		
Gamma Spectrum	Isotopic Analysis		

(1) Refers to Table 1 of Reg. Guide 1.97 related to design and qualification requirements.

(2) An installed capability should be provided for obtaining containment sump, ECCS pump room sumps, and other similar auxiliary building sump liquid samples.

1.2.1 Post Accident Sampling System Capabilities

The survey results show a wide diversity in capabilities and equipment used to meet the post accident sampling requirements in NUREG-0737. Table 2 gives a perspective on those capabilities regarding the samples that can be taken and the sophistication of the sampling/analysis methodology. The manual sampling/analysis capability refers to the need to manually draw the sample and then analyze the sample in the radio-chem laboratory. Automatic sampling/analysis capability refers to on-line sampling capability wherein the sample is analyzed using computerized equipment. Manual vs. automatic is not a measure of the accuracy of the PASS capability, only of the need for manual activities which infers both potential dose rates and time delays in obtaining sample results.

The survey also indicated that there is not a uniform standard for post accident sampling. A number of utilities indicated post accident sampling capabilities that were not included in the definition of post accident sampling for other plants. This can be easily seen from Table 2 where about half of the plants include the capability for sampling/analyzing reactor coolant oxygen, while the other half does not treat this capability as a post accident sampling function. For reactor coolant conductivity and containment oxygen, only a small number of plants include these in the post accident sampling capabilities.

1.2.1.1 Containment Hydrogen Monitor

Each plant has the capability to measure containment hydrogen concentrations in the range of (at a minimum) 0 to 10 volume percent with a reported accuracy of 1 volume percent. A typical containment hydrogen monitor continuously draws a containment air sample from the containment, processes the sampled stream though the analyzer section of the equipment and then returns the sample stream to the containment. The containment hydrogen monitor is normally maintained in a standby condition during plant operation and can be put in-service by opening the sample line isolation valves.

1.2.2 Post Accident Sampling System Experience

In addition to the diversity in equipment and methods used to satisfy the PASS requirements, the survey provided an opportunity for the respondents to describe problems associated with the present PASS capabilities and/or requirements, as well as to provide information related to the costs of maintaining PASS. This information is present in Table 3.

Table 2
PASS Survey Results

Plant	Liquid Samples							Gas Samples		Radio-nuclides	
	Dissolved Gas										
	H2	O2	Gas	pH	Cl^-	B	Cond	H_2	O_2	Ctmt	RCS
A	A	M	A	A	A	A	M	A	A	M	M
B	M		M	A	M	A	A	M		M	M
C	M		M	A	M	M		M		M	M
D	A	A	A	A	A	A		A		A	A
E	A		A	A	A	A	A	A		A	A
F	A	A	A	A	A	A		A		A	A
G	M		M	A	M	M		M		M	M
H	A		A	A	M	M		M		M	M
I	A	A	A	A	A	A		M		M	M
J	M	M	A	A	M	M		M		M	M
K	M	M	A	A	M			M		M	M
L	A	A	A	A	A	A			M	A	A
M	M		M	M	M	M		M		M	M
N	A		A	A	A	A		M		A	A
O	A	A	A	A	M	M		M		M	M
P	M		M	A	M	M		M		M	M
Q	A	A		A	A	M	A	A	A	M	M
R	A	A	A	A	A	A		M		M	M

Legend: M = Manual sampling and analysis required

A = Automatic, on-line sampling and analysis capabilities (may require manual action to un-isolate system following an accident)

Table 3
Post Accident Sampling System Issues and Costs

Plant	Major Issues	Planned PASS Modifications
A	Gas and Ion Chromatograph maintenance	None
B	Routine surveillance testing and calibration (low levels and sample dilution)	Upgrade equipment
C	Calibration and maintenance, chloride sample contamination	None
D	Instrument calibration and operability tests; maintenance in hi rad areas	None
E	Upkeep of chemical analyzers and containment isolation valves	None
F	Maintaining required flow rates under transient conditions	None
G	pH collection and analysis	None
H	Training, Calibration and Maintenance	Upgrade equipment
I	Maintenance for leakage on valves and sample accuracy at low concentrations	None
J	Maintenance on liquid sample valves and flow transmitters	None
K	Maintenance and calibration of analyzers	Upgrade equipment
M	Maintenance of valves	None
M	Surveillance testing of diluted samples	Recent upgrades
N	Valve repair and gas analysis	None
O	Maintenance on leaking containment ISVs, unreliable on-line chemistry	Abandon on-line analysis; use grab sampling
P	Valve and regulator maintenance; chromatograph availability	Upgrade equipment

Also several utilities reported issues related to maintaining and calibrating the on-line containment hydrogen monitor. In particular, during calibration checking, the hydrogen analyzer requires more than 30 minutes for the reading to stabilize after the gas sample used for calibration checking is introduced into the analyzer. Also, the hydrogen analyzer requires continual re-calibration to meet the advertised ± 1 volume percent accuracy.

Thus, the technical basis for on-line continuous measurement of containment hydrogen was also included as part of the development of a current technical basis for the post accident sampling system.

1.2.3 Post Accident Sampling System Costs

Other information provided in the survey responses indicates that some utilities are planning to spend as much as one million dollars over the next few years to replace aging PASS system components and/or to upgrade capabilities where problems presently exist. In addition, utilities are spending as much as 2000 man-hours per year for the PASS system, which includes training, maintenance, surveillance and testing. While the $1 M capital expenditure and the 2000 man-hours are the extreme cases, they point out the potential savings that could accrue with relaxed PASS requirements.

1.2.4 Post Accident Sampling System Limitations

Drawing samples from the reactor coolant system, containment sump or containment atmosphere cannot provide instantaneous information regarding the radionuclide content of the plant fluids. There is a wide variation in the PASS methodology among the WOG member plants. A few plants have "on-line" capability whereby once started, a sample can be drawn and automatically processed by the installed equipment with no further significant interaction by plant personnel. At the other end of the spectrum, a sample is drawn into a container when requested and subsequently transported to the plant radiochemistry laboratory for analysis.

The typical procedures for drawing a sample with the post accident sampling system require that the sample lines be purged for 10 to 15 minutes before drawing a sample for analysis. The sample is then drawn over a period of 5 to 10 minutes to ensure that a representative sample is obtained. With either the automatic or the manual system, the initiation of the sampling process requires a considerable time period to assure a representative sample (e.g., circulating fluid through the sample lines prior to sampling/analysis). The specification for the sampling for radioactivity content of plant fluids generally (based on a survey of a limited number of WOG member utility plants) requires that the first sample results be available within two to twenty-four hours of the accident initiation, depending on actual plant requirements. The requirement for subsequent samples is contained in NUREG-0737 II.B.3, Clarification #8: "Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of an accident and at least one sample per week until the accident condition no longer exists. Thus, the process of drawing a representative sample and then analyzing the sample means that the resultant measurement is tens of minutes to hours in arrears with respect to current conditions. Analyses with the Modular Accident Analysis Program (MAAP) code indicate that an uncovered core can proceed from a point just prior to

the onset of significant hydrogen generation (e.g., all fuel rod cladding temperatures less than 1800°F) to core melting in a matter of 30 minutes. Thus, the measurement of radioactivity in plant fluids via sampling is not useful in estimating core damage in a transient situation.

There may also be some limitations presented by core damage accident conditions on the ability to use the PASS and the hydrogen analyzer, such as pressure and temperature of the plant stream from which the sample is being drawn. For example, if containment pressure during a core damage accident exceeds the containment design pressure, samples of containment atmosphere may not be able to be drawn due to the pressure capability of the sampling lines and associated equipment. The same containment pressure limitation also applies to the on-line containment hydrogen monitor. Also, the reactor coolant samples generally require an RCS pressure of several hundred psi in order to obtain adequate flow rates in the sample lines. Thus, for some Loss of Coolant Accidents (LOCA) and other events where the reactor coolant system is depressurized, it would not be possible to obtain a reactor coolant sample. A sample from the containment sump may be representative of the reactor coolant system for LOCA events where the reactor coolant is spilling to the containment sump. However, in many plants, the containment sump sample is drawn from the low pressure Emergency Core Cooling System (ECCS) recirculation line. A review (reference 10) across all of the Individual Plant Examination Studies (IPE) indicates that the failure to establish ECCS recirculation is, generically, a dominant contributor to core damage. Thus ECCS recirculation may not be in-service if core damage is occurring. After recovery from a core damage accident wherein ECCS recirculation is in-service, containment sump samples could be obtained; however, recovery from a core damage accident does not necessarily require the use of ECCS recirculation. Therefore, these samples would only be useful for assessing the plant condition following recovery from some core damage accidents, many hours after the usefulness of the information for prevention and /or mitigation of core damage has passed.

Recent analytical studies have also indicated that the dense aerosols concentrations in the reactor coolant system and the containment during a core damage accident could plug sample lines and render the post accident sampling system permanently inoperable for the remainder of the accident. Thus, there is some intuitive logic for not using the sampling system during the early phases (e.g., the first day) of a core damage accident when dense aerosols may result in sample line plugging.

1.3 RELATIONSHIP OF POST ACCIDENT SAMPLING TO EMERGENCY PLANNING

In response to the NRC's post-TMI requirements outlined in NUREG-0737, post accident sampling capabilities were implemented at all plants, albeit as described in the previous section, there was no standard by which the required post accident sampling capabilities were achieved. As implemented, the post accident sampling system was directly tied to two other elements of emergency planning: core damage assessment and emergency action levels. In addition, the post accident sampling capability was indirectly linked to the "new" symptom based emergency operating procedures (EOPs) that were also required by NUREG-0737. For WOG member utilities, the symptom based EOPs, which were based on the WOG Emergency Response Guidelines (references 11 and 39), only covered plant conditions up to the onset of

core uncovery and overheating. However, it was believed that the post accident sampling capability would also provide useful information for accidents that progressed beyond that point. This was based on the WASH-1400 report (reference 12), which was the first attempt to quantitatively describe the progression and consequences of accidents that progressed to core overheating and melting. Coincidentally, WASH-1400 was the only comprehensive treatment of core damage accidents at the time of the TMI-2 accident and the subsequent development of the new regulatory requirements.

In the area of core damage assessment, in 1984 the Westinghouse Owners Group issued a report (reference 13) that described a generic methodology for estimating the amount of core damage. This methodology, which was used to develop a plant specific methodology at most WOG member plants, relied heavily on the results of radioactivity analysis of samples taken from the plant after core damage occurred. In 1999, a new WOG program was completed to update the core damage assessment methodology to reflect the current understanding of severe accidents (reference 14). This new investigation concluded that the results of radioactivity analyses of samples of plant fluids was too unreliable to make predictions regarding the amount of core damage that has occurred. The unreliability of samples was two-fold: a) the transport and deposition of radionuclides in the plant is very dependent on many details of the accident that cannot be diagnosed nor measured during the accident, and b) the time delay in obtaining and analyzing a sample for radionuclide content did not provide reliable information for decision making, especially during the transient parts of the accident when the information would be most valuable. Therefore, the new WOG Core Damage Assessment Guideline relies only on the indications that are provided by fixed in-plant instrumentation. The report cautions against making core damage assessments based on the results of radiological analysis of samples of plant fluids. The report concludes that the only radiological analysis of a sample of plant fluids that may be of any use is the noble gas content of the containment atmosphere, and only after recovery from the accident has been completed.

The usefulness of information obtained from radiological analysis of plant fluids for emergency action level classification was also investigated in the development of the 1999 WOG Core Damage Assessment Guidance. The investigation concluded that the radiological criteria in the classification scheme reported in NUREG-0654 was unnecessary for two reasons: a) the information would not be timely for decision making, and b) in the event of a core damage accident, other criteria would trigger the appropriate emergency action level in a more timely manner. A similar conclusion was drawn with respect to the new NEI emergency action level classification scheme.

1.4 RECENT REGULATORY ASSESSMENTS

In 1986, the NRC published an assessment of the regulatory requirements associated with 10 CFR Part 50 that, if deleted or appropriately modified, would improve the efficiency and effectiveness of the NRC regulatory program without adversely affecting safety. The report, NUREG/CR-4330, Volume 1 (reference 15), identified 45 regulatory requirements that could be eliminated or relaxed without adversely impacting safety. The post accident sampling system regulatory requirements (which were one of the 45 requirements mentioned directly above)

were identified as an area with a relatively high level of concern over the regulatory burdens associated with the system.

In 1987, the NRC published a more detailed assessment of the regulatory requirements related to the post accident sampling system in NUREG/CR-4330, Volume 3 (reference 16). The assessment examined a relaxation of the time requirements to obtain samples and the elimination of some samples. The effect on public health and safety was estimated to be marginal. The estimated benefits of this new regulatory position were judged by the NRC and its contractors to be "...rather insignificant because installation of the system is complete at all plants and the potential operational cost savings are small". The following recommendations were made in NUREG/CR-4330, Volume 3:

- Relaxation of time requirements for sample and analysis - boron is the only area where prompt analysis could aid in mitigation or arrest of an accident. The timing of other PASS sample results leads them to become marginal or even negligible in affecting public risk. The time allotted for collecting and analyzing samples during this time could be increased (e.g., to six hours).

- Elimination of the requirement to conduct radiological analysis - because of timing, this information is not available for accident management or emergency response decision making. Other indicators are more readily available. The information is useful to the plant only in accident recovery (after the plant has been stabilized).

- Elimination of the requirement to conduct hydrogen analysis of the containment atmosphere, since redundant, safety grade hydrogen monitoring equipment is now "standard".

- Elimination of the requirement to conduct dissolved gas analysis of coolant, because of the incorporation of reactor head vents and reactor vessel level instrumentation.

- Elimination of the requirement for heat tracing in sample lines if radioiodine is not being used for core damage estimation.

In light of the results of the Westinghouse survey discussed previously in Section 1.2.2 of this report, the NRC and its contractors have underestimated the costs of the post accident sampling system. Thus, the basis for not considering regulatory relief for post accident sampling in NUREG/CR-4330, Volume 3 is not supported by current experience. However, the technical basis established for the regulatory relief in that document is still valid and should be pursued.

In April of 1993, the NRC issued their policy position related to the design and licensing of the evolutionary and advanced light water reactor designs in SECY-93-087 (reference 17). While this position was explicitly developed to address design and licensing issues for the evolutionary and advanced reactor designs, the technical basis provides a foundation for further assessment of PASS requirements for the current generation designs, provided that the evolutionary and advanced designs do not contain some feature that is central to the NRC position. In their 1993 policy on post accident sampling requirements for the evolutionary and

advanced designs, the NRC staff recommends that the Commission approve deviations from the requirements outlined in Item II.B.3 of NUREG-0737:

- the post accident sampling system should have the capability to take boron samples at 8 hours after an accident,

- the post accident sampling system should have capability to take dissolved gas and chloride measurements at 24 hours after an accident, and

- the post accident sampling system should have the capability to take radioactivity measurements at 24 hours after an accident.

On the first issue (boron sampling from the RCS), the NRC staff agreed with the Electric Power Research Institute (EPRI) position (reference 18) that sampling for boron would not be required in the first 8 hours of an accident. This is based on the requirement for the evolutionary and advanced designs to include neutron flux monitoring capability that meets the Category I criteria of Regulatory Guide 1.97. Thus, a design feature of the evolutionary and advanced plants plays a role in the recommended relaxation of this requirement.

On the second issue (dissolved gas and chloride measurements), the NRC staff has taken the position that the need for dissolved gas measurements stems from consideration of partially mitigated accident sequences that do not involve early RCS depressurization. For these accidents, the dissolved gas samples are needed to assure that establishing natural circulation for decay heat removal can proceed without interference from noncondensible gases in the reactor coolant system. The availability of reactor coolant vessel level instrumentation and high point vents (also required by NUREG-0737) did not alter this position. However, the NRC did recommend that the need for post accident sampling of dissolved gases would not be necessary for evolutionary and passive ALWRs of the boiling water type.

On the third issue (radioactivity measurements), the NRC staff recognizes that core damage assessment will not be based on sampling of radionuclides from plant fluids, but rather in fixed instrumentation such as the containment hydrogen monitor, the containment high range area radiation monitor and the core exit thermocouples. The need for PASS activity measurements will arise during plant recovery when the degree of core damage and general plant contamination will have to be evaluated.

It is interesting to note that the NRC staff positions in SECY-93-087 (1993 vintage) are less restrictive than several positions espoused in NUREG/CR-4330, Volumes 1 and 3.

1.5 APPLICATIONS FOR RELAXATION OF POST ACCIDENT SAMPLING REQUIREMENTS

There have been several submittals to the NRC requesting relaxation of some of the post accident sampling system requirements.

In 1994, the Combustion Engineering Owners Group (CEOG) submitted a proprietary report, CEN-415, that contained the technical justification for generic relaxation of a number of post accident sampling system requirements for CEOG member plants. Following review of the technical justification contained in the report, the NRC issued a Safety Evaluation Report (SER) (reference 19) approving generic deletion or relaxation of a number of the requested PASS requirements for CEOG member plants. Those approved include:

- Deletion of the requirement for measurement of pH of reactor coolant,
- Deletion of the requirement for measurement of containment hydrogen concentration via sampling,
- Deletion of the requirement for heat tracing of sample lines,
- Deletion of the requirement for oxygen analysis of reactor coolant, and
- Relaxation of the sample points requirements to include only containment atmosphere and reactor coolant samples.

However, the NRC did not approve deletion or relaxation of several PASS requirements that were requested in the CEOG submittal. Those NUREG-0737 requirements that were not approved include:

- Relaxation of the 3 hour PASS sampling time, and
- Elimination of the measurement of hydrogen and total gas in the primary coolant system as a means of determining the conditions inside the reactor vessel.

In not approving the relaxation of the 3 hour PASS sampling time, the NRC took the position that NUREG-0737 clearly states the required capability to draw samples within three hours of the beginning of an accident, and that further revision or relaxation of those requirements would require approval by the Commission. However, the NRC SER on the CEOG request (reference 19) indicated that the NRC was in the process of revising some of the time limits.

For PWRs, the new time limits for taking the samples and analyzing them for dissolved gas and activity could be extended to 24 hours, and for boron to 8 hours after the end of power operation.

In not approving the deletion of measurement of hydrogen and total gas in the reactor coolant system, the NRC took the position that the concentration of gases in the reactor coolant system is one of the most direct parameters in diagnosing problems related to interruption of natural circulation caused by release of noncondensible gases in the reactor vessel. The NRC claims that the reactor coolant dissolved gas concentrations also constitute an important parameter in determining the degree of core damage from the estimate of the amount of fuel rod cladding oxidation. However, the NRC SER on the CEOG request concedes that an option for using either the PASS or the normal sampling system for taking coolant samples is acceptable,

provided that the radiation exposure limits of General Design Criterion 19 (reference 20) are not exceeded.

In 1995, TVA submitted a report to the NRC requesting relaxation or deletion of a number of PASS requirements for the Sequoyah nuclear power station (references 21 and 22). The TVA effort covers most of the CEOG areas and extends well beyond the relaxation requested by the CEOG. The scope of the TVA request includes the following PASS requirements:

- Relaxation of Sampling/Analysis Times
  - Change RCS and/or containment sump boron to 8 hours
  - Change RCS and/or containment sump gamma spectrum to 24 hours
  - Change RCS and/or containment sump gross activity to 24 hours
  - Change RCS and/or containment sump chloride to 24 hours
  - Change RCS dissolved hydrogen or total gas to 24 hours
  - Change Containment atmosphere gamma isotopic to 24 hours
  - Change RCS and/or containment sump chloride to 24 hours
- Elimination of required samples
  - Elimination of hydrogen and oxygen analysis on containment atmosphere samples
  - Elimination of potential hydrogen (pH) analyses on RCS or containment sump samples
  - Elimination of dissolved oxygen analysis on RCS samples
- Relaxation of sample analysis accuracies
  - Decrease required accuracy for RCS and containment sump boron

In their PASS relaxation submittal for the Sequoyah plant, TVA also referenced the definitions of "Potential Core Damage" and "Stable Core" conditions, based on the NUMARC EALs (reference 7) for potential loss of the fuel clad barrier. TVA maintained that following a potential core damage accident, sampling/analysis will be completed for parameters that support accident recovery within the stated response times AFTER stable core conditions are regained.

The TVA submittal defined a potential core damage accident and used the following criteria for potential loss of fuel clad barrier:

- Core exit thermocouples indicate greater than 700 degrees F, or
- Inadequate core cooling (orange on status tree) or heat sink (red on status tree), i.e., RHR shutdown cooling not in service, or
- Reactor vessel level indicates less than 40 percent with no reactor coolant pump running.

Using this methodology, stable core conditions are then determined to be regained by the following criteria:

- Core exit thermocouples indicate less than 700 degrees F,
- Adequate core cooling and heat sink exists, and
- No immediate threats to safety systems required to maintain core cooling.

These definitions are consistent with the EAL classifications in that stable core conditions are defined such that the potential will not continue to exist for loss of fuel clad barrier and propagation of core damage.

In addition, TVA clarified/defined total sample and analysis accuracies; the NRC only required analysis accuracies and omitted sampling accuracy in the NUREG-0737 requirements.

The initial TVA submittal (reference 21) argued that the "clock" for taking and analyzing samples should start at the time when a stable core condition is regained following a potential core damage accident. In response to NRC questions, TVA amended the submittal (reference 22) to include the statement: *"Following a potential core damage accident, sampling and analysis is capable of being performed within the stated response times after the accident"*. This was provided to meet the intent of 10 CFR 50.34 (f)(2)(viii).

The NRC approved (reference 38) the TVA request, except for the relaxation of the post accident radionuclide sampling requirements. The NRC stated that post accident radionuclide sampling requirements were under generic consideration by the NRC staff and could not be approved at this time. However, TVA could submit a request for relaxation of the post accident radionuclide sampling requirements following generic resolution of this issue. A matrix of the PASS requirements and the areas of relaxation approved by the NRC for CEOG and TVA is shown in Table 4.

Table 4 Summary of Relaxations For Post Accident Sampling Requirements				
Sample Point	**Parameter**	**Delete Analysis**	**Relax Analysis Range/Accuracy**	**Relax Response Time**
RCS	Boron	-	Sequoyah	Sequoyah
	PH	Sequoyah/CEOG	-	-
	Chlorides	-	Sequoyah	Sequoyah
	Conductivity	Sequoyah	-	-
	Dissolved Gas	-	-	Sequoyah
	Dissolved Hydrogen	-	Sequoyah	Sequoyah
	Dissolved Oxygen	Sequoyah/CEOG	-	-
	Radionuclides	-	-	-
	Gross Activity	-	Sequoyah	Sequoyah
Containment Sump	Boron	CEOG	Sequoyah	Sequoyah
	PH	Sequoyah/CEOG	-	-
	Chlorides	CEOG	Sequoyah	Sequoyah
	Radionuclides	CEOG	-	-
	Gross Activity	CEOG	Sequoyah	Sequoyah
Containment Atmosphere	Radionuclides	-	-	-
	Hydrogen	Sequoyah/CEOG	-	-
	Oxygen	Sequoyah	-	-
All	Heat Tracing	CEOG	-	-

Note 1: Sequoyah also defined "Potential Core Damage" and "Stable Core Conditions" based on the NUMARC Emergency Action Levels (reference 7) for potential loss of fuel cad barrier.

Note 2: Other utilities have requested and receive approval for relaxation of specific requirements based on plant specific concerns. This table only includes the more global efforts of TVA and the CEOG.

2 CHARACTERIZATION OF CORE DAMAGE ACCIDENTS

This section, and the remainder of the report, deals with accidents in which some damage has occurred to the reactor core. If there is no core damage, the radiation levels in the reactor coolant system and the containment would not prohibit the use of the normal sampling system. The normal sampling system is also available for conditions in which an iodine spike occurs as a result of the reactor trip or RCS depressurization. Only in the case of a loss of fuel rod integrity do the reactor coolant and containment radiation levels increase to a level where the post accident sampling system is required. Thus, the post accident sampling system is directly tied to accidents which result in core damage. In some cases, this impacts the procedures and guidance in use in the control room and the other emergency response facilities. The discussions in the remainder of this report assume that some level of core damage has occurred, and any guidance or procedure that is not used after core damage is not relevant to the post accident sampling system requirements.

This section of the report presents a summary of the current understanding of core damage accidents as it relates to the need for post accident sampling. The overall core behavior during a core damage accident is briefly discussed, followed by a summary of the fission product behavior after release from the fuel matrix. It is very important to establish the basis for the progression of a core damage accident and the behavior of the fission products and other important chemical species before attempting to formulate the required capabilities of the post accident sampling system that would provide useful information for accident management. 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.

2.1 CORE BEHAVIOR DURING AN ACCIDENT

The fuel rods in a Westinghouse PWR are constructed of a zirconium 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 with helium and during normal operation, the internal pressure of the fuel rod increases as a result of the heat generated by the fissioning of the uranium and the migration of some of the fission products, primarily xenon and krypton to the gap space. The fuel rod cladding serves two primary purposes: a) to keep the fuel pellets in a geometry that assures a controlled and efficient fission process, and b) to contain radioactive fission products, thereby preventing their release into plant systems or the atmosphere.

From the perspective of the post accident sampling system, there are three broad accident classes that can occur in a Westinghouse PWR:

1. Accidents with elevated coolant activity levels (e.g., iodine spiking), but with no fuel rod overheating and no gross loss of fuel rod integrity,

2. Accidents with a gross loss of fuel rod integrity but without high indicated fuel temperatures, and

3. Accidents with the loss of fuel rod integrity accompanied by high indicated fuel temperatures.

The three classes of accidents are similar to the composite core damage classifications described in NUREG-0737: a) fuel rod cladding failure, b) fuel pellet overheating, and c) core melting. However, for the purposes of the following discussions, the three broad classes of core damage listed above will be used since they better describe the need for information from post accident sampling.

The first class of accident 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 23). In this case, the in-plant response to the accident would be according to the plant Emergency Operating Procedures (EOPs). Additionally, the Emergency Action Level (EAL) classification scheme would indicate no potential loss of the fuel clad barrier and therefore there is no reason to escalate the global response level based on the core state. Except in the case of a large LOCA, the reactor coolant system remains filled with water during the time that samples may be requested. Depending on the accident, reactor coolant may be released to the containment and the containment sump may contain water. The core may be cooled by any number of means, including: the ECCS recirculation from the containment sump, natural circulation using the steam generators, forced circulation using the reactor coolant pumps and steam generators, or the normal Residual Heat Removal (RHR) system. Also, 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.

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. In this case, there would be a release of fission products from the fuel rods to the reactor coolant system. The in-plant response to the accident would be according to the plant EOPs. The Emergency Action Level (EAL) classification scheme would indicate a potential loss of the fuel clad barrier and therefore may be reason to escalate the offsite response level based on the core state. Except in the case of a large LOCA, the reactor coolant system remains filled with water during the time that samples may be requested. Depending on the accident, reactor coolant may be released to the containment and the containment sump may contain water. The core may be cooled by any number of means, including: ECCS recirculation from the containment sump, natural circulation using the steam generators, forced circulation using the reactor coolant pumps and the steam generators, or the normal RHR System. Also, 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. Compared to the first case, there is a more urgent need for information concerning the plant status when the gross fission product levels in plant systems and/or containment are higher than anticipated, as indicated by the plant area radiation monitors.

The third, 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 and the exact status of the core is more uncertain than in the previous cases.

In this case, the in-plant response to the accident would initially be from the plant EOPs. However, if prolonged high core temperatures are indicated, the plant operating staff would be directed to transfer to the plant Severe Accident Management Guidance (SAMG) (reference 24). The EAL classification scheme would indicate a potential loss of the fuel clad barrier and therefore there is reason to escalate the offsite response level based on the core state. 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. In the time just preceding high core temperature indications, the water level in the reactor vessel would be decreasing and the fuel rods would become uncovered. Depending on the accident, reactor coolant may be released to the containment and the containment sump may contain water. If the accident is successfully recovered, the core may be cooled by any number of means, including: ECCS recirculation from the containment sump, natural circulation using the steam generators, forced circulation using the reactor coolant pumps and steam generators, or the normal RHR System. Also, if reactor coolant has been released to the containment, the containment atmosphere would contain high levels of airborne fission products. 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.

This third accident class, one in which there has been inadequate core cooling, can result in three distinctly different states that can possibly influence the post accident sampling capabilities:

- Core geometry intact - in this condition, the fuel rods have experienced overheating and possibly some melting but the pre-accident fuel rod geometry remains. In this case, high core temperatures normally associated with core damage (e.g., greater than 1800°F) are not required to cause failure of the fuel rod cladding in some of the fuel rods in the core. A small LOCA or a large LOCA, where the emergency core cooling system operates as designed, can result in temperatures in the core which are only above the normal operating temperatures for a short period of time. This results 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, due for example to cold water injection, can result in localized failures of the fuel rod cladding. This type of fuel rod failure can be widespread throughout the core, depending on actual conditions. In this type of accident, some of the control rods may be damaged due to mechanical stresses in the core region (e.g., the LOCA blowdown forces) and may not be able to be fully inserted after reactor trip.

- Core 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). In this type of event, 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.

If cooling is not restored to the overheating core, the loss of fuel rod integrity will progress to a state in which the original core geometry is drastically changed as a result of melting and downward relocation of the core and cladding materials. Once the local fuel rod temperatures exceed about 2600°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 in the core. The fuel rod cladding and the fuel does not begin to relocate downward until the fuel rod temperatures approach 4000°F. In a typical PWR, where the core heatup rate is on the order of 5°F per second or greater, the period of time between the beginning of the control rod downward relocation and the beginning of the core downward relocation is therefore on the order of about 5 minutes. In a core damage accident, the core region is filled with steam and hydrogen, which assures that the core is in a subcritical condition. When the control rod material begins to relocate downward, subcriticality is still maintained due to the absence of moderator in the core region. Once the core begins its downward relocation, subcriticality is further assured as a result of the compaction of the fissile material and the reduction in moderator volume. However, in the brief interval between the time when the control rods begin downward relocation and when the core begins its downward relocation, if the reactor vessel were refilled with water, the normal shutdown margins may not be available due to the loss of the control rod material; only the borated water is available for maintaining subcriticality.

- 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. Due to the changes in the fuel geometry, recriticality of core debris outside of the reactor vessel is generally considered to be impossible.

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 in individual fuel rods in the core. A summary of the behavior, taken from NUREG-1228 (reference 25), is provided in Table 5. This information is generally consistent with the core behavior modeled by best estimate accident codes such as MAAP 3.0B (reference 26) and MAAP 4.0 (reference 27), and with the technical basis for other core damage accident fission product source assessments such as the NRC's Light Water Reactor (LWR) Accident Source Term (reference 28), EPRI's Technical Basis for Severe Accident Management (reference 29) and the passive light water reactor fission product source term (reference 30).

Table 5
Expected Core Behavior During an Accident

Core Temperature	Core Behavior
5400°F	
4800°F	Melting of fuel pellets
4200°F	Release of all volatile fission products from fuel
3600°F	Possible formation of uncoolable core
3000°F	Fuel pellets dissolve in melt components
2400°F	Very rapid release of volatile fission products from fuel pellets
1800°F	Very rapid Zirc water reaction; formation of hydrogen and failure of fuel rod cladding
1200°F	Possible fuel cladding burst - release of fission products in gap space
	Little possibility of fuel cladding rupture

The remainder of this section provides a summary of the behavior of the parameters measured by a post accident sampling system that meets the requirements described by NUREG-0737.

2.2 RCS AND CONTAINMENT SUMP BORON

Prior to an accident, the reactor coolant system contains some amount of boron, in the form of boric acid, to control core reactivity. The amount of boron in the reactor coolant system prior to an accident is dependent on the burnup of the fuel in a given fuel cycle. The RCS boron concentration at the beginning of a fuel cycle may be in excess of 1500 ppm, while at the end of a fuel cycle, the boron concentration can be less than 10 ppm. Following a safety injection signal, the emergency core cooling pumps would inject water from the Refueling Water Storage Tank (RWST) which typically contains boron at a concentration greater than 2000 ppm. If safety injection (SI) continues to run (e.g., a LOCA), the RCS boron concentration will approach the RWST boron concentration. If there is no safety injection (SI) (e.g., no SI signal or failed SI injection) or if safety injection is terminated per plant Emergency Operating Procedures, then the RCS boron concentration may be significantly below the RWST boron concentration.

Assuming that the reactor vessel is filled with water, there are two accident sequences where the RCS boron concentration may not be representative of the boron concentration in the reactor core: steam generator tube rupture and some large LOCA events. In the tube rupture, the boron concentration in the loop with the ruptured Steam Generator (SG) tube may be lower than the boron concentration in the other loops due to: a) higher circulation rates in the reactor coolant loops with intact steam generator tubes during the reactor coolant system cooldown, and b) possible backflow of unborated (or low concentration boron) water from the steam generator to the reactor coolant system through the ruptured steam generator tube. In the case of the large LOCA with the break in a reactor coolant loop cold leg, core cooling is accomplished by boiling water in the reactor vessel; excess safety injection flow is provided to the reactor vessel to assure that the core remains covered. However, the boiling will tend to concentrate the boron in the reactor vessel as the steam transported from the reactor vessel to the containment will contain only limited boron (i.e., only a small fraction of the water concentration). For these large LOCA events, the RCS cannot be refilled completely, since excess safety injection flows out of the RCS break and the RCS boron concentration may vary greatly depending on the safety injection flow paths and the location of the RCS break.

If the accident results in discharge of water from the reactor coolant system, the water in the containment sump will contain boric acid. However, if the discharge from the reactor coolant system is steam (which will be condensed in the containment and the condensate will be returned to the containment sump area), the containment sump water will contain a lower concentration of boron. Thus, the containment sump boron concentration is highly dependent on the details of the accident scenario and may range from near zero ppm to over 2000 ppm. For the Westinghouse plants with ice condenser containments, the boron in the containment sump will be much closer to 2000 ppm due to unique plant design features. However, for most accident sequences in non-ice condenser containment plants, the reactor coolant and containment sump boron concentrations are closely inter-related. A low containment sump boron concentration is indicative of a concentrating mechanism in the reactor coolant system (e.g., relief from the RCS is only steam). Thus, if the containment sump water is used for safety injection in the recirculation mode, the boron concentration in the sump water may be of interest, but should not be a factor in choosing an accident recovery strategy. In this case, the use of low concentration borated water from the containment sump is offset by the very high concentration of boron in the reactor vessel water and subcriticality is still assured.

If the core is ex-vessel, the reactor coolant system cannot be refilled due to the breach in the bottom head of the reactor vessel. In this case, RCS boron concentration is not meaningful. However, diagnosis of an ex-vessel core condition is very difficult as discussed in the background material for the WOG SAMG (reference 24) and is not required to implement the appropriate accident management strategies after core damage has occurred. In this case, the containment sump boron, as indicated by samples of containment sump fluid, is likely to show a long term slow decrease in concentration. This is due to the buildup of boron in the core debris as the sump water is boiled to remove heat from the core debris. The steam, which is condensed by containment heat sinks and returned to the containment sump, will dilute the boron in the containment sump water thereby reducing its concentration. As discussed in the EPRI Severe Accident Management Technical Basis report, subcriticality is not an issue due to both the compaction of the core material and the boron buildup in the core debris.

The only other accident scenarios that can lead to anomalies in the reactor coolant or containment sump boron concentrations are beyond design basis events where unborated water is introduced into the reactor coolant system or the containment. Introduction of unborated water to the reactor coolant system is permitted in the plant EOPs (specifically at ECA-1.1, "Loss of Emergency Coolant Recirculation") if ECC recirculation is not available and the RWST level is approaching empty. In this case, the EOPs instruct the emergency response crew to refill the RWST using any available water source. The same guidance is provided in the plant SAMG. However, in this case, the plant staff is highly aware of the potential for recriticality and, as explained in Section 5.6, existing plant instrumentation is more effective in monitoring the potential for a return to criticality. Introduction of unborated water to the containment can also occur if there is significant leakage in service water or component cooling water lines (e.g., the lines supplying cooling water to the containment fan cooler units), or if the emergency response staff chooses an unborated water option in the SAMG to bring the containment sump water level to the plant specific target value in the SAMG. The plant staff would become aware of the former situation by an indication of excessive containment sump level. In the latter case, the actions are taken with full knowledge of the plant staff. In either case, the decision to use the resultant low boron concentration water for core cooling would be made with a knowledge of the potential for recriticality. The existing plant instrumentation would permit real-time monitoring of the potential for recriticality under these conditions.

2.3 RCS DISSOLVED GASES

Following an accident, significant amounts of noncondensible gases may be present in the RCS under two distinctly different sets of circumstances: a) nitrogen from accumulator injection, and b) hydrogen from metal water reactions between overheated fuel rod cladding and the steam in the reactor vessel. For accident sequences in which there is significant relief from the reactor coolant system, the noncondensible gases will be discharged from the reactor coolant system to the containment. However, for accident sequences in which there is little or no discharge from the reactor coolant system, the noncondensible gases may accumulate at one of the high points of the reactor coolant system.

When the reactor coolant system is at a high pressure, the noncondensible gases will either be dissolved in the reactor coolant system water or they will accumulate in very small pockets. However, as the reactor coolant system is cooled down and depressurized following an accident, the noncondensible gases will come out of solution as the pressure and temperature are decreased and the small accumulations of noncondensible gases will enlarge as the pressure is decreased. The issue is whether the noncondensible gas accumulations can either: a) result in core uncovery due to the growth of a large accumulation of noncondensible gases in the reactor vessel head, or b) block cooling of the core, especially in the natural circulation mode of cooling. With respect to natural circulation, emphasis is placed on establishing this core cooling mode following an accident in which high levels of radioactivity are contained in plant fluids as a result of core damage, because it does not involve circulating radioactive reactor coolant outside of the containment where leakage to the environs can occur. The recirculation of radioactive fluids outside of the containment can potentially contaminate plant areas or result in increased offsite radiation levels as a result of the leakage. Thus, recirculation of water outside

containment, while part of the plant design basis, is not desirable if an alternative means is available.

2.4 REACTOR COOLANT AND CONTAINMENT SUMP PH LEVELS

Following an accident, the pH of the reactor coolant is important only if chlorides may have been introduced into the water by either: a) recirculation from the containment sump where leakage of brackish or seawater from cooling systems may have occurred, b) use of non-demineralized makeup water for the reactor coolant system, or c) backflow of water through a ruptured steam generator tube. The presence of chlorides in water with a low pH (e.g., unbuffered boric acid) solution at temperatures near ambient boiling can result in stress corrosion cracking of stainless steel piping. Evidence shows that cracking would not be expected for at least 48 hours after the introduction of chlorides into the water (reference 31). The pH of the containment sump is important for the same reason, since it can be drawn through stainless steel piping when emergency core cooling recirculation is in use for long term heat removal from the core. The pH of the containment sump water is also important from the perspective of iodine retention in sump water. Any radioactive iodine that is stripped from the containment atmosphere following an accident is held in solution in the containment sump water. If the pH of the sump water is too low, the radioactive iodine may evolve from the water and go back into the containment atmosphere where is it can be more easily released from the containment to the environment.

For the design basis LOCA event, the pH of the reactor coolant and the containment sump are usually adjusted by automatic addition of a buffering solution that is a very strong base (e.g., sodium hydroxide) via the containment spray system. For ice condenser plants, the ice baskets contain a buffering solution that is released as the ice melts. In the design basis LOCA for most plant designs, the spray is automatically initiated by high containment pressure and the contents of the spray additive tank are "educted" into the spray flow to the containment. For those plants that do not have a spray additive tank, other passive means, such as baskets of trisodium phosphate in the containment sump, are used to ensure that the pH of the containment sump water following a design basis LOCA is within specified limits.

Accident sequences where the pH is a concern are those sequences in which there is a discharge of reactor coolant to the containment. However, since there are automatic means to adjust the containment sump pH for design basis accidents, these accidents do not have the highest potential for consequences due to a low pH. For accident sequences where either: a) additional water (above that contained in the RWST) is injected into either the containment or the reactor coolant system, or b) where containment spray is not operated (this condition is not applicable to ice condenser plants), special efforts are required to assure that the proper long term pH is achieved. As noted in the boron discussion, both the EOPs and SAMG recommend the refilling of the RWST to continue safety injection and/or injection of water into the containment.

2.5 CONTAINMENT HYDROGEN

If the core temperatures exceed about 1800°F as a result of the inability to cool the core following an accident, the reaction between the zirconium fuel rod cladding and the steam in

the reactor vessel will produce hydrogen. The hydrogen from zirconium water reactions is introduced into the reactor coolant system where it would be released to the containment through any points in the reactor coolant system where the pressure boundary integrity is not maintained. The principle means of releasing hydrogen to the containment is via a break in the reactor coolant piping as a result of the accident, a high reactor coolant system pressure that causes the pressurizer safety valves to open, the opening of the pressurizer power operated relief valves (PORVs) or the use of the reactor vessel head vent. The use of the pressurizer PORVs and the reactor vessel head vent are governed by the plant Emergency Operating Procedures and the plant Severe Accident Management Guidance.

If the hydrogen accumulates in the containment and reaches a concentration of approximately 4 percent by volume, an ignition source (such as a spark from a relay closing) can result in the burning (deflagration) of the hydrogen in the containment. At the 4 volume percent level, the burn would not completely consume all of the hydrogen in the containment. As discussed in the EPRI Severe Accident Management Technical Basis report, at hydrogen concentrations of 6 volume percent or greater all of the hydrogen in the containment can be consumed in a burn. The burning of hydrogen is an exothermic reaction that adds heat (energy) to the containment; the amount of heat generated by a hydrogen burn is directly proportional to the amount of hydrogen in the containment. Most of the heat generated during a hydrogen burn is initially added to the containment atmosphere; only a small portion of the heat is expected to be absorbed by structures via radiation heat transfer from the flame front. A hydrogen burn in a PWR containment is predicted to consume all of the hydrogen in the containment in about 10 to 15 seconds. Over this time period, the amount of heat removed from the containment atmosphere via convective heat transfer can be shown to be small in comparison to the total heat added to the containment. In a bounding sense, a hydrogen burn can be treated as an adiabatic process (no heat removal by containment structures). This bounding estimation is not significantly different from a more detailed assessment that includes the effects of heat sinks because the hydrogen burn occurs very rapidly.

Analyses (e.g., reference 33) indicate that an uncovered core can proceed from a point just prior to the onset of significant hydrogen generation (e.g., all fuel rod cladding temperatures less than 1800°F) to core melting in a matter of about 30 minutes. As shown in Table 6, for a core damage accident for a typical Westinghouse PWR, the amount of hydrogen generated may range from as little as about 200 pounds, to as much as 1500 pounds, depending on the accident sequence.

These analyses of core damage accidents also indicate that most of the hydrogen generated in the core region during a core damage accident is quickly released to the containment. As shown in Table 6, for the case of a core damage accident in which there is a breach in the reactor coolant system (e.g., a LOCA), almost 100% of the hydrogen generated is released to the containment at the same rate that it is being generated in the core by zirconium water reactions; very little of the hydrogen is retained in the reactor vessel or reactor coolant system. For a transient event where there is no breach of the reactor coolant system, the actuation of the pressurizer relief or safety valves will result in a significant fraction of the hydrogen generated in the core region not being released to the containment. Most transient events go to core damage because the steam generator heat sink is lost (i.e., the steam generators dry-out). When this occurs, the core decay heat goes to raising the pressure and temperature of the RCS water until the pressurizer relief or

safety valves open and discharge reactor coolant system inventory to the containment (via the pressurizer relief tank).

Table 6
Comparison of Hydrogen in the RCS and Containment for a typical Westinghouse NSSS PWR

	Large LOCA			Transient		
Time (1) (minutes)	Produced (lbm)	RCS Inventory (lbm)	Containment Inventory (lbm)	Produced (lbm)	RCS Inventory (lbm)	Containment Inventory (lbm)
0	0	0	0	0	0	0
5	40	10	30	30	15	15
10	77	12	65	50	25	25
15	111	11	100	70	35	35
20	147	12	135	120	60	60
25	173	3	170	600	300	300
30	173	2	171	1300	600	700
35	174	1	173	1450	550	900
40	175	<1	174	1475	525	950
45	176	<1	176	1480	520	960
50	177	<1	177	1480	510	970
55	178	<1	179	1480	475	1005
60	180	<1	180	1480	450	1030

(1) Time is relative to the onset of significant zirc water reaction

Analyses of core damage accidents show that the pressurizer relief or safety valves continuously modulate open and closed prior to and after core damage has occurred. This results in the continual release of hydrogen to the containment, but as much as 50% of the hydrogen generated in the reactor vessel by zirconium water interactions is trapped in the reactor coolant system.

The containment hydrogen inventory is further complicated by operator actions as directed by the Emergency Operating Procedures and the Severe Accident Management Guidance. As shown in reference 33, the operator actions to open the pressurizer PORVs if the reactor coolant system is at high pressure after core damage has occurred would result in the hydrogen that is trapped in the reactor coolant system being released to containment over a time period of as little as 15 minutes. For the case in which 50% of the hydrogen produced by zirconium water reactions was trapped in the reactor coolant system, the containment hydrogen inventory would double in 15 minutes.

Another factor in the progression of core damage accidents is the impact of recovery of an overheated core in the reactor vessel on the RCS and containment hydrogen inventories. As discussed in the EPRI Severe Accident Management Technical Basis report, in the case of the Three Mile Island accident, the hydrogen generation during the initial core overheating, but prior to starting the 'B' reactor coolant pump, was on the order of 100 to 150 pounds of hydrogen. If all of this hydrogen were released to the containment, the containment hydrogen concentration would have been in the range of 2 to 3 volume percent, which is not flammable under any conditions. As a result of starting the 'B' reactor coolant pump and reflooding the core, an additional 200 to 250 pounds of hydrogen were generated over a time frame of less than 10 minutes. With the addition of this hydrogen to containment, the containment atmosphere quickly became flammable and, in fact, a burn occurred shortly after this time. During the subsequent second uncovery and final reflood, less than 50 additional pounds of hydrogen were generated. The point is that with the potential for very rapid hydrogen generation during in-vessel recovery from an overheated core condition, the containment flammability can change very quickly.

Additional hydrogen may be generated if the progression of a core damage accident cannot be arrested while the core is retained within the reactor vessel. If the reactor vessel fails and the core material drains to the containment, it will lie in contact with the concrete basemat. The ex-vessel core material will continue to produce decay heat and if it cannot be cooled, the underlying concrete will begin to heatup and decompose. The decomposition of the concrete basemat leads to additional noncondensible and flammable gas generation. Hydrogen can be generated from water in the concrete that is released during decomposition and passes through (as steam) overlying molten core debris. It can react with any zirconium, chromium or nickel in the core debris to form hydrogen. Hydrogen can also be generated as the rebar in the concrete basemat is contacted by the core debris. Flammable carbon monoxide can be generated from the decomposition of the limestone aggregate used in the concrete for some containments. Overall, the flammable gas generation from core concrete interactions can equal that generated in the reactor vessel as the core melts. However, the rate of hydrogen generation (and thus the rate of change of the containment hydrogen inventory) from ex-vessel conditions is very slow compared to the in-vessel hydrogen transient. In the ex-vessel case, "hours" to "days" are required to double the containment hydrogen inventory.

Most plants have hydrogen recombiners for controlling the containment hydrogen concentrations from a design basis accident where the in-vessel zirconium water reaction is very limited. For these design basis accidents, the containment hydrogen that must be controlled to prevent a flammable containment condition is the very slow hydrogen generation (tens of pounds per hour) from metal corrosion in the containment sump and from radiolysis of water circulating through the reactor vessel. Thus, the recombiners have a very limited capacity (e.g., 100 cubic feet of containment atmosphere per minute) for eliminating hydrogen from the containment atmosphere. Analyses have shown that recombiners have no impact on containment hydrogen concentration during the early stages of a core damage accident. Because of the small containment volume of the ice condenser containments, those plants have installed hydrogen igniters which keep hydrogen levels in containment at very low levels following a core damage accident.

The Individual Plant Examinations for each plant show the potential for a hydrogen burn to challenge the integrity of the containment, as discussed in NUREG-1560. In general, for PWRs with large dry containments, the potential for challenging containment integrity due to a hydrogen burn only occurs for those cases in which there is ex-vessel core concrete interactions. However, if large quantities of hydrogen are generated during in-vessel recovery from a core damage accident, some of the PWR large dry containments may be challenged due to the potential burning of hydrogen. For core damage accident sequences that progress to the point of reactor vessel failure and core concrete interactions, most PWR large dry containments can be challenged due to burning of flammable gases that accumulate in the containment. For the ice condenser containment plants, there is no containment challenge due to burning of hydrogen as long as the hydrogen igniters have been operational since the beginning of core overheating. If the hydrogen igniters are not operational, the potential for a challenge to the containment integrity due to burning of hydrogen is quite high.

2.6 FISSION PRODUCT INVENTORIES AND BEHAVIOR

Before getting into the diagnosis of core damage, we need to establish some basic information regarding fission product inventories and behavior.

2.6.1 Behavior in the Reactor Core

Radioactive fission products can be divided into 6 broad groups, based on their chemical properties and their potential for health effects (reference 32):

- Noble gas - These radioactive species exist as a pure gas at all accident conditions and do not react with other materials to form new compounds.

- Iodine - These radioactive species exist as cesium iodide in the fuel matrix. Cesium iodide is a volatile substance with a melting point of about 1150°F and is quite volatile at temperatures over 2000°F. A small amount of the cesium iodide can also react with organic substances to form various organic iodide compounds which can be extremely volatile under accident conditions.

- Cesium - These radioactive species exist as cesium iodide and cesium uranate in the fuel matrix. Under oxidizing conditions, the cesium uranate would become cesium hydroxide. Both of these are volatile species under accident conditions, with cesium hydroxide having a melting point of about 525°F.

- Tellurium - These radioactive species may exist as a solid in the fuel matrix or as cesium telluride. In either case, tellurium combines with metallic zirconium at accident temperatures. As the zirconium is oxidized, the tellurium is liberated and it too oxidizes to form a tellurium oxide. Tellurium oxide has a low volatility under most accident conditions.

- Strontium - These radioactive species exist as a solid in the fuel matrix and are known to be volatilized under accident conditions. They readily oxidize to form strontium oxide which has a low volatility under accident conditions.

- Ruthenium - These radioactive species exist as a solid in the fuel matrix and have a very low volatility under accident conditions. The very small amount that might be volatilized under accident conditions will be oxidized in the steam environment, and the resultant oxide would likely exist only as an aerosol.

The gaseous and volatile radioactive species exist primarily within the solid ceramic fuel pellets during normal operation. However, a small fraction of these species exist in the gap space between the fuel pellets and the fuel rod cladding as a result of migration of these volatile fission products out of the fuel pellet matrix. On the other hand, almost 100% of the non-volatile species are retained within the ceramic fuel pellet matrix during normal operation.

During normal operation, a small amount of the volatile fission products in the fuel rod gap space can be released to the reactor coolant system if a small defect in the fuel rod cladding occurs. It has also been observed, as discussed in WCAP-8637 (reference 23), that if there are one or more fuel rods in the core with defects, a significant change in the reactor coolant iodine and cesium fission product concentrations occurs anytime the reactor is tripped or the reactor coolant system is depressurized. While this latter phenomena is termed iodine spiking, the reactor coolant system noble gas and cesium concentrations also increase upon depressurization of the reactor coolant system.

The current understanding of fission product behavior during an accident is not significantly different than that given in reference 32. NUREG-1465 (reference 28) provides the most recent regulatory understanding of fission product behavior in the core and, together with the Industry Degraded Core Rule-Making (IDCOR) report (reference 34), provides the basis for the discussion below.

During an accident, if a failure of the fuel rod cladding of one or more fuel rods occurs, the gaseous and some of the volatile fission products in the fuel rod gap space will be released (termed gap release) to the reactor coolant system. In addition, some gaseous and volatile fission products that reside in the fuel pellet near the outer periphery will also be released with or shortly after the fission products in the cladding gap space (called embedded gas release). The overall release has been termed burst release. For noble gases, the burst release can be between 1% and 25% of the total rod inventory of noble gases. Cladding failure would result in 0.04 to 5% of the rod iodine inventory and 0.02 to 5% of the rod cesium inventory being released to the reactor coolant system as a burst release. The iodine and cesium release quantities are highly dependent on the pre-accident operating characteristics of the rods and the temperatures experienced just prior to rod burst. The other fission products discussed above would not be expected to be released to the reactor coolant system in significant quantities for fuel rod cladding failures. With respect to the burst release of noble gases, iodines and cesiums, NUREG-1465 recognizes an immediate burst release of 3% of the rod inventory that is assumed to occur at the time of fuel rod failure, and an additional long term release of 2% of the rod inventory over the next 30 minutes.

Further, if the fuel pellets begin to overheat during an accident, additional gaseous and volatile fission products may be released from the fuel pellet matrix, through the ruptured fuel rod cladding and into the reactor coolant system. This phenomena is termed diffusional release. When the fuel pellet temperature exceeds approximately 2465°F, the noble gases, iodines and cesiums accumulated at the grain boundaries of the fuel pellets will begin to be released. For fuel rods operating at low power levels during the pre-accident period of operation, fuel rod temperatures as high as 3250°F might be required to effect the grain boundary release. As much as 100% of the rod inventory of noble gases, iodines and cesiums may be released via this mechanism.

At the completion of the burst, diffusional and grain boundary releases, very little of the rod inventory of noble gases, iodines and cesiums may still remain in the fuel pellet grains. Release from the grains is insignificant at clad burst temperatures (e.g., 1400 to 2000°F), but the release rate is a strong function of temperature. At about 3650°F, the release rate can be as high as 10% of the remaining inventory per minute. At this rate, the release of noble gases, iodines and cesiums from the fuel pellets is nearly 100% complete by the time the fuel reaches its melting temperature of 4000°F. The low volatility fission products would not be released in significant quantities from the fuel grains to the reactor coolant system, even after fuel melting has been initiated, due to their low volatility.

From the perspective of emergency response, there are only three levels of core damage that are important: no damage, fuel rod cladding damage and fuel overtemperature damage. The majority of the noble gas and volatile fission products are already released from the fuel prior to the onset of core melting. These are also the most important fission products (i.e., noble gases, iodines and cesiums) from an offsite radiological protection perspective. The small quantities of nonvolatile fission products that may be released only at core melting are not as important with respect to emergency response activities.

2.6.2 Behavior in the Reactor Coolant System and Containment

During core degradation under core overheat conditions, as fission products are released from the fuel rod cladding or the fuel pellets, they are swept from the active core region by the circulation of gases in the reactor vessel. The circulation of the superheated gases in the reactor vessel, which are primarily steam and hydrogen, moves the fission products to the upper plenum region of the reactor vessel. As the gases carrying the fission products enter the upper plenum, they mix with cooler gases from other regions of the core. Since the metal structures in the upper plenum region are cooler than the fission product laden gases, heat transfer occurs from the hot gases to the cooler metal surfaces. This gives rise to either condensation of the volatile and nonvolatile fission products onto the metal surfaces, or condensation in the gas phase on nucleation sites. The noble gas fission products are unaffected by these removal processes. Condensation on the metal surfaces effectively immobilizes some fission products, while those that condense in the gas phase form aerosols that remain suspended in the gas and can be carried out of the upper plenum region. There are a number of removal processes that can remove the aerosol fission products from the gas stream; the primary mechanism is gravitational settling on surfaces, both in the reactor coolant loop and in the containment. The

exact accident scenario will determine the relative amounts of aerosol fission products in each location.

Once fission products have been removed from the gas stream and deposited on a surface (either in the reactor vessel, the reactor coolant loop, or the containment), there are several mechanisms that can result in their re-introduction into the gas stream. Within the reactor vessel and reactor coolant system, if the decay heat from the deposited fission products cannot be absorbed by the surface where they are located, the fission product material will begin to heat up. For the volatile fission products, this can result in their revaporization. The fission products will again be in the gas stream where they can again condense on colder surfaces or form aerosols. For those fission products that have formed aerosols and deposited on surfaces, the aerosols can be re-entrained in the gas flow if the velocity of the gas flow is sufficient. Finally, if water is washed across the surface on which the fission products are deposited, the fission products can be entrained in the water flow. Thus for example, if the RCS is refilled with water to recover a degraded core or if containment sprays are operated to depressurize the containment, deposited fission products will be washed off of the surfaces on which they are adhering and will be present in the water as suspended aerosols.

The point of the above discussion is that all fission products are quite mobile within the reactor vessel and reactor coolant loops during a core damage accident. During core overheating, the coolant loops are filled with steam and hydrogen. The flow from the reactor vessel is due only to the boiling off of the remaining water or, in the case of an intact RCS, natural circulation flow. These flows are relatively weak compared to the break flow from a LOCA, and allow a significant residence time for volatile and nonvolatile fission products to deposit in the upper reactor vessel and coolant loops. To illustrate, the results of several MAAP runs, as documented in reference 33 are discussed below. For an accident sequence in which the reactor coolant system remains at the pressurizer PORV setpoint during the time that fission products are being released from the fuel rods, upwards of 98% of the core inventory of iodines and cesiums are deposited on the internal surfaces of the reactor coolant system. Perfect measurement for iodine and cesium in the containment would only "see" 2% of the iodine and cesium core inventory for a completely melted core. However, if the control room operators had opened the pressurizer PORV at the time fission products were being released from the core, as much as 50% of the iodines and cesiums would be transported to the containment. In the hypothetical case of a hot leg large LOCA, as much as 98% of the iodines and cesiums would be transported to the containment while for a cold leg LOCA, the release to containment is on the order of 10% to 50%. Thus, the amount of volatile fission products in the containment vs. the reactor coolant system is a strong function of the size and location of openings in the RCS at the time fission products are released from the fuel rods.

3 DIAGNOSIS OF PLANT CONDITIONS

Sampling of plant fluids using the Post Accident Sampling System (PASS) can provide information on the chemical and radioactive content of fluids in the reactor coolant system, containment sump and containment atmosphere. The impact of the core damage accident on the radioactive content of the samples can be broken down into two time phases: (a) during core degradation and core damage, and (b) after the core has been recovered.

3.1 DIAGNOSIS DURING CORE DEGRADATION

During core degradation, the reactor coolant system will not contain water, but rather a steam and hydrogen mixture along with radioactive gases and aerosols. Some limited water may be present on the inner surfaces of the steam generators and reactor coolant system piping as a water liquid film. During this time, any sample of the reactor coolant system will be primarily gaseous and contain only the fission products that are still in the reactor coolant system and which have not deposited on reactor coolant system piping. Since most of the chemicals of interest are not represented in a gas sample of the RCS, the samples would provide no useful information related to the RCS chemistry during this phase of the accident. As discussed in the previous section related to fission product behavior in the RCS (Section 2.6.2), an RCS sample would contain a variable amount of noble gases, iodines and cesiums. Results of severe accident analyses discussed previously show that the iodine and cesium inventory in the reactor coolant system may range from less than 10%, to over 90% of the total core inventory, while the noble gases might range from less than 1%, to 50% of the total core inventory. During the transport of the iodines and cesiums to the sample station (through small diameter lines that may or may not be heat traced), much of the iodine and cesium would be expected to deposit in the sample lines (see reference 34). The deposition of iodine and cesium in the sample lines would be expected irrespective of heat tracing, since the heat tracing does not raise the line temperature to the point where revaporization of the cesium iodide would occur (revaporization requires temperatures in the range of 800 to 1000°F). A second point related to use of the RCS samples during the time that the core is overheating and damage is progressing is that significant aerosol concentrations may exist in the RCS during these times (not only radioactive aerosols but other metal-based aerosols from overheating the core and reactor internals). As described in the IDCOR report (reference 34), the aerosol deposition in small diameter piping has the potential for plugging the line, thereby rendering the sample line unavailable for all future times. Thus, RCS samples taken during the time that core damage is occurring would not be expected to provide any useful information on noble gas, iodine or cesium inventories released from the fuel rods, and may result in the inability to obtain RCS samples at times after the core is recovered.

Containment airborne samples for determining the inventory of hydrogen and fission products released from the fuel rods may provide some useful information related to noble gas inventories, but are expected to provide little or no useful information related to the other volatile and nonvolatile radionuclides. As discussed previously in the section on the behavior of fission products in the RCS and containment (Section 2.6.2), for all core damage accident sequences except steam generator tube rupture and LOCA outside containment, the hydrogen

and noble gases are predicted to be released to the containment at approximately a rate proportional to their release rate from the fuel rods during core damage. As much as 50% of the hydrogen and noble gases, and as much as 90% of the volatile fission products may be held up in the reactor coolant system for events that occur at a high reactor coolant system pressure. Thus, knowing the containment hydrogen and noble gas fission product inventory may provide a key to estimating the amount of core damage that has occurred. However, for ice condenser containment plants which have hydrogen igniters to prevent accumulation of hydrogen in the containment, the containment hydrogen is not a reliable indicator of the amount of core damage. On the other hand, there is a large dependence on the details of the accident sequence when looking at the other volatile and nonvolatile fission products. In addition to deposition of these radionuclides in the reactor coolant system, there are significant deposition processes that take place in the containment. Deposition of fission products in the containment does not guarantee that they will be transported to the containment sump, as some of the processes are independent of steam condensation that is required to carry these deposited fission products to the containment sump. In addition, it is expected that there would be significant deposition of these volatile and nonvolatile fission products in the containment sample lines which, in turn: (a) results in the sample analysis not being representative of the containment atmosphere inventory, and (b) can lead to plugging of the sample line by deposited aerosols, which would render the sample line unavailable for future use. It should be noted that containment fission product sampling is not required for core damage assessment using the new WOG CDAG (reference 14).

Sampling of the containment sump for determination of the radionuclide inventories is not expected to provide any useful information for making an assessment of the degree of core damage. The containment sump water is not expected to contain any noble gases, and the iodine and cesium inventory in the sump water, as explained earlier, is highly dependent on the accident scenario. For example, if significant iodine and cesium is deposited on the internal surfaces of the reactor coolant system, they would not be carried into the containment sump. Also, for this reason it is difficult to attempt to identify the iodine and cesium in the reactor coolant system, containment atmosphere and containment sump, and then estimate the degree of core damage based on the total identified iodine and cesium. Thus, radionuclide analysis of samples of containment sump water is not expected to provide any useful details for assessing core damage and is not used in the new WOG CDAG.

Sampling the containment sump for chemistry could provide information related to the sump pH and the sump boron concentration. The most likely request for a chemistry sample is the boron concentration of the containment sump water when an accident management decision related to the use of emergency core cooling recirculation is under advisement. However, in this case, the time required to obtain a sample and report the boron concentration is likely to be longer than the time frame available for accident management decisions in this phase of the accident.

3.2 DIAGNOSIS AFTER CORE RECOVERY

The second phase of a core damage accident, from the perspective of core damage assessment from analysis of samples of plant fluids, begins after the reactor vessel is reflooded, and the

reactor coolant system is water filled. In the case of a LOCA where the reactor coolant system cannot be refilled, sampling (provided that it can be accomplished at low reactor coolant system pressures) is expected to provide no representative information on RCS chemistry or radionuclide inventories due to the large uncertainties concerning what the sample represents. Assuming that the reactor coolant system is refilled with water, most of the RCS chemicals and the volatile and non-volatile radionuclides deposited in the RCS would be expected to be dissolved (in the case of ionic compounds), suspended (in the case of aerosols) in the water, or remain coherently attached to the RCS piping. Provided that enough time has passed for the water to thoroughly mix the radionuclides, the samples may yield some information regarding the RCS chemistry, and the inventory of iodines and cesiums that were deposited in the reactor coolant system. However, in the case of radionuclides, the process of reflooding the core after core damage by injection of cold water onto an overheated core may result in additional cladding failure (due to thermal shock) and the subsequent release of additional gap activity. In the end, the question of the usefulness of the radiological information obtained from these samples must be asked. Given the large uncertainties in iodine and cesium fission product behavior during the accident (deposited in the reactor coolant system vs. airborne in containment vs. in containment sump water vs. deposited on containment surfaces), these samples of reactor coolant system water, at best, can only be used to confirm the order of magnitude of core damage estimates reached by other means. In the case of RCS chemistry, the issue of how well the samples represent the bulk RCS conditions must be addressed. There are two key points in making this determination: a) the potential for stagnation of fluids in one or more portions of the reactor coolant system, and b) the location of the sample point in relation to any water being injected into the reactor coolant system. The most useful information provided by RCS samples is for decisions related to long term cleanup after the accident.

For accident scenarios involving a breach in the reactor coolant system, recovery of the core by refilling the reactor vessel will result in some of the fission products that were deposited in the reactor coolant system being washed out into the containment sump. For scenarios that to not involve a breach of the reactor coolant system, only those fission products that are transported with the steam relief from the reactor coolant system would be present in the containment sump. In these cases, interpretation of the results of samples of containment sump water for volatile and nonvolatile fission product inventories suffers from some of the same uncertainties as the interpretation of sample results of reactor coolant water after core recovery. Again, the most useful information provided by RCS samples is for decisions related to long term cleanup after the accident.

3.3 SUMMARY

Based on the information presented above (Sections 3.1 and 3.2) it is concluded that, with the exception of the containment noble gases, analysis of samples for radionuclide inventories does not provide any significant information related to estimating core damage during or following recovery from a core degradation event, due to the behavior of the volatile and nonvolatile species under core damage accident conditions. That is, the transport and deposition of radionuclides during core degradation and following recovery has large uncertainties when viewed from the perspective of the knowledge of the exact conditions of the core, reactor coolant system and containment based on information available to the emergency response

teams during the accident (see the previous discussion on fission product behavior during an accident). While the radionuclide transport and deposition can be predicted with a degree of certainty by core damage accident models that track a specific accident scenario, the same information is not available to the emergency response teams during the accident due to instrumentation limitations. Therefore, only the analysis of the radioactive noble gases can provide potentially useful information for estimating core damage. However, as discussed in a previous section regarding hydrogen samples, the time delay involved in the analysis of samples only makes this information useful when a quasi-steady state has been achieved. At this time, the accident progression has been arrested and further escalation of offsite emergency response actions are not likely.

4 ACCIDENT MANAGEMENT GUIDANCE

As used in this report, accident management guidance refers to the entire realm of guidance and procedures at an operating nuclear power station that can be used in the event of an accident to bring the plant to a safe stable state and to protect the health and safety of the public in the area around the nuclear station. Using this definition, the accident management guidance consists of three major parts: the Emergency Operating Procedures (EOPs), the Severe Accident Management Guidance (SAMG), and the Site Emergency Plan. This section examines each part of the accident management program to determine the possible requirements for post accident sampling.

4.1 EMERGENCY OPERATING PROCEDURES

Following the accident at Three Mile Island, the NRC required that plant emergency operating procedures be upgraded to provide guidance on recovery from accidents that approach core damage (reference 3). In response to this regulatory requirement, the Westinghouse Owners Group developed a set of generic symptom-based emergency operating procedures to serve as a basis for the development of plant specific Emergency Operating Procedures (EOPs). The plant EOPs provide clear and concise steps for the control room personnel to stabilize the plant following an accident and then to bring the plant to a state where the plant can safely remain for a period of time while the plant status is evaluated and decisions can be reached regarding the possibility of going back to power, or to one of several shutdown conditions for repair and maintenance. The EOPs are divided into three parts: Optimal Recovery Procedures (designated by an E or ES procedure prefix), Contingency Procedures (designated by an ECA procedure prefix), and Functional Restoration Procedures (designated by an FR procedure prefix). Each part deals with plant conditions that become more degraded in terms of the ability to recover from the accident.

Within the EOP framework, there are several types of sampling that are required for diagnosis of plant conditions that are used to place the plant in a safe, stable condition. The normal sampling system would be used, unless there are high radiation levels. This section provides an examination of the impact of the ERG activities when the normal sampling system cannot be used due to high radiation levels.

The following discussion of the Westinghouse Owners Group generic Emergency Response Guidelines is based on Revision 1B (reference 11). Since the original issuance of this topical report, Revision 1c has been issued (Reference 39) which is based on an assessment of a significant quantity of feedback on usage of Rev. 1B during operator training and examinations. An assessment has been performed (see Appendix A) that concludes that all of the conclusions drawn with respect to the interaction of PASS with the Rev. 1B ERGs is bounding for the Rev. 1C ERGs.

4.1.1 RCS Boron

There are several places in the generic WOG Emergency Response Guidelines (ERGs) where the control room staff is instructed to obtain information regarding boron concentrations in the reactor coolant system. The purpose of these instructions is to assure that there is adequate shutdown margin.

The specific ERGs where such instructions are given include:

- ES-0.2, Natural Circulation Cooldown, Step 3, "Verify Cold Shutdown Boron Concentration (in the RCS) by Sampling".
- ES-1.2, Post LOCA Cooldown and Depressurization, Note prior to Step 7, "Shutdown margin should be monitored during RCS cooldown".
- ES-1.2, Post LOCA Cooldown and Depressurization, Step 21, "Verify Adequate Shutdown Margin, and step 21 (a), Sample RCS".
- ES-3.1, Post- SGTR Cooldown Using Backfill, Step 3, "Verify Adequate Shutdown Margin, step 3 (a), Sample Ruptured SGs, and Step 3 (b), Sample RCS".
- ES-3.2, Post- SGTR Cooldown Using Blowdown, Step 3, "Verify Adequate Shutdown Margin, Step 3 (a), Sample Ruptured SGs, and Step 3 (b), Sample RCS".
- ES-3.2, Post- SGTR Cooldown Using Steam Dump, Step 3, "Verify Adequate Shutdown Margin, Step 3 (a), Sample Ruptured SGs, and Step 3 (b), Sample RCS".
- ECA-0.1, Loss of All AC Power Recovery Without SI Required, Step 17, "Verify Adequate Shutdown Margin, Step 17 (a), Sample RCS".
- ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired, Step 25, "Verify Adequate Shutdown Margin, Step 25 (a), Sample Ruptured SG(s), and Step 25 (b), Sample RCS".
- ECA-3.2, SGTR With Loss of Reactor Coolant - Saturated Recovery Desired, Step 19, "Verify Adequate Shutdown Margin, Step 19 (a), Sample Ruptured SG(s), and Step 19 (b), Sample RCS".
- ECA-3.3, SGTR Without Pressurizer Pressure Control, Step 23, "Verify Adequate Shutdown Margin, Step 23 (a), Sample Ruptured SG(s), and Step 23 (b), Sample RCS."

The inability to obtain such a sample would not result in an unsafe plant condition, since insufficient boration would result in a slow approach to criticality that would be apparent from increasing source range and/or intermediate range ex-core neutron detector indications, and from an increase in the startup rate. This change in conditions would be diagnosed on the "Critical Safety Function Status Tree" (CSFST) which is part of the WOG ERGs. The ERG

CSFSTs are continually monitored and prompt action to resolve an abnormal condition is required. In this case, one of the FR-S procedures would be implemented which calls for boration of the reactor coolant system. Thus, while a decrease in shutdown margin may exist for a short period of time, the inability to sample RCS boron concentration would not result in a worsening of the accident conditions.

In addition, in the scheme of executing ERG instruction steps, if a particular step cannot be executed (for example, due to unavailability of the information), the step would be skipped and the procession through the subsequent ERG steps would continue. This is the case with all of the ERG steps described above, except for the ES-0.2 procedure.

In the ES-0.2 procedure, the inability to verify the cold shutdown boron concentration would result in execution of the "Result Not Obtained" (RNO) portion of that step which, in turn, prevents the execution of subsequent steps. In this case, the ERG developers thought that it was imperative that the reactor coolant boron concentration be verified prior to proceeding with natural circulation cooldown. However, it is very unlikely that this procedure would be in use if core damage has occurred due to core uncovery and heatup, since the procedure can only be entered if SI has been neither actuated nor required. In the cases of recovery after core damage that resulted from core uncovery and heatup, the procedures would direct the user to the ES-1.2, Post LOCA Cooldown and Depressurization procedure. In this procedure, the inability to ascertain the RCS boron concentration to verify shutdown margin would not result in a termination in the accident recovery. Thus, for the ES-0.2 procedure, the normal RCS sampling system could be used, since the procedure cannot be entered for accidents resulting in damage to the reactor core.

In the case of recovery following core damage where the core was not uncovered (e.g., an Anticipated Transient Without Scram (ATWS) event), the plant emergency response organization would be monitoring the ex-core nuclear instrumentation very closely. Any unexpected increase in the indications from the ex-core neutron detectors would be diagnosed and dealt with promptly. Thus, the need for RCS boron samples in this case is not required to ensure that a safe shutdown condition is achieved and maintained. Additionally, the boron sample would not provide a "first" indication of a degrading accident condition; any degradation of conditions would first be indicated by the ex-core neutron detectors.

Thus, the ability to obtain the results of a reactor coolant system boron sample, following an event that results in high radiation levels in the RCS, is not required for achieving a safe stable state.

4.1.2 Plant Status

There are several places in the generic WOG ERGs where the control room staff is instructed to obtain information regarding the plant status from samples of plant fluids. The purpose of these instructions is to provide additional information to the plant emergency response staff regarding the overall plant conditions prior to exiting the ERGs and going into a less structured long term recovery mode of emergency response. The specific ERGs where such instructions are given include:

- E-1, Loss of Reactor or Secondary Coolant, Step 12 (c), "Obtain Samples [Enter Plant Specific List]".
- E-1, Loss of Reactor or Secondary Coolant, Step 19, "Evaluate Long Term Plant Status: Consult Plant Engineering Staff".
- ES-1.2, Post LOCA Cooldown and Depressurization, Step 32, "Evaluate Long Term Plant Status: Consult Plant Engineering Staff".
- ES-3.1, Post- SGTR Cooldown Using Backfill, Step 12, "Evaluate Long Term Plant Status: Consult Plant Engineering Staff".
- ES-3.2, Post- SGTR Cooldown Using Blowdown, Step 16, "Evaluate Long Term Plant Status: Consult Plant Engineering Staff".
- ECA-2.1, Uncontrolled Depressurization of All Steam Generators, Step 43, "Evaluate Long Term Plant Status: Consult Plant Engineering Staff".
- ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired, Step 38, "Evaluate Long Term Plant Status: Consult Plant Engineering Staff".
- ECA-3.2, SGTR With Loss of Reactor Coolant - Saturated Recovery Desired, Step 32, "Evaluate Long Term Plant Status: Consult Plant Engineering Staff".
- ECA-3.3, SGTR Without Pressurizer Pressure Control, Step 37, "Evaluate Long Term Plant Status: Consult Plant Engineering Staff".

In the case of step 12(c) of the E-1 procedure, a survey of plant specific EOPs indicates that RCS samples are normally specified in the EOPs for a number of analyses according to each plant's sampling system capabilities. In the remainder of the procedure steps listed above, no further clarification is provided in the plant specific EOPs.

The inability to obtain and analyze specific samples would not preclude the ability to place or maintain the plant in a safe, stable state. In all of the procedure steps listed above, if the results of samples are not available, the procedure steps are continued. In the case of the instruction to obtain sample results to evaluate the long term plant status, this instruction is at or very near the end of the procedures. No subsequent procedures require knowledge of sample results for possible actions. Thus, the inability to provide the control room staff using the EOPs with results of analyses of samples of plant fluids does not preclude placing the plant in a safe stable state following a core damage accident.

4.1.3 RCS Gases

There is one place in the generic WOG ERGs where the control room staff is instructed to obtain information regarding the potential noncondensible gas content of the reactor vessel head. The purpose of this instruction is to provide information regarding the use of the reactor vessel head

vent to prevent a disruption of core cooling flow when safety injection is still in use. The specific ERG where such instruction is given:

- E-1, Loss of Reactor or Secondary Coolant, Step 16, "Determine if Reactor Vessel Head Should be Vented: Consult Plant Engineering Staff".

The placement of this step is after a determination has been made, based on plant parameters such as RCS pressure and safety injection flow, that continued long term core cooling can only be achieved using safety injection and safety injection recirculation. Thus, this step has no bearing on the potential for noncondensible gases to inhibit long term cooling of the core using natural circulation in the reactor coolant system. The ERG Background Document for this step discusses the basis for making the determination of the need for reactor vessel head venting based on plant instrumentation. There are no steps where samples of reactor coolant for dissolved gases are required, or even helpful, in making this determination. Thus, the inability to sample and analyze the reactor coolant fluid for dissolved gases does not impact this procedure step.

For plant conditions where natural circulation cooling is an alternative method for long term cooling of the core, the assessment described in the ERGs is based solely on plant parameters obtained from instrumentation; the potential for noncondensible gases in the reactor coolant system to accumulate in the reactor coolant system and inhibit natural circulation cooling is not part of the decision process. For plant conditions in which natural circulation can provide long term core cooling, the ERGs describe a series of steps, based on plant parameters obtained from instrumentation (such as temperatures, pressures and levels) that assures that natural circulation can be established and is an effective heat removal process. If natural circulation cooling cannot be established, the ERGs provide an effective means for diagnosis, based on plant instrumentation, and establishing another method of long term core cooling. In the event that voids form in the reactor vessel upper head, the WOG ERGs provide a method of diagnosis and instructions for removing the voids. This condition is diagnosed in the symptom based functional restoration procedures from Functional Restoration Status Tree F-0.6, "Inventory" and recovery instructions are provided in the associated Functional Restoration Procedure FR-I.3, "Response to Voids in Reactor Vessel". This symptom-based diagnostic and recovery is based solely on plant parameters obtained from instrumentation.

The WOG ERGs have been specifically evaluated for the case of noncondensible gas accumulation in the reactor coolant system. The results of the evaluation show that the WOG ERGs provide an effective means to diagnose the conditions in which natural circulation cooling may be inhibited by noncondensible gas accumulations, and to provide instructions for establishing alternate core cooling processes.

Thus, the ability to obtain the results of a reactor coolant system dissolved gas or hydrogen sample is not required, nor suggested, by the EOPs for achieving a safe stable state following an accident.

4.1.4 Steam Generator Tube Integrity

There are several places in the generic WOG ERGs where the control room staff is instructed to obtain information regarding radioactivity on the secondary side of the steam generators. The purpose of these instructions is to provide information regarding the integrity of the steam generator tubes. Note that this section is included for completeness; there is no regulatory requirement for post accident sampling of the steam generator secondary side. The specific ERGs where such instructions are given include:

- E-2, Faulted Steam Generator Isolation, Step 6, "Check Secondary Radiation", and Step 6 (a), "Request periodic activity samples of all SGs: [Enter plant specific means]".

- E-3, Steam generator Tube Rupture, Step 2, "Identify Ruptured SG(s): High radiation from any SG sample".

- FR-H.3, Response to Steam Generator Overfill, Step 7, "Check Affected SG(s) Radiation [enter plant specific means]".

The inability to obtain such a sample would not result in an unsafe plant condition since the radioactivity on the secondary side of the steam generator could be detected by either the main steam line radiation monitor, the condenser air ejector radiation monitor or the steam generator blowdown radiation monitor. In addition, there are no procedure steps that require knowledge of the results of these sample analyses before proceeding with subsequent recovery steps.

4.1.5 Containment Hydrogen Concentration

There are several places in the generic WOG ERGs where the control room staff is instructed to obtain information regarding hydrogen concentration in the containment. The purpose of these instructions is to provide information regarding the possible approach to flammable conditions in the containment so that actions can be initiated to preclude the establishment of flammable conditions in the containment. The specific ERGs where such instructions are given include:

- FR-C.1, Response to Inadequate Core Cooling, Note prior to Step 8, "This guideline should be continued while obtaining a hydrogen sample in Step 8, Check Containment Hydrogen Concentration", and Step 8 (a), "Obtain hydrogen concentration [enter plant specific means]".

- FR-C.1, Response to Inadequate Core Cooling, Step 8 (b) RNO (if containment hydrogen concentration is greater than 6 volume percent), "Consult plant engineering staff for additional recovery actions".

- FR-C.1, Response to Inadequate Core Cooling, Step 8 (c) RNO (if containment hydrogen concentration is greater than 0.5 volume percent), "Turn on Hydrogen Recombiner System".

- FR-Z.1, Response to High Containment Pressure, Step 7, "Check Hydrogen Concentration", and Step 7(a), "Obtain a current [containment] hydrogen measurement".

- FR-Z.1, Response to High Containment Pressure, Step 7 (b) RNO (if containment hydrogen concentration is greater than 6 volume percent), "Consult plant engineering staff for additional recovery actions".

- FR-Z.1, Response to High Containment Pressure, Step 7 (c) RNO (if containment hydrogen concentration is greater than 0.5 volume percent), "Turn on Hydrogen Recombiner System".

- FR-Z.1, Response to High Containment Pressure, Step 9, "Periodically Obtain a Hydrogen Concentration Measurement".

- FR-I.3, Response to Voids in Reactor Vessel, Step 12, "Obtain containment hydrogen concentration measurement: [enter plant specific means]".

Note that all of the above FR-Z.1 steps were recommended to be deleted from Revision 1B of the WOG ERGs in 1997 (ERG Maintenance DW-96-030). In the following discussion the reference to FR-Z.1 should no longer apply.

It should be noted that in all cases, the instruction only states that a value for containment hydrogen concentration should be obtained. Further, in the case of the FR-C.1 and FR-Z.1 procedures, the containment hydrogen measurement provides information related to a decision whether to turn on the hydrogen recombiners (containment concentration greater than 0.5 volume percent, but less than 6.0 volume percent). If this information is not available (from either the on-line monitor or sample analysis), the recombiners may not be started. Also, in the FR-C.1 and FR-Z.1 procedures, if the containment hydrogen concentration is greater than 6 volume percent, the engineering staff is to be consulted. If the hydrogen concentration is not available, a flammable hydrogen mixture could exist in the containment without the knowledge of the plant staff.

It can be argued that based on the interface between the ERGs and the SAMG, the ERG instructions should be discontinued (and the SAMG started) long before appreciable hydrogen is generated by zirconium water reactions of the fuel rod cladding. Using this argument, the potential for hydrogen concentrations greater than 6 volume percent while still in the ERG instructions is highly unlikely. However, the transition from ERGs to SAMG is only made if the core exit thermocouples are indicating core temperatures in excess of 1200°F and increasing, AND all of the ERG actions have been unsuccessful. It is possible that recovery from a core damage accident with significant hydrogen generation from zirconium-water reactions could occur in the ERGs without the transition to the SAMG being made. This would be the case when recovery actions are initiated prior to reaching the SAMG transition step in the ERGs, but recovery is not rapid enough allowing for significant hydrogen generation before the core is completely recovered.

In the case of the FR-I.3 instruction step, the decision regarding venting of the reactor vessel to relieve noncondensible gas voids in the reactor vessel head is directly tied to the ability to obtain a containment hydrogen reading. If the containment hydrogen concentration is greater than 3 volume percent, reactor vessel head venting is delayed while the containment hydrogen concentration is reduced below 3 volume percent. However, a similar step appears in the E-1 procedure, "Loss of Reactor or Secondary Coolant" at step 16 without the need for an indication of the containment hydrogen concentration. Thus, venting of the reactor coolant system may be delayed or prohibited from the FR-I.3 procedure, but may be performed irrespective of containment hydrogen concentration obtained from the E-1 procedure.

Thus, the inability to obtain an indication of the containment hydrogen concentration may delay or prohibit reactor coolant head venting and may result in flammable hydrogen concentrations in the containment without the knowledge of the plant staff. Therefore, it is considered important to have at least one means of determining the containment hydrogen concentration.

4.1.6 Containment Sump Activity Level

There is one place in the generic WOG ERGs where the control room staff is instructed to obtain information regarding radioactivity in the containment sump. The purpose of this instruction is to provide information regarding the possible radiation levels in plant areas following transfer of containment sump water to storage tanks in the plant (such water transfers woud be done to avoid overfill of the containment sump and the subsequent possible loss of some equipment and/or instrumentation located above the design basis containment sump water level. The specific ERG where such an instruction is given is:

- FR-Z.2, "Response to Containment Flooding", Step 2, Check containment sump activity level [Enter plant specific means] .

The inability to obtain such a sample would not result in an unsafe plant condition since a bounding estimate could be made based on assuming all of the core fission products are in the containment sump water. Lacking a firm estimate of containment sump radioactivity levels, a small quantity of containment sump water could be transferred to a radwaste tank and the gross activity could be measured using portable radiation monitoring equipment.

Alternately, if significant radioactivity exists in the containment sump water, the containment radiation monitor would be reading abnormally high. An estimate of the core damage could be made using the WOG Core Damage Assessment Guidance (Ref. 14). This core damage estimate would then provide information for estimating possible containment sump activity levels.

Either method would provide adequate information to enable a decision to be made concerning the most appropriate place to store any water transferred from the containment sump. Thus, the inability to obtain such a sample would not result in an unsafe plant condition.

4.1.7 Containment Sump pH

The generic WOG ERGs do not address recovery actions related to the containment sump pH following an accident. In the development of the ERGs, the identification of and recovery actions for containment sump pH was left to the plant specific EOPs. For example, as described above, one of the last generic ERG steps in recovery from a LOCA is to initiate evaluation of plant status (Procedure E-1, step 12). In 1993, a Westinghouse Nuclear Safety Advisory Letter (NSAL) (reference 31) was issued to advise utilities of the need to have a procedure step to check and adjust the containment sump pH following an accident to assure that chloride induced stress corrosion cracking or long term evolution of radioactive iodine from the containment sump water would not occur.

In design basis analyses, the sump pH would be buffered by a chemical additive to the containment spray system flow. In this case, the sump pH would be at a correct level for both stress corrosion cracking control and for maintaining radioactive iodine in the sump water. However, the NSAL points out that for some accident sequences (e.g., some small LOCAs) the containment spray system would not be automatically actuated due to reduced containment pressure from these accidents. In this case, the sump pH would have to be adjusted by some strategy for chemical addition, based on the actual sump pH.

If sampling of containment sump water and subsequent analysis for pH is not available, there are alternate methods to estimate the sump pH. By knowing the containment sump water level (from instrumentation) and the correlation between water level and volume (for example from SAMG Computational Aid 5, "Containment Water Level and Volume") the volume of water in the containment sump at any time can be known. Since the tanks that supply most sources of water to the containment sump also have level instrumentation, the volumetric sources of the containment sump water can be determined. By knowing the pH of the water sources, the resultant pH can be determined. However, it should be noted that a number of plants prohibit the use of any SAMG material while the EOPs are still in use due to concerns with the SAMG material relative to 10 CFR 50.59.

Thus, while the ability to sample the containment sump water and determine the sample pH may be an important aspect of recovery from an accident (particularly for salt water sites with only a single barrier between the cooling water and the inside of the containment), there are alternate methods for estimating containment sump pH. Therefore, the inability to measure the containment sump pH through sampling does not impact the ability to return the plant to a safe stable state following an accident.

4.2 SEVERE ACCIDENT MANAGEMENT GUIDANCE

In response to an industry-wide commitment to enhance accident management capabilities to cover accidents in which the core is severely damaged (reference 35), the Westinghouse Owners Group developed a set of generic symptom-based severe accident management guidance to serve as a basis for the development of plant specific Severe Accident Management Guidance (SAMG). The plant SAMG provide detailed guidance for the control room and Technical Support Center emergency response teams to diagnose the plant conditions and select the most

appropriate strategy to mitigate fission product releases and return the plant to a controlled stable state. The SAMG diagnostics, as well as the SAMG monitoring of changes in plant state following implementation of recovery strategies, are based on fixed in-plant instrumentation.

During the development of the SAMG it was determined that sampling could not provide timely information during the transient portions of a core damage accident. The SAMG development further concluded that sampling may only be of value after recovery is complete and the plant is in a controlled stable state. Since the SAMG is applicable only to the portion of a core damage accident from the time that the core begins to overheat and degrade until a controlled stable state is attained, the SAMG deals mostly with transient conditions. Thus, SAMG requirements for PASS, with the exception of hydrogen monitoring, do not exist.

However, the SAMG also identifies some long term concerns that need to be addressed after recovery from a core damage accident is well underway. Some of these long term concerns apply only after a controlled stable state is attained. They were included in the SAMG, not because they are important for recovery, but rather are important after recovery is complete in order to assure that a long term stable state is maintained.

4.2.1 Containment Hydrogen

One of the possible challenges to the integrity of the containment following a core damage accident is from the containment pressure increase associated with burning hydrogen that has accumulated in the containment. The severity of the challenge to the containment integrity is a function of the hydrogen and steam concentrations in the containment and the containment pressure. In the WOG SAMG, a computational aid (CA-3, "Hydrogen Flammability in Containment") is provided to assist in the diagnosis of potential challenges to containment integrity from a hydrogen burn. This computational aid requires that the containment hydrogen concentration be known, either from the on-line containment hydrogen monitor or from the analysis of containment gas samples for hydrogen. Since a core damage accident can result in rapid changes in the containment hydrogen concentration, the WOG SAMG relies on the on-line containment hydrogen monitor as the primary means of measuring containment hydrogen concentration. However, default values are also provided which represent bounding containment hydrogen conditions for cases in which the containment hydrogen conditions are not known. Thus, indication of the containment hydrogen conditions via the on-line hydrogen monitor is not a requirement of the WOG SAMG. Use of the default conditions until a sample of the containment atmosphere can be drawn and analyzed for hydrogen is an acceptable method.

The potential for a hydrogen burn is also included as a negative impact associated with the implementation of most of the other core damage recovery strategies (e.g., inject water into the RCS, depressurize containment, etc.) in the SAMG. Prior to recommending the implementation of one of these other strategies, the emergency response organization SAMG *evaluator* would consult the SAMG computational aid for containment hydrogen challenges (CA-3) and use either the actual or default containment hydrogen concentration.

Thus, the containment hydrogen concentration is required by the SAMG to respond to challenges to containment integrity and to recover the plant to a controlled stable state. The on-line hydrogen monitor is the preferred method of acquiring the required information.

4.2.2 Containment Sump pH

The only place that the results of samples of containment sump water are referred to in the SAMG is in the guidelines that contain strategies for assuring that an adequate water level exists in the containment. These guidelines are SAG-4, "Inject into Containment" and SAG-8, "Flood Containment". In both of these guidelines, the containment sump pH is identified as a long term concern that should be monitored following implementation of strategies to put water into the containment. The rationale given in these guidelines is that the containment sump pH is important for both: a) retention of radioactive iodine in the containment sump water which reduces the possible fission product leakage through the containment and containment penetrations, and b) preventing long term chloride induced stress corrosion cracking of stainless steel piping which prevents a worsening of the accident. However, the need to sample the containment sump to determine the pH of the sump water is not suggested until after the accident condition has been stabilized, and the plant has been returned to a controlled stable state.

As identified in the Westinghouse Nuclear Safety Advisory Letter (NSAL) (reference 31), the containment sump pH does not require adjustment in the first 48 hours of the accident to preclude the potential for stress corrosion cracking of stainless steel piping. In the absence of the capability to determine the containment sump pH from sampling, the water inventory method used for determining containment sump boron concentration, as explained in the Emergency Response Guideline section of this report, could be used. The pH of the various water sources that exist in the containment, in conjunction with the containment water level, could be used to estimate the containment sump pH. As noted in the in the NSAL, a containment sump pH between 7.0 and 9.5 is required for both prevention of chloride induced stress corrosion cracking of stainless steel piping, and for iodine retention in the containment sump water. Given this wide range, the accuracy of the estimation method would clearly be adequate.

4.2.3 Reactor Coolant pH

Although not specifically mentioned in the WOG SAMG, the capability to ascertain the RCS pH is an important long term concern when water, other than from the containment sump, that potentially contains chlorides is used to reflood the reactor vessel. The WOG SAMG provides guidance to inject water into the RCS from any source in order to arrest the progression of a core damage accident. As discussed previously, the WOG-based EOPs also provide the same guidance when ECCS recirculation capability is lost. Priorities are given to borated water, but the plant specific list of alternative water sources may contain some sources that, for salt-water or brackish water plants, potentially contain significant levels of chlorides, such as refilling the RWST from the plant fire protection system. If a source of water that potentially contains chlorides is used to refill the reactor vessel, then the pH of the reactor coolant becomes a long term concern to prevent chloride induced stress corrosion cracking of reactor coolant system

piping; the long term integrity of the reactor coolant system piping is required to maintain the core in a controlled stable state after recovery from a core damage accident.

Thus, it is important to identify the pH of the reactor coolant system water so that appropriate actions can be taken to adjust the coolant pH to prevent stress corrosion cracking of the RCS piping. In the absence of the capability to determine the RCS pH from sampling, a method similar to the water inventory method used for determining containment sump boron concentration, as explained in the Emergency Response Guideline section of this report, could be used. The pH of the various water sources that exist in the reactor coolant system, in conjunction with the RCS water inventory, could be used to estimate the RCS pH. As noted in the NSAL, a pH between 7.0 and 9.5 is required for prevention of chloride induced stress corrosion cracking of stainless steel piping. Given this wide range, the accuracy of the estimation method would clearly be adequate.

4.2.4 Reactor Coolant Boron

Although not specifically mentioned in the WOG SAMG, the capability to ascertain the RCS boron concentration is an important long term concern when water, other than the original RWST inventory, is used to reflood the reactor vessel or to flood the containment. The WOG SAMG provides guidance to inject water into the RCS and/or the containment from any source, in order to arrest the progression of a core damage accident. Again, the WOG-based EOPs also provide the same guidance when ECCS recirculation capability is lost. Priorities are given to borated water, but the plant specific list of alternative water sources may contain some sources that potentially contain boron concentrations insufficient to assure subcriticality in the core. An example of such a case is injection into the RCS after refilling the RWST from the plant fire protection system. In this case, the primary concern is core cooling. If boron level in the injected water is not sufficient to achieve subcriticality, the core will return to a low power level but in a cooled state. The combination of core temperature and moderator voiding will limit the power return to a low level where the heat generation can still be totally removed. Only at this point is criticality a concern and that can be directly monitored from the ex-core neutron detectors. The SAMG does not rely on RCS sampling for boron concentration, since it would not provide timely feedback.

In the long term after recovery from a core damage accident, there would be a long term concern related to the shutdown margin of the core. Sampling the RCS for boron concentration would provide a key piece of information for this assessment. However, other alternative means are available including: a) trending of the ex-core neutron detector output to assure a subcritical state and a negative or stable startup rate, and b) an inventory balance of borated water used to recover the core. Due to the time required to obtain and analyze an RCS sample for boron concentration, the RCS sample would only provide confirmation of the core status already derived from the alternative methods. Thus, RCS boron sampling for core damage accidents that result in usage of the SAMG is not required to achieve a controlled stable state.

4.3 SITE EMERGENCY PLAN

The Site Emergency Plan consists of three parts that may be impacted by the post accident sampling system: the Core Damage Assessment, the Offsite Dose Assessment and the Emergency Action Level (EAL) classification.

4.3.1 Core Damage Assessment

In 1999, the generic WOG methodology for assessing core damage, as required by NUREG-0737, was updated to be consistent with the most recent knowledge related to the progression of core damage accidents and the behavior of fission products that are produced during operation of the plant. This is documented in WCAP-14969-A, Revision 1, "Westinghouse Owners Group Core Damage Assessment Guidance" (reference 14). The previous generic core damage assessment methodology (1984 vintage in reference 13) required sampling of the reactor coolant, containment atmosphere and containment sump to make a quantitative assessment of core damage. Thus, some of the requirements for the post accident sampling system were tied to the ability to make a core damage assessment using the 1984 methodology. However, the latest core damage assessment guidance does not rely on the results of analysis of any samples of plant fluids. Thus, once the generic WOG methodology is implemented at each plant, the requirement for post accident sampling to support the core damage assessment capability no longer exists.

The 1999 WOG core damage assessment guidance relies on fixed in-plant instrumentation to make a quantitative estimate of the amount of core damage that has occurred. The guidance relies on the core exit thermocouple indications and containment high range area radiation monitor indications to provide a quantitative estimate of the amount of core damage. This estimate is then qualitatively confirmed by indications from the containment on-line hydrogen monitor, hot leg Resistance Temperature Detectors (RTDs), reactor vessel level, and ex-core source range instrumentation levels. Technical arguments are presented to support the conclusion that analysis of samples of containment atmosphere, containment sump and reactor coolant system fluids do not provide accurate or timely indications of the amount of core damage prior to the time that stable core conditions are recovered.

WCAP-14696 also presents information to show that the timeliness of obtaining results from the analysis of samples is very poor with respect to the purpose of quantitative estimation of the amount of core damage. For example, if a sample is requested when the core becomes overheated and damage is just beginning, the entire core can be melted by the time the sample results are available. Also, the report shows that the results of samples can be very misleading in terms of the amount of core damage during the time prior to regaining a stable core condition due to the behavior of fission products in the reactor coolant system and containment. Even if results of samples from all three post accident sampling system locations (reactor coolant, containment atmosphere and containment sump) could be obtained instantaneously, the significant deposition of fission products in other locations in the plant causes large uncertainties in any conclusions drawn from the sample results.

4.3.2 Offsite Dose Assessment

Part of the Site Emergency Plan includes the capability to make offsite dose assessments. The purpose of this capability is to enable responsible emergency response personnel to make recommendations regarding offsite radiological protective actions to protect the health and safety of the public.

While the overall requirement for offsite dose assessment capability is fixed, the manner in which it is carried out at each plant site varies. In particular, the radiological source term for the offsite dose assessment can come from a variety of sources including: design basis accident analyses, realistic analyses that are part of the plant Probabilistic Safety Analysis (PRA), offsite radiological monitoring indications, plant radiation monitoring indications and the results of analyses of samples of plant fluids. There is no standard methodology among the existing plants.

The offsite dose assessment capability is a very important part of the plant accident management capabilities, since it provides the basis for making recommendations regarding offsite radiological protective actions. In so far as sampling of plant fluids is concerned, there are three separate conditions for which the results of samples may impact the capability to make accurate offsite dose assessments: a) normal leakage from the containment following an accident, b) venting the containment following an accident, and c) intentional releases from the steam generators following an accident.

The source term for assessing the potential offsite doses following an accident in which the containment is intact, but above atmospheric pressure, is the fission product inventories that are airborne in the containment. This source term can be assessed in a variety of ways including: a) use of a pre-calculated value (e.g., design basis or PRA source term), b) the containment high range area radiation monitor, or c) the results of samples of the containment atmosphere.

The source term for assessing the potential offsite doses following an accident in which the intentional releases from either the containment or the steam generators are planned is the fission product inventories that are airborne in the containment or steam generators. Like the assessment of offsite doses from containment leakage, this source term can be assessed in a variety of ways including: a) use of a pre-calculated value (e.g., design basis or PRA source term), b) the containment high range area radiation monitor, or c) the results of samples of the containment atmosphere.

4.3.3 Emergency Action Level Classification

The Emergency Action Level (EAL) classification is a method developed after the TMI-2 accident to denote, in broad classifications, the condition of the plant following an accident and the potential for releases of radioactive fission products to the environment. The original scheme of classifying accidents is contained in NUREG-0654 and is based on the potential for breach of one or more of the barriers that prevent releases of fission products to the environment. A later scheme was developed under the auspices of the Nuclear Energy Institute (reference 7) which is based on an updated knowledge base related to the challenges to the

fission product barriers. For PWRs, many of the criteria for the classification of the appropriate emergency action level are based on the Critical Safety Function Status Trees in the WOG ERGs. Both of these classification schemes are examined below for their impact on the post accident sampling system, since either of the classification systems may be used.

From the perspective of the post accident sampling system, the NUREG-0654 scheme requires that radiological analysis of samples of reactor coolant system be available. NUREG-0654 uses the value equal to the plant Technical Specification limit for reactor coolant activity associated with iodine spiking as a criterion for declaring an "Unusual Event" classification. In addition, NUREG-0654 uses a value of 300 μCi/cc equivalent I-131 in the reactor coolant system as a criterion for declaring an "Alert" classification, unless other fission product barriers are lost, or potentially lost in which case it is classified at a higher level. All of the other criteria related to radioactive fission products are expressed as instrumentation indications. For example, failed fuel monitor indications are used to declare Unusual Event and Alert states, an increase of a factor of 1000 in direct radiation readings is a criterion for an Alert state, and <u>instrumentation</u> to detect inadequate core cooling, coolant activity and/or containment activity is used for Site Emergency status.

The NEI scheme also suggests a value of 300 μCi/cc equivalent I-131 in the reactor coolant system as an EAL criterion for the Alert classification, as suggested in NUREG-0654. All other NEI suggested EAL classification criteria related to radiation levels and radioactivity are expressed in terms of radiation levels to be measured by fixed in-plant instrumentation or portable off-site radiation surveys. Therefore, the capability to sample and analyze reactor coolant system radionuclides following an accident is required to support the Emergency Action Level classification that is part of the plant Site Emergency Plan.

An assessment was performed to identify possible accident sequences where the RCS fission product inventories can approach the trigger value for either the NEI or NRC based EAL classifications. This assessment could not identify any accident sequences where high RCS fission product activity would be the first indication of plant conditions requiring the declaration of an EAL classification. The range of accident sequences considered in the assessment ranged from design basis events that result in fuel rod clad damage (including postulated clad damage due to departure from nucleate boiling), to beyond-design basis events that result in fuel rod melting. In all cases, another of the EAL classification criteria for the appropriate EAL classification level would reach its trigger value before fuel rod damage occurs. Given the time delay involved in obtaining and analyzing an RCS sample for radioactive content, it can be concluded that the EAL criteria related to RCS fission product inventories is not required, since it is redundant to other EAL criteria.

5 EVALUATION OF PASS REQUIREMENTS

This section provides a technical assessment of the post accident sampling system requirements in light of current knowledge of core damage accidents, including their progression, severe accident phenomena, and the behavior of certain significant chemical species and fission products in a post accident plant environment. The intent of this section is to provide a technical justification for an updated post accident sampling system that provides the needed support for accident management and protection of the health and safety of the offsite public.

In this section, each of the current post accident sampling system capabilities is examined individually. Each assessment includes the NUREG-0737 and Regulatory Guide 1.97 requirements in terms of their intent, and whether they can be technically supported based on current understanding of core damage accidents. The ties to other accident management and emergency planning functions are also included in each section. The conclusion of each assessment provides recommendations for an updated post accident sampling system for Westinghouse Owners Group member plants. The recommendations for each post accident sampling system capability include, in some cases, alternative methods of obtaining information to that provided by post accident sampling system capabilities. In some cases, the post accident sampling system information could be replaced by equivalent information obtained or inferred from fixed in-plant instrumentation.

5.1 REACTOR COOLANT DISSOLVED GASES

The purpose of sampling the reactor coolant system for dissolved gases is to assure that, upon depressurization of the reactor coolant system, a void of noncondensible gases will not form in the reactor vessel head or the high point of the reactor coolant system and disrupt natural circulation cooling that might be used for long term decay heat removal after the accident.

The post accident sampling capability to measure the dissolved gas content of the reactor coolant is an explicit requirement of NUREG-0737 and Regulatory Guide 1.97. NUREG/CR-4330, Volume 3 suggested that this requirement could be deleted based on the installation of reactor vessel head vents and reactor vessel level instrumentation systems. The EPRI Utility Design Requirements document for ALWRs does not include a requirement for sampling and analysis of reactor coolant for dissolved gases. However, the NRC policy, as expressed in SECY-93-087, on the design of the advanced light water reactors requires the capability to take dissolved gas measurements 24 hours after an accident.

The capability to measure dissolved gases in the reactor coolant system supports the current WOG ERGs. In the E-1 procedure, if safety injection is in use, the control room staff is directed to determine if the reactor vessel head should be vented by consulting the engineering staff. The engineering staff could rely on the results of dissolved gas analysis of reactor coolant samples if they were available. However, given the substantial time delay involved in obtaining and analyzing a sample, it is anticipated that the technical support staff would use the alternate indications to make a decision regarding the use of the reactor vessel head vent.

Note that dissolved gas is an indication that voids could form if the RCS is subsequently depressurized. If voids do form in the reactor vessel head, the ERGs provide a "fail-safe" indication via Critical Safety Function Status Tree, F-06, "Inventory" and the associated Functional Restoration procedure FR-I.3, "Response to Voids in the Reactor Vessel", using the reactor vessel level indication. If the dissolved gas content of the RCS cannot be determined, the engineering staff has other indicators that could be used to make this determination, such as reactor coolant system pressure, break location, whether core overheating has previously occurred (as an indicator of hydrogen generation from zirc-water reactions), pressurizer pressure, and reactor vessel level indication.

Although post accident sampling provides useful information for the plant engineering staff to make a decision regarding the need to use the reactor vessel vent, there are other methods that can be used to reach this decision and, if voids do form in the reactor vessel upper head, an EOP procedure is in-place (i.e., FR-I.3, "Response to Voids in Reactor Vessel") for diagnosis and response to this condition.

Thus, post accident sampling and analysis of reactor coolant for noncondensible gases is not required to reach a safe, stable state following an accident. Based on the above assessments of the need for post accident sampling capabilities for reactor coolant dissolved gases, this post accident sampling function should be deleted for all plants.

5.2 REACTOR COOLANT HYDROGEN

The purpose of sampling the reactor coolant system for dissolved hydrogen is to assure that, upon depressurization of the reactor coolant system, a void of noncondensible gases will not form in the reactor vessel head or the high point of the steam generator tubes and disrupt natural circulation cooling that might be used for long term decay heat removal after the accident.

The post accident sampling capability to measure the dissolved hydrogen content of the reactor coolant is an alternative to the requirement for measuring total reactor coolant system dissolved gases in both NUREG-0737 and Regulatory Guide 1.97. Based on reference 36, the dissolved hydrogen reactor coolant concentration could be used as a screening for determining whether reactor coolant dissolved oxygen and/or chlorides need to be closely monitored (e.g., from reference 36, with dissolved hydrogen greater than 10 cc/kg of coolant, dissolved oxygen may be assumed to be less than 0.1 ppm which, in turn indicates that chloride induced stress corrosion cracking is not a concern). The EPRI Utility Design Requirements document for ALWRs does not include a requirement for sampling and analysis of reactor coolant for hydrogen. The NRC policy, as expressed in SECY-93-087, on the design of the advanced light water reactors does not require the capability to take reactor coolant hydrogen measurements.

- There are no accident management or emergency planning functions that require identification of the RCS hydrogen (see also the conclusion above regarding dissolved gases). Thus, post accident sampling and analysis of RCS hydrogen is not required to reach a safe, stable state following an accident.

As detailed in Section 5.1, "Reactor Coolant Dissolved Gases," the appropriate actions do not require sampling and analysis of reactor coolant system liquid.

Thus, it can be concluded that there is no need to maintain a post accident system capability to sample and analyze the reactor coolant system hydrogen. Based on the above assessments of the need for post accident sampling capabilities for reactor coolant hydrogen, this post accident sampling function should be deleted for all plants.

5.3 REACTOR COOLANT OXYGEN

The purpose of sampling the reactor coolant system for dissolved oxygen is to assess the potential for chloride induced stress corrosion cracking of the stainless steel RCS piping.

The post accident sampling capability to measure the dissolved oxygen content of the reactor coolant is recommended in NUREG-0737 and is a requirement in Regulatory Guide 1.97. The EPRI Utility Design Requirements document for ALWRs does not include a requirement for sampling and analysis of reactor coolant for oxygen. The NRC policy, as expressed in SECY-93-087, on the design of the advanced light water reactors does not require the capability to take reactor coolant oxygen measurements.

Regulatory Guide 1.97 states, in footnote 20: *"Within the first 30 days after an accident, oxygen analysis need not be performed until chloride analysis indicates a chloride concentration greater than 0.15 ppm. Once the chloride concentration exceeds this value, oxygen should be determined within 3 hours. For this 30 day period, it is acceptable to verify that dissolved oxygen is less than 0.1 ppm if the measured dissolved hydrogen residual is 10 kg/cc or less. However, consistent with minimizing personnel radiation exposures (ALARA), direct monitoring for dissolved oxygen is recommended. This applies only to primary coolant, not to the sump."*

There are no accident management or emergency planning functions that require identification of the reactor coolant oxygen content. Thus, post accident sampling and analysis of reactor coolant for oxygen is not required to reach a safe, stable state following an accident. The requirement for reactor coolant oxygen sampling and analysis is tied to preventing chloride induced stress corrosion cracking of stainless steel piping, which ensures that continued long term cooling of the core is not compromised. If reactor coolant chloride concentrations, in the range where stress corrosion cracking may be an issue, are indicated or suspected, then the appropriate actions would be to either:

- ensure that the reactor coolant oxygen concentration is at a level where stress corrosion cracking cannot occur, or
- adjust the pH of the reactor coolant to the point where stress corrosion cracking cannot occur.

As detailed in Section 5.4, "Reactor Coolant System Chlorides," the appropriate actions do not require sampling and analysis of reactor coolant system liquid. Thus, it can be concluded that there is no need to maintain a post accident system capability to sample and analyze the reactor

coolant system oxygen. Based on the above assessments of the need for post accident sampling capabilities for reactor coolant oxygen, this post accident sampling function should be deleted for all plants.

5.4 REACTOR COOLANT CHLORIDES

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

The post accident sampling capability to measure the reactor coolant chlorides is a NUREG-0737 and a Regulatory Guide 1.97 requirement. The EPRI Utility Design Requirements document for ALWRs does not include a requirement for sampling and analysis of reactor coolant for chlorides. The NRC policy, as expressed in SECY-93-087, on the design of the advanced light water reactors does not require the capability to take reactor coolant chloride measurements. The advanced LWR lack of a requirement for chloride measurement of reactor coolant system water is based on the design of the advanced plant that reduces the potential for chloride concentrations in contact with stainless steel piping. Thus, the ALWR basis for eliminating the requirement to monitor reactor coolant chlorides during and following an accident may not be applicable to the current generation of PWRs.

There are no generic accident management or emergency planning functions that utilize a sample of reactor coolant chlorides. The plant specific long term plant monitoring step in the EOPs described previously in Section 4.1.2 under the heading of "Check Plant Status" may specify monitoring the reactor coolant chloride concentrations.

Chlorides can be introduced into the reactor coolant system in three different ways: a) recirculation of water from the containment sump (design basis emergency core cooling recirculation), b) refilling the RWST with non-demineralized water (e.g., brackish river water) to continue injection of water into the reactor coolant system per the WOG ERG procedure ECA-1.1, "Loss of Emergency Coolant Recirculation" or WOG SAMG guideline SAG-3, "Inject into the RCS", or c) backfilling the RCS through a ruptured steam generator tube per the WOG ERG procedure ES-3.1, "Post SGTR Cooldown Using Backfill".

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 location. In terms of the time at which the first sample for chlorides must be taken, the NRC has recognized that fresh water plants and brackish water (or salt-water) plants with more than one barrier between the containment and the ultimate heat sink are much less likely to have high chloride concentrations in plant systems, compared to brackish water plants with only one barrier between the potential source of chlorides and the containment. In the first instance (fresh water plants and brackish water plants with more than one barrier between the containment and ultimate heat sink) the initial chloride sample is not required for 96 hours (4 days). In the latter case (brackish water plants with only one barrier), the first chloride samples are required in 24 hours.

The determination of the need for sampling and analysis must first consider the indications available to the plant operating staff in terms of suspecting that high chloride concentrations could exist in plant systems. In the case of the containment sump, high levels of chlorides would result from leakage from cooling water systems inside containment that contain high levels of chlorides. This is only applicable to salt water or brackish water sites with single barrier cooling systems inside containment; the potential for high chloride concentrations in the containment sump from all other plants due to leakage from cooling systems is considered sufficiently remote. Any significant leakage from cooling water systems into the containment would be indicated by an unexplained increase in the containment sump water level. Such increases would likely be quickly detected by the emergency response staff from the containment sump level indication. The other two cases (refilling the RWST with water containing high chloride levels and backfilling the RCS from a steam generator) are intentional actions to provide cooling to the core. The impact of using water containing high levels of chlorides would be identified prior to the initiation of these actions. Thus, in all cases, the suspected presence of chlorides in the reactor coolant system would be known very early in the event and appropriate contingency actions (such as pH adjustments) would be planned. As will be discussed further in the next paragraphs, these contingency actions are independent of the level of chlorides in the reactor coolant system and therefore sampling and analysis of reactor coolant for chlorides is not required to achieve a safe stable state.

In the case of high chloride concentrations in the containment sump water being transferred to the reactor coolant system via ECCS recirculation, if containment sprays have operated and the spray additive tank contents have been emptied, the pH of the recirculated water will eliminate the potential for chloride induced stress corrosion cracking, regardless of the chloride concentration in the sump water. For those plants with passive pH control in the containment sump and for ice condenser containment plants, containment sump pH control does not depend on operation of the containment spray. For all other plants, there are a number of accident sequences (e.g. ,small LOCA) where the containment pressure does not reach the setpoint value for automatic activation of the containment spray. In these sequences, unless manual actuation of the spray occurs, the sump pH will not be adjusted by the spray additive tank contents and the chloride concentration can become important.

For plants at fresh water sites, the potential concentration of chlorides that can be introduced in to the reactor coolant system is generally quite low. Additionally, if the pH of the water in the reactor coolant system can be estimated and adjusted (see section 5.5), there is no need to know the exact chloride concentrations in the water.

Based on the above assessments of the need for post accident sampling capabilities for reactor coolant chlorides, this post accident sampling function should be deleted for all plants.

5.5 REACTOR COOLANT PH

The purpose of sampling the reactor coolant system for pH is to assure that chloride induced stress corrosion cracking of stainless steel piping will not occur in the long term, and to assure that radioactive iodine is retained in the water. Sampling and analysis of reactor coolant for pH may be an alternative to sampling the reactor coolant for chlorides, since chloride induced stress

corrosion cracking is only an issue if the pH of the water is below 7.0. Another consideration in determining the reactor coolant pH is that it provides an indication of the pH of the containment sump water for retention of radioactive iodines for design basis accidents or accidents in which emergency core cooling is operational in the recirculation mode.

The post accident sampling capability to measure the reactor coolant pH is not a NUREG-0737 requirement. However, the requirement for pH sampling of reactor coolant is a Regulatory Guide 1.97 requirement. Also, the EPRI Utility Design Requirements document for ALWRs does not include a requirement for sampling and analysis of reactor coolant for pH. The NRC policy, as expressed in SECY-93-087, on the design of the advanced light water reactors does not require the capability to take reactor coolant pH measurements.

There are no accident management or emergency planning functions that inquire about the reactor coolant pH sample. However, as described previously, there is a requirement to assure that the pH of any water containing high concentrations of chlorides in contact with stainless steel piping is in the correct range in order to minimize the potential for chloride induced stress corrosion cracking of stainless steel piping. If emergency core cooling recirculation is being used to remove decay heat from the core in the reactor vessel, the measurement of reactor coolant pH could substitute for the measurement of sump pH.

In the case of a core damage accident in which the ECCS recirculation is not used for long term core cooling, the containment sump pH would not provide a meaningful indication of the potential for stress corrosion cracking of the reactor coolant piping due to chlorides in the reactor coolant. This case is possible, for example, when the RWST has been refilled to provide extended injection to the RCS for core cooling. Both the ERGs (ECA-1.1, Loss of Emergency Coolant Recirculation") and SAMG (SAG-3, "Inject into the RCS") provide guidance to refill the RWST with any water source that is available to re-establish or continue injection to the core when other methods of core cooling are not available. If the RWST is refilled with a water source containing high concentrations of chlorides (e.g., brackish river water), then the pH of the reactor coolant will need to be adjusted to prevent long term stress corrosion cracking of the RCS piping. This would require knowledge of the RCS pH. As discussed previously, if samples from the RCS are not available after recovery, the pH of the RCS can be estimated from the RCS water inventory and the pH of the various water sources that were used to inject into the RCS. It should be noted that the use of low chloride water sources (less than about 25 ppm chlorides) does not pose a major threat to long term stress corrosion cracking of stainless steel piping.

Based on the assessments of the need for post accident sampling capabilities for reactor coolant pH, this post accident sampling function should be deleted.

5.6 REACTOR COOLANT BORON

The purpose of sampling the reactor coolant system for boron is to assure that there is adequate shutdown margin 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 requirement. The EPRI Utility Design Requirements document for

ALWRs also includes a requirement for sampling and analysis of reactor coolant for boron. The NRC policy, as expressed in SECY-93-087, on the design of the advanced light water reactors is in agreement with the NUREG-0737 and EPRI positions on post accident sampling of reactor coolant boron.

In addition, the capability to measure boron in the reactor coolant system supports the WOG ERGs. As discussed previously in Section 3, there are at least ten different procedures in the ERGs where the control room emergency response staff is directed to verify adequate shutdown margin exists, but without explicit reference to obtaining RCS boron samples. In all of the procedures except one (ES-0.2 described below), the inability to verify adequate shutdown margin would result in the procedure step being skipped, and the emergency response continuing with the next procedure steps. In practice, the length of time required to obtain and analyze a reactor coolant sample for boron concentration has resulted in the use of alternate methods (plots, nomographs, etc) for assessing shutdown margin while the EOPs are in use. The ERGs provide a "fail-safe" for this condition (proceeding with subsequent EOP steps if boron sample analysis is not available) in F0.1, Subcriticality. In this Critical Safety Function Status Tree, an intermediate range startup rate greater than -0.2 decades per minute or a positive source range startup rate would trigger the use of FR-S.2, Response to Loss of Core Shutdown. Thus, the ability to achieve a safe, stable plant state would not be compromised by the inability to obtain a reactor coolant boron sample. The exception to this, as mentioned above, is in the procedure ES-0.2, Natural Circulation Cooldown. In the case where the reactor coolant radiation levels are too high to use the normal sampling system, the inability to obtain a reactor coolant boron sample to verify adequate shutdown margin may result in stopping further recovery actions until adequate shutdown margin can be verified via sampling.

Based on the above assessments of the need for post accident sampling capabilities for reactor coolant boron, this post accident sampling function should be deleted.

5.7 REACTOR COOLANT CONDUCTIVITY

There is no clear documentation regarding the purpose of sampling the reactor coolant system for conductivity .

The post accident sampling capability to measure the conductivity of the reactor coolant is not specified in NUREG-0737, Regulatory Guide 1.97, nor the EPRI Utility Design Requirements document for advanced LWRs. The NRC policy, as expressed in SECY-93-087, on the design of the advanced light water reactors does not require the capability to take reactor coolant conductivity measurements.

There are no accident management or emergency planning functions that utilize the reactor coolant conductivity sample. Thus, post accident sampling and analysis of reactor coolant for conductivity is not required to reach a safe, stable state following an accident and should be deleted.

5.8 REACTOR COOLANT RADIONUCLIDES

The purpose of sampling the reactor coolant system for radionuclide content is to assure that the integrity of the fuel rod cladding is not breached during an accident.

The post accident sampling capability to measure the reactor coolant radionuclide content is a NUREG-0737 and Regulatory Guide 1.97 requirement. The EPRI Utility Design Requirements document for ALWRs also includes a requirement for sampling and analysis of reactor coolant for both gross activity and gamma spectrum (radionuclide content). The NRC policy, as expressed in SECY-93-087, on the design of the advanced light water reactors is in agreement with the NUREG-0737 and EPRI positions on post accident sampling of reactor coolant radioactivity.

The capability to measure radionuclides in the reactor coolant system supports the Emergency Action Level classification in the Site Emergency Plan, regardless of whether the NUREG-0654 or the NUMARC/NESP-007 classification scheme is used. As discussed previously in Section 4.3.3, one of the criterion for declaring an emergency is the identification of greater than 300 µCi/cc of equivalent I-131 in the reactor coolant system. This level of activity has been found to be indicative of fuel rod cladding failures in 5 to 10% of the core, and represents a loss of the fuel rod fission product barrier. The loss of a fission product barrier was determined to warrant the declaration of an Alert condition. The other criteria for escalating an accident condition to an Alert or higher condition, from NUMARC/NESP-007 include: high core exit thermocouple indication, low reactor vessel water level indication, high containment radiation level indication, loss of RCS subcooling, a safety injection signal, or indication of a failure to achieve subcriticality following reactor trip. Considering the accidents that could result in core damage, as discussed previously, these alternate indications (alternate to high coolant activity as diagnosed from RCS sampling for radioactivity) would always result in a classification of an Alert or higher Emergency Action Level if the fuel rod cladding were failed. Additionally, considering the time required to obtain and analyze a sample of RCS fluid for radioactivity, the alternate indications would always result in a more rapid declaration of an Alert or higher condition.

Based on the above assessments of the need for post accident sampling capabilities for reactor coolant radionuclide content, this post accident sampling function is not necessary. Based on the above assessments of the need for post accident sampling capabilities for reactor coolant radionuclides, this post accident sampling function should be deleted for all plants.

5.9 CONTAINMENT ATMOSPHERE HYDROGEN

The purpose of sampling the containment atmosphere for hydrogen concentration is to assure that the integrity of the containment is not threatened by the combustion of an accumulated mixture of hydrogen in the containment.

Need for Containment Hydrogen PASS Capability

The post accident sampling capability (PASS) to measure the containment hydrogen concentration is a NUREG-0737 and Regulatory Guide 1.97 requirement. The NUREG/CR-4330, Volume 3 assessment of post accident sampling capabilities concluded that the capability to measure containment hydrogen by sampling was not required, since redundant, safety grade on-line hydrogen monitors (which are part of the Post Accident Monitoring System or PAMS) are now installed in PWR containments. In addition, the EPRI Utility Design Requirements document for ALWRs does not include a requirement for sampling and analysis of containment hydrogen. The NRC policy, as expressed in SECY-93-087, on the design of the advanced light water reactors is in agreement with the NUREG/CR-4330, Volume 3 and EPRI positions on post accident sampling of containment hydrogen.

The on-line containment hydrogen monitor (PAMS) is also a NURGE-0737 and Regulatory Guide 1.97 requirements. The combination of the PASS and PAMS capabilities provide a redundant means to measure containment hydrogen following an accident. The following discussion applies to both of these means of measuring containment hydrogen.

The capability to measure hydrogen concentration in the containment supports several accident management functions. The ERGs require knowledge of the containment hydrogen concentration for decisions regarding the use of the hydrogen recombiners (FR-Z.1, E-1, ES-1.2, ECA-1.1, ECA-3.1 and ECA-3.2) and the reactor coolant head vent (FR-I.3). The ERG requirements are applicable for accident scenarios in which core uncovery is quickly mitigated and a transition is not made to the SAMG. For these types of accident scenarios, there are no sustained core exit thermocouple temperatures in excess of 1200°F, indicating that there has not been significant core overheating. Hydrogen generation is principally from radiolysis and corrosion. Typically, the accumulation of hydrogen in the containment would be limited to nonflammable concentrations for at least the first day of the accident. The SAMG, which are used if core damage cannot be arrested quickly, requires knowledge of the containment hydrogen concentration to protect the containment integrity from a potential hydrogen burn challenge and to take actions to place the plant in a controlled stable state. While the SAMG presents a method for bounding the containment hydrogen concentration in the event that indication from the on-line hydrogen monitor is not available (i.e., by assuming a predetermined bounding amount of hydrogen generation), the SAMG default method is not a long term substitute for measuring the actual containment hydrogen concentration.

It should also be noted that the generic WOG SAMG does not recognize containment sampling as an alternative to the containment on-line monitor indication. The only alternative presented in the generic SAMG is the bounding estimate of hydrogen generation. While sampling containment atmosphere is an alternative for determining the containment hydrogen

concentration for SAMG assessments, it was determined that the time lag between requesting the sample and determining the actual concentration was too long in light of potential transients in containment hydrogen concentration during the event. Therefore, containment hydrogen concentrations obtained by sampling the containment atmosphere are not included in the SAMG list of alternatives for determining containment flammability.

In addition to the SAMG, the Core Damage Assessment, which is one of the Emergency Preparedness tools used in the Technical Support Center, uses the containment hydrogen concentration as a means to validate the core damage estimates obtained from correlations using the core exit thermocouples and the containment high area radiation monitor. As used in the revised WOG Core Damage Assessment Guidance (reference 14), the containment hydrogen concentration is not required unless significant fuel overheating has occurred as diagnosed from indications of several core exit thermocouples indicating off-scale high.

Based on the above assessments and requirements, the post accident sampling capability for containment hydrogen (PASS) should be deleted for all plants. The containment on-line hydrogen monitor (PAMS) provides the necessary indication of containment hydrogen for both the ERGs and the SAMG.

5.10 CONTAINMENT ATMOSPHERE OXYGEN

The purpose of sampling the containment atmosphere for oxygen concentration is to assure that the integrity of the containment is not threatened by the combustion of an accumulated mixture of hydrogen in the containment.

The post accident sampling capability to measure the containment oxygen concentration is not a NUREG-0737 nor a Regulatory Guide 1.97 requirement for PWRs. In addition, the EPRI Utility Design Requirements document for ALWRs does not include a requirement for sampling and analysis of containment oxygen. The NRC policy, as expressed in SECY-93-087, on the design of the advanced light water reactors is in agreement with the NUREG-0737 and EPRI positions on post accident sampling of containment oxygen.

The capability to measure oxygen concentration in the containment does not support any accident management or emergency planning functions.

Based on the above assessments of the need for post accident sampling capabilities for containment oxygen, this post accident sampling function is not necessary and should be eliminated for all plants.

5.11 CONTAINMENT AIRBORNE RADIOACTIVE SAMPLES

The purpose of sampling the containment for radionuclide content is to enable offsite dose assessments to be made from both post accident containment leakage, as well as the potential for a sudden release of the containment inventory of radionuclides.

The post accident sampling capability to measure the containment radionuclide inventory is a NUREG-0737 and a Regulatory Guide 1.97 requirement. The EPRI Utility Design Requirements document for ALWRs also includes a requirement for sampling and analysis of containment gross activity and radionuclide content. The NRC policy, as expressed in SECY-93-087, on the design of the advanced light water reactors is in agreement with the NUREG-0737, Regulatory Guide 1.97 and EPRI positions on post accident sampling of containment radionuclides.

As discussed in Section 4.3.2 of this report, the capability to measure the radionuclide content of the containment atmosphere supports some of the offsite dose assessment procedures and the 1984 WOG Core Damage Assessment Methodology in the Site Emergency Plan. However, an assessment of the timeliness and accuracy of the samples reveals that the intent of NUREG-0737 cannot be met through the use of the results of samples of the containment atmosphere. In addition, current knowledge of core damage accidents indicates that, except for noble gases, the radionuclides in a sample of containment atmosphere are not indicative of core damage or of potential releases in the event that the containment fission product boundary is breached as a result of the accident, or if intentional releases from the containment are contemplated.

When NUREG-0737 was originally conceived in the early 1980's, 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 the containment atmosphere would not be timely. Also, considering the behavior of fission products, as discussed in Section 2 of this report, it is apparent that the sample results are not very accurate. 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 underestimated. 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.

In the development of the 1999 WOG Core Damage Assessment Guidance, a correlation was developed to assess the degree of core damage from the containment radiation monitor indication, in conjunction with the core exit thermocouple indications and the containment hydrogen concentration. Results of analysis of containment samples for radionuclide content was determined to be an unreliable indicator of core damage.

In the case of the Offsite Dose Assessment, sampling the containment atmosphere to obtain a source term for offsite dose calculations is not a reliable means of predicting offsite doses. For containment leakage, the use of the samples would likely over-predict the actual releases due to deposition of aerosol fission products in the release pathway from the containment to the

atmosphere. In the case of containment failure or containment venting, the use of containment atmosphere samples would likely under-predict the actual releases due to re-evolution of aerosol fission products from surfaces within the containment, as well as transport of fission products in the reactor coolant system, as the containment pressure is reduced. 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. Thus, the offsite dose assessment should be based on a method of predicting the containment fission product source term that relies on a correlation to the containment radiation monitor, rather than containment gas samples.

After recovery from a core damage accident is completed, per the EOPs or the SAMG, there may be a need to accurately determine the airborne containment fission products so that post-accident recovery actions can be planned. 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.

Based on the above assessments of the need for post accident sampling capabilities for containment radionuclides, this post accident sampling function is not necessary, and should be deleted for all plants.

5.12 CONTAINMENT SUMP RADIONUCLIDES

The purpose of sampling the containment sump for radionuclide content is to enable offsite dose predictions from emergency core coolant system recirculation leakage to be made.

The post accident sampling capability to measure the containment radionuclide inventory is not a NUREG-0737 requirement. However, Regulatory Guide 1.97 requires containment sump sampling for radionuclides. The EPRI Utility Design Requirements document for ALWRs also does not include a requirement for sampling and analysis of containment sump radioactivity. The NRC policy, as expressed in SECY-93-087, on the design of the advanced light water reactors is in agreement with the NUREG-0737 and EPRI positions on post accident sampling of containment sump radionuclides.

As discussed in Section 4.3.2 of this report, the capability to measure the radionuclide content of the containment atmosphere supports some of the offsite dose assessment procedures and the 1984 WOG Core Damage Assessment Methodology in the Site Emergency Plan. However, an assessment of the timeliness and accuracy of the samples reveals that the intent of NUREG-0737 cannot be met through the use of the results of samples of the containment sump.

Based on the above assessments of the need for post accident sampling capabilities for containment sump radionuclides, this post accident sampling function is not necessary and should be deleted for all plants.

5.13 CONTAINMENT SUMP PH

The purpose of sampling the containment sump for pH is to assure that the sump pH is within the allowable range to maximize radioiodine retention in the sump water and to minimize the potential for chloride induced stress corrosion cracking of stainless steel piping.

The post accident sampling capability to measure the containment sump pH is not a NUREG-0737 requirement. However, Regulatory Guide 1.97 requires sampling and analysis of containment sump pH. The EPRI Utility Design Requirements document for ALWRs also does not include a requirement for sampling and analysis of containment sump pH. The NRC policy, as expressed in SECY-93-087, on the design of the advanced light water reactors is in agreement with the NUREG-0737 and EPRI positions on post accident sampling of containment sump pH.

As discussed in Section 4.1.7 and 4.2.2 of this report, the capability to measure the pH of the containment sump water supports the plant specific Emergency Operating Procedures and SAMG, respectively. In both cases, TSC guidance is available for the evaluation of containment sump pH for all of the credible accident scenarios covered by both the ERGs and the SAMG, including the information in the NSAL (reference 31) described previously.

Following an accident, the pH of the containment sump water is dependent on a large number of factors, including: the amount of reactor coolant and accumulator water accumulated in the containment sump, the operation of the containment spray system (i.e., the spray additive tank injection), the amount of RWST water accumulated in the sump, whether any additional water has been injected into either the RCS (e.g., RWST refill) or the containment (e.g., Severe Accident Management Guideline SAG-4, Inject into Containment), and for ice condenser plants, the amount of ice that has melted (where the ice melt is accumulated in the containment sump).

For plants with passive containment sump pH control, the containment sump pH will be within the acceptable range for iodine retention and for chloride induced stress corrosion cracking, unless additional water (e.g., water addition in SAMG SAG-4 from the demineralized water storage tank) has been added to the containment sump. For plants with active containment sump pH control (typically via the containment spray additive tank), the containment sump pH will be within the acceptable range for iodine retention and for chloride induced stress corrosion cracking if the pH control is activated and no additional water (e.g., water addition in SAMG SAG-4 from the demineralized water storage tank) has been added to the containment sump. For the case where active containment sump pH control is not automatically actuated (e.g., automatic actuation of containment spray for small LOCA events), guidance is available for the plant engineering staff (see Section A.1.7 of Appendix A) to determine the need for pH adjustment via other means (e.g., manual actuation of containment spray). For the case of water addition to the containment sump, the plant engineering staff guidance described in Section A.1.7 of Appendix A recommends that the sump pH can be approximated from calculations of the containment sump level indication and the sources of water in the containment sump and the chemical composition of the water.

Based on the above assessments of the need for post accident sampling capabilities for containment sump pH, this post accident sampling function should be deleted.

5.14 CONTAINMENT SUMP CHLORIDES

The purpose of sampling the containment sump for chlorides is to assure that the sump pH is within the allowable range to minimize the potential for chloride induced stress corrosion cracking of stainless steel piping.

The post accident sampling capability to measure the containment sump chlorides is not a NUREG-0737 requirement. The EPRI Utility Design Requirements document for ALWRs also does not include a requirement for sampling and analysis of containment sump chlorides. The NRC policy, as expressed in SECY-93-087, on the design of the advanced light water reactors is in agreement with the NUREG-0737 and EPRI positions on post accident sampling of containment sump chlorides. However, the evolutionary and advanced plant designs are sufficiently different from some of the current generation of plants in terms of water source and chemical buffers that the requirements for advanced plants are not generically applicable to all of the current generation of plants. For plants on fresh water sites and for plants with passive pH control in the containment sump, the advanced/evolutionary plant requirements would be applicable

Similar to the discussion in Section 5.13 "Containment Sump pH" of this report, the capability to measure the chloride content of the containment sump water only indirectly supports the plant specific Emergency Operating Procedures and SAMG. The assessment of the need for sampling presented in Section 5.13 is equally applicable to containment sump chlorides.

Based on the above assessments of the need for post accident sampling capabilities for containment sump chlorides, this post accident sampling function should be deleted.

5.15 CONTAINMENT SUMP BORON

The purpose of sampling the containment sump for boron concentration is to assure that the core will remain subcritical if containment sump water is used for long term cooling of a damaged core that remains within the reactor vessel.

The post accident sampling capability to measure the containment sump boron is not a NUREG-0737 requirement. However, Regulatory Guide 1.97 requires sampling and analysis of containment sump boron. The EPRI Utility Design Requirements document for ALWRs also does not include a requirement for sampling and analysis of containment sump boron. The NRC policy, as expressed in SECY-93-087, on the design of the advanced light water reactors is in agreement with the NUREG-0737 and EPRI positions on post accident sampling of containment sump boron.

As discussed in Section 4, there is no EOP or SAMG basis for requiring measurement of containment sump boron concentration. Following an accident, the boron in the sump water is primarily dependent on the sources of water used to inject into the reactor coolant system and/or the containment. For all plants, the design basis water that can accumulate comes from the reactor coolant system, the RWST and the accumulators. The RWST and accumulators have sufficient boron to assure that the water in the containment sump will have the proper boron

concentration to prevent recriticality if the sump water is used for emergency core cooling recirculation. For ice condenser plants, the ice contains a boron additive that assures that the containment sump remains at the proper boron concentration considering the accumulation of water in the containment sump from the melting ice.

Thus, the only scenario where the containment sump boron concentration could be at a level where recriticality may be a concern when the water is used for emergency core cooling recirculation, is when unborated water is added to the containment. As discussed previously, these scenarios involve either: a) the intentional injection of unborated water to the reactor coolant system or containment, or b) significant leakage of water into the containment from cooling systems inside containment. In either case, monitoring the containment sump water level in combination with knowledge of the water sources that are accumulated in the containment sump can provide an acceptable method to estimate the containment sump boron concentration.

Based on the above assessments of the need for post accident sampling capabilities for containment sump boron, this post accident sampling function can be eliminated.

6 SUMMARY AND CONCLUSIONS

A summary of the PASS recommendations, based on current knowledge of core damage accidents described previously in this report is presented in Table 7.

Table 7
Summary of Post Accident Sampling Requirements and Recommendations

Sample Point/ Analysis	Requirement			Recommendation	Comments
	Regulatory	Accident Management	Emergency Planning		
RCS:					
Dissolved Gases	0737/1.97	ERG	N/A	Delete PASS Requirement	
Hydrogen	0737/1.97	N/A	N/A	Delete PASS Requirement	
Oxygen	1.97	N/A	N/A	Delete PASS Requirement	
pH	1.97	N/A	N/A	Delete PASS Requirement	
Chlorides	0737/1.97	N/A	N/A	Delete PASS Requirement	
Boron	0737/1.97	ERG/SAMG	N/A	Delete PASS Requirement	
Conductivity	N/A	N/A	N/A	Delete PASS Requirement	
Radionuclides	0737/1.97	N/A	EAL/ODCM	Delete PASS Requirement	Requires deleting Site Emergency Plan EAL criteria of 300 μCi/ml for Unusual Event condition
Containment Atmosphere					
Hydrogen	0737/1.97	ERG / SAMG	EAL	Delete PASS Requirement	On-line H_2 monitor (PAMS) provides adequate capability
Oxygen	N/A for PWRs	N/A	N/A	Delete PASS Requirement	
Radionuclides	0737/1.97	SAMG	ODCM	Delete PASS Requirement	
Containment Sump					
pH	*1.97*	*EOP/SAMG*	*N/A*	Delete PASS Requirement	
Chlorides	1.97	N/A	N/A	Delete PASS Requirement	May require change to plant specific EOPs if core damage has occurred
Boron	1.97	N/A	N/A	Delete PASS Requirement	
Radionuclides	1.97	ERG	ODCM	Delete PASS Requirement	

A brief discussion of the recommended change in PASS regulatory requirements contained in Table 7 is provided for clarification:

Delete PASS Requirement means that the capability to obtain a sample and analyze the sample for the specific radiation or chemical component under core damage conditions should be deleted from the plant features and emergency planning processes and plant technical specifications. While the requirement to obtain a sample using dedicated plant design basis equipment and systems should be deleted, plant features should be available to enable a sample of reactor coolant liquid, containment sump liquid and containment atmosphere to be safely obtained under core damage conditions, if and when it is requested. There should be no criteria for the minimum time after an accident when such a sample must be available, no criteria related to the accuracy of the sample analysis and no criteria related to where the analysis would be performed. There are also no criteria to demonstrate the capability to obtain a sample under core damage conditions. In practice, some pre-planning would be required to assure that the sample can be safely obtained according to applicable radiological protection standards. There are no criteria for pre-planning or demonstrating the method or equipment to be used.

Since there are a number of different methods by which WOG utilities originally committed to the implementation of the NUREG-0737 PASS requirements, it is expected that WOG utilities would implement these changes in accordance with established utility change processes for those methods.

As discussed previously in this report, there are a number of areas where the present emergency response model for a utility may need to be reassessed to consider the deletion of PASS requirements. For those plants using the Westinghouse Owners Group generic models for Core Damage Assessment Guidelines (CDAG), Emergency Response Guidelines (ERGs) and Severe Accident Management Guidelines (SAMG), these are would include:

- Update the Core Damage Assessment methodology to assure that sampling of plant fluids is not required to complete the core damage assessment. The recently approved model in WCAP-14694-A is one such methodology.

- Review plant specific EOPs to assure that there are alternate means (other than sampling) to obtain information to make assessments for accident sequences that may involve minimal amounts of core damage (e.g., recovery of core cooling in FR-C.1, "Response to Inadequate Core Cooling" without transition to SAMG). A review of the generic WOG ERGs for Revision 1B and Revision 1C is provided in Appendix A.

- Review plant specific SAMG to assure that post accident sampling is not referenced as a means to obtain information to make assessments for accident sequences that may involve core damage. A review of the generic SAMG has been completed as part of the SAMG Addendum program in early 2000.

- Update the plant specific EAL classification methodology to delete the 300 microcurie per gram of reactor coolant activity as a trigger for an Alert and assessment of capability to declare an Alert level based on other plant conditions for an equivalent event.

- Review the Plant Emergency Plan and Emergency Plan Implementing Procedures to assure that there are alternate means (other than sampling) to obtain information to make assessments for accident sequences that may involve core damage.

- Identification of a conceptual method to obtain samples of reactor coolant liquid, containment sump liquid and containment atmosphere following a core damage accident. This does not include demonstration that the sample can be obtained.

The technical justification for deletion of PASS, as presented in this report and as approved by the NRC in their review of this report, in conjunction with the plant specific emergency response review outlined above ensures that the effectiveness of the plant emergency response is not decreased as a result of PASS elimination. Therefore, prior NRC approval is not required in accordance with 10CFR50.54(q) provided the conditions established in this report, as approved by the NRC, are followed.

7 REFERENCES

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APPENDIX A

Westinghouse Owners Group

Comparison of ERG Rev. 1B and Rev. 1C

A.1 EMERGENCY OPERATING PROCEDURES

Within the EOP framework, there are several types of sampling that are recommended for diagnosis of plant conditions that are used to place the plant in a safe, stable condition. This section compares the recommendations in Rev. 1B (reference 11) and Rev. 1C (reference 39) of the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs).

A.1.1 RCS Boron

There are several places in the generic WOG ERGs where the control room staff is instructed to obtain information regarding boron concentrations in the reactor coolant system. The purpose of these instructions is to assure that there is adequate shutdown margin. The specific ERGs where such instructions are given include:

WOG ERG Rev. 1B vs. Rev. 1C Comparison RCS Boron Sampling			
ERG Guideline	**Rev. 1B Step**	**Rev. 1C Step**	**Assessment of Impact of Differences**
ES-0.2	Step 3	Step 3	No Impact
ES-1.2	Note prior to Step 7	Note prior to step 7	No Impact
	Step 21	step 21	No Impact
	Step 21 (a)	step 21 (a)	No Impact
ES-3.1	Step 3	Step 3	No Impact
	Step 3 (a)	step 3 (a)	No Impact
	Step 3 (b)	step 3 (b)	No Impact
ES-3.2	Step 3	Step 3	No Impact
	Step 3 (a)	Step 3 (a)	No Impact
	Step 3 (b)	Step 3 (b)	No Impact
ES-3.3	Step 3	Step 3	No Impact
	Step 3 (a)	Step 3 (a)	No Impact
	Step 3 (b)	Step 3 (b)	No Impact
ECA-0.1	Step 17	Step 15	Different Step Number; No Impact
	Step 17 (a)	Step 15 (a)	Different Step Number; No Impact
ECA-3.1	Step 25	Step 26	Different Step Number; No Impact
	Step 25 (a)	Step 26 (a)	Different Step Number; No Impact
	Step 25 (b)	Step 26 (b)	Different Step Number; No Impact
ECA-3.2	Step 19	Step 20	Different Step Number; No Impact
	Step 19 (a)	Step 20 (a)	Different Step Number; No Impact
	Step 19 (b)	Step 20 (b)	Different Step Number; No Impact
ECA-3.3	Step 23	Step 23	No Impact
	Step 23 (a)	Step 23 (a)	No Impact
	Step 23 (b)	Step 23 (b)	No Impact

The comparison of the ERG Rev. 1B and Rev. 1C shows that there are no differences in the two versions of the guidelines related to RCS boron samples. In a few instances the step numbers are different due to changes in other steps within the guideline. However, there are no differences in the steps themselves or their placement in the guideline relative to other guideline steps.

A.1.2 Plant Status

There are several places in the generic WOG ERGs where the control room staff is instructed to obtain information regarding the plant status from undefined samples of plant fluids. The purpose of these instructions is to provide additional information to the plant emergency response staff regarding the overall plant conditions prior to exiting the ERGs and going into a less structured long term recovery mode of emergency response. The specific ERGs where such instructions are given include:

WOG ERG Rev. 1B vs. Rev. 1C Comparison
Plant Status

ERG Guideline	Rev. 1B Step	Rev. 1C Step	Assessment of Impact of Differences
E-1	Step 12 (c)	Step 11 (c)	Different Step Number; No Impact
	Step 19	Step 20	Different Step Number; No Impact
ES-1.2	Step 32	Step 33	Different Step Number; No Impact
ES-3.1	Step 12	Step 12	No Impact
ES-3.2	Step 16	Step 16	No Impact
ECA-2.1	Step 43	Step 46	Different Step Number; No Impact
ECA-3.1	Step 38	Step 40	Different Step Number; No Impact
ECA-3.2	Step 32	Step 34	Different Step Number; No Impact
ECA-3.3	Step 37	Step 37	No Impact

The comparison of the ERG Rev. 1B and Rev. 1C shows that there are no differences in the two versions of the guidelines related to plant status. In a few instances the step numbers are different due to changes in other steps within the guideline. However, there are no differences in the steps themselves or their placement in the guideline relative to other guideline steps.

A.1.3 RCS Gases

There is one place in the generic WOG ERGs where the control room staff is instructed to obtain information regarding the potential noncondensible gas content of the reactor vessel head. The purpose of this instruction is to provide information regarding the use of the reactor vessel head vent to prevent a disruption of core cooling flow when safety injection is still in use. The specific ERG where such instruction is given:

WOG ERG Rev. 1B vs. Rev. 1C Comparison RCS Gases			
ERG Guideline	**Rev. 1B Step**	**Rev. 1C Step**	**Assessment of Impact of Differences**
E-1	Step 16	Step 16	No Impact
F-0.6/FR-I.3	General	General	No Impact

The comparison of the ERG Rev. 1B and Rev. 1C shows that there are no differences in the two versions of the guidelines. In addition, as discussed in the report, RCS gases are not sampled to provide input to FR-I.3 in Rev. 1B. There is no change in this diagnosis method in Rev. 1C.

A.1.4 Steam Generator Tube Integrity

There are several places in the generic WOG ERGs where the control room staff is instructed to obtain information regarding radioactivity on the secondary side of the steam generators. The purpose of these instructions is to provide information regarding the integrity of the steam generator tubes. Note that this section is included for completeness; there is no regulatory requirement for post accident sampling of the steam generator secondary side. The specific ERGs where such instructions are given include:

WOG ERG Rev. 1B vs. Rev. 1C Comparison Steam generator Tube Integrity			
ERG Guideline	**Rev. 1B Step**	**Rev. 1C Step**	**Assessment of Impact of Differences**
E-2	Step 6	Step 6	No Impact
	Step 6 (a)	Step 6 (a)	No Impact
E-3	Step 2	Step 2	No Impact
FR-H.3	Step 7	Step 8	Different Step Number; No Impact

The comparison of the ERG Rev. 1B and Rev. 1C shows that there are no differences in the two versions of the guidelines related to steam generator integrity. In a few instances the step numbers are different due to changes in other steps within the guideline. However, there are no differences in the steps themselves or their placement in the guideline relative to other guideline steps.

A.1.5 Containment Hydrogen Concentration

There are several places in the generic WOG ERGs where the control room staff is instructed to obtain information regarding hydrogen concentration in the containment. The purpose of these instructions is to provide information regarding the possible approach to flammable conditions in the containment so that actions can be initiated to preclude the establishment of flammable conditions in the containment. The specific ERGs where such instructions are given include:

WOG ERG Rev. 1B vs. Rev. 1C Comparison Containment Hydrogen			
ERG Guideline	**Rev. 1B Step**	**Rev. 1C Step**	**Assessment of Impact of Differences**
E-1	17 (a)	17 (a)	No Impact
	17 (b) RNO	17 (b) RNO	No Impact
	17 (c) RNO	17 (c) RNO	No Impact
ES-1.2	31 (a)	31 (a)	No Impact
	31 (b) RNO	31 (b) RNO	No Impact
	31 (c) RNO	31 (c) RNO	No Impact
ECA-1.1	37 (a)	37 (a)	No Impact
	37 (b) RNO	37 (b) RNO	No Impact
	37 (c) RNO	37 (c) RNO	No Impact
ECA-3.1	38 (a)	38 (a)	No Impact
	38 (b) RNO	38 (b) RNO	No Impact
	38 (c) RNO	38 (c) RNO	No Impact
ECA-3.2	32 (a)	32 (a)	No Impact
	32 (b) RNO	32 (b) RNO	No Impact
	32 (c) RNO	32 (c) RNO	No Impact
FR-C.1	Note Prior to Step 8	Note Prior to Step 8	No Impact
	Step 8 (a)	Step 8 (a)	No Impact
	Step 8 (b) RNO	Step 8 (b) RNO	No Impact
	Step 8 (c) RNO	Step 8 (c) RNO	No Impact
FR-Z.1	Step 7 (See Note 1)	Deleted	No Impact since it was already deleted from Rev. 1B
	Step 7 (a) (See Note 1)	Deleted	
	Step 7 (b) RNO (See Note 1)	Deleted	
	Step 7 (c) RNO (See Note 1)	Deleted	
	Step 9 (See Note 1)	Deleted	
FR-I.3	Step 12	Step 12	No Impact
NOTE 1: The entire Step 7 was deleted from Rev 1B in 1996 by the WOG ERG Maintenance			

The comparison of the ERG Rev. 1B and Rev. 1C shows that there are no differences in the two versions of the guidelines.

Addressing hydrogen in FR-Z.1 is not necessary from an accident management standpoint. A hydrogen challenge to the containment integrity would only exist when significant cladding oxidation occurred (for the WOG ERG reference plant design). Such oxidation would only occur under extreme conditions in the core (cladding temperatures in excess of 1700°F). The operator would be in either FR-C.1, FR-S.1, or ECA-0.0, under such conditions (all of which take priority over FR-Z.1). If core exit temperature exceeded 1200°F in FR-S.1 or ECA-0.0, the operator would transition to the SAMG and would never get to FR-Z.1. The only other location the operator could be in is FR-C.1, which already contains steps for checking hydrogen concentrations and starting the recombiners. Therefore, the operator would have already performed the hydrogen control steps that are contained in FR-Z.1.

For the design basis large LOCA, hydrogen generation is postulated to occur due to a small amount of cladding oxidation, radiolysis of water in the core region, and oxidation of aluminum and zinc in the containment. The hydrogen buildup from these sources is very slow (typically on the order of several days to several weeks) and is already addressed by the hydrogen control steps in E-1 and other guidelines where a LOCA may exist (eg.: ES-1.2, ECA-3.1, ECA-3.2). These guideline steps, which address the long term hydrogen concern, remain unchanged from Rev. 1B.

Based on the above discussion, addressing hydrogen in FR-Z.1 for the reference plant is not appropriate since there is no relationship between high containment pressure and a containment challenge due to a hydrogen burn. Eliminating these steps allows for faster performance of FR-Z.1, which is a benefit since the operator could be in FR-Z.1 during the initial recovery from a LOCA or steamline break. In this instance, a containment atmosphere sample cannot be obtained due to high containment pressure. In addition, there are no actions required with respect to containment hydrogen while in this guideline:

The containment atmosphere would be steam inerted and flammable mixture of hydrogen could not exist, and typical electric hydrogen recombiners, which are the hydrogen control measure in the ERGs for containment hydrogen concentrations above a prescribed setpoint value (typically 0.5 volume % hydrogen), are not operable at these pressures.

Once the containment pressure is returned to a value below the setpoint for this guideline and the guideline is exited, another check of containment hydrogen is made in a succeeding guideline.

Thus, there is no impact resulting from the removal of this guideline step.

A.1.6 Containment Sump Activity Level

There is one place in the generic WOG ERGs where the control room staff is instructed to obtain information regarding radioactivity in the containment sump. The purpose of this instruction is to provide information regarding possible transfer of containment sump water to other places in

the plant to avoid overfill of the containment and the subsequent possible loss of some equipment and/or instrumentation. The specific ERG where such instructions is given:

WOG ERG Rev. 1B vs. Rev. 1C Comparison Containment Sump Activity Level			
ERG Guideline	**Rev. 1B Step**	**Rev. 1C Step**	**Assessment of Impact of Differences**
FR-Z.2	Step 2	Step 2	No Impact

The comparison of the ERG Rev. 1B and Rev. 1C shows that there are no differences in the two versions of the guidelines.

A.1.7 Technical Plant Engineering Staff Support Center Guidance

The WOG ERG Maintenance Program provides a mechanism for utility members to provide feedback items based on plant observations, license operator simulator training, and/or industry events. The intent of this program is to maintain the ERGs as a living document and assure that they are current with industry experience. There is a WOG program to maintain the ERGs based on feedback from utilities. The utility feedback comes primarily from usage of the plant specific EOPs, which were derived from the WOG ERGs, during accident simulations for operator training and licensed operator examinations. One of the feedback items recommended the development of generic information to assist the plant engineering staff (e.g., the Technical Support Center) in making assessments that are referred to them by ERG steps. In other words, where an ERG step refers to a plant engineering staff evaluation, generic guidance for the plant engineering was recommended. This was completed in 1999 (reference 40). Several portions of this guidance are directly applicable to the proposed elimination of the post accident sampling system described in this report:

Evaluating Need and Actions for Venting Reactor Vessel Head (applicable to the need for sampling reactor coolant system for dissolved gases),

Evaluating Containment Sump pH (applicable to the need for sampling containment sump for pH),

Evaluating Containment Sump Level and Activity (applicable to the need for sampling containment sump for radionuclides),

Evaluating Containment Radiation Levels (applicable to the need for sampling containment atmosphere for radionuclides),

Evaluating Hydrogen Flammability and Recombiner Operability (applicable to the need for sampling containment atmosphere for hydrogen), and

Evaluating Plant Long Term Status (applicable to the need for all samples)

This guidance for the plant engineering staff was developed after the development of the technical basis for the elimination of the post accident sampling system contained in this report. Therefore, it provides guidance for the case where samples of plant fluids are available using the normal sampling system, as well as cases where no samples can be obtained due to high radiation levels in plant fluids. Therefore, the considerations in the plant engineering staff guidance directly support the elimination of the post accident sampling system.

A.2 Severe Accident Management Guidance

The SAMG diagnostics, as well as the SAMG monitoring of changes in plant state following implementation of recovery strategies, are based on fixed in-plant instrumentation. During the development of the SAMG it was determined that sampling could not provide timely information during the transient portions of a core damage accident. The SAMG development further concluded that sampling may only be of value after recovery is complete and the plant is in a controlled stable state. There is a current effort to provide updated SAMG information based on feedback from utilities during the use of plant specific SAMG (developed from the generic WOG SAMG) in SAMG table top drills. While this effort is not yet complete, there is no feedback to suggest that any post accident sampling function is useful in carrying out the SAMG diagnostics or recovery strategy assessments.

A.3 Summary and Conclusions

The technical basis for the proposed elimination of post accident sampling system was developed based on emergency response guidance for ERGs and SAMG that was current in 1996. An assessment of the impact of the latest versions of the ERGs and SAMG concludes that the technical basis described in this report is still valid. The most recent information available for plant engineering staff assessments while using the ERGs enhances the capabilities described in this report for diagnosing conditions and recommending appropriate actions for accident sequences that involve high radiation levels in plant fluids without the need for sampling.

APPENDIX B

Westinghouse Owners Group Responses to

NRC Questions and Comments

WOG LETTER OG -99-041

OG-99-041
April 28, 1999

Project Number 694
WCAP-14986-P, Rev. 1
WCAP-14987-NP, Rev. 1

Mr. Peter C. Wen
Project Manager,
Generic Issues and Environmental Projects Branch
Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Westinghouse Owners Group
Transmittal of Responses to NRC Comments from the March 25, 1999 Post Accident Sampling System Meeting (MUHP-3035)

Reference: 1) Westinghouse Owners Group Letter, OG-98-108, L.F. Liberatori to Document Control Desk, "Transmittal of Reports: WCAP-14986-P, Rev. 1 (Proprietary) and WCAP-14987-NP, Rev. 1 (Non-Proprietary), Entitled 'Westinghouse Owners Group Post Accident Sampling System Requirements: A Technical Basis'," October 26, 1998.

2) P. C. Wen, "Summary of March 25, 1999, Meeting with Westinghouse Owners Group Regarding WCAP-14986 WOG Post Accident Sampling System, April 2, 1999.

Dear Mr. Wen:

In October 1998 the Westinghouse Owners Group (WOG) submitted Westinghouse topical report WCAP-14986 Rev. 1, "Post Accident Sampling System (PASS) Requirements: A Technical Basis," (Ref. 1). At the NRC and Westinghouse Owners Group (WOG) meeting on March 25, 1999, the NRC provided a list of comments/questions on WCAP-14986 Rev. 1 (Ref. 2). These comments/questions and the associated WOG responses were discussed during the meeting. Attachment A provides additional clarification on specific comments/questions as requested by the Staff. Pending final resolution of these comments/questions, the WOG will revise WCAP-14986 as necessary. If you require further information, feel free to contact Mr. Ken Vavrek in the Westinghouse Owners Group Project Office at 412-374-4302.

Very truly yours,

Signed Original on File in WOG Project Office

Louis F. Liberatori, Jr., Chairman
Westinghouse Owners Group

attachments
OG-99-041
April 28, 1999

cc: WOG Steering Committee (1L, 1A)
WOG Primary Representatives (1L, 1A)
WOG Analysis Subcommittee Representatives (1L, 1A)
WOG Licensing Subcommittee Representatives (1L, 1A)
A. P. Drake, Westinghouse, ECE 5-16 (1L, 1A)
J. B. O'Brien, USNRC OWFN 9H15 (1L,1A)
R. J. Palla, Jr., USNRC OWFN 8H7 (1L,1A)

Attachment A

Responses to NRC Questions/Comments on WCAP-14986, Rev. 1
Post Accident Sampling System Requirements: A Technical Basis

A. General Comments/Questions - 1. Removal of Requirements from Licensing Basis

WCAP 14986 recommends that the capability to obtain certain samples be deleted as a requirement but retained to assist in planning long term recovery actions (but not within the plant licensing basis). It is not clear what is meant by removing the capability from the licensing basis. What controls would be applied to ensure licensee's capabilities would not be degraded or eliminated.

March 25, 1999 WOG/NRC Meeting Discussion:

The WOG discussed that the form and location of commitments relative to PASS differ among licensees with some being in license conditions, some in technical specifications, and others in the Final Safety Analysis Report. NRR discussed a desire for uniformity in the form and location of PASS commitment resulting as a result of licensees adopting the PASS topical. WOG Agreed that this was desirable.

March 25, 1999 WOG/NRC Meeting Actions:

The WOG took an action to evaluate how PASS commitments are captured under standard technical specifications. NRR to evaluate what controls may be appropriate for PASS.

RESPONSE:

Specification 5.5.3 in NUREG-1431, Rev. 1 contains the Programmatic requirement for PASS. The WOG proposes to delete this requirement from NUREG-1431 since it does not satisfy any of the criteria in 10CFR50.36, and is not consistent with the content of the Improved Standard Tech Specs. The WOG proposes that the PASS requirements be contained in the Final Safety Analysis Report. The plant specific PASS requirements will be determined in accordance with WCAP-14986.

B. General Comments/Questions - 3. Plugging of PASS Lines

WCAP-14986 states that one rational for not taken PASS samples is due to the potential for plugging in sample lines. However, in accordance with NUREG-0737, these sample lines were designed to prevent plugging and plateout. Further, information is needed to evaluate the predicted extent of plugging and plateout in sample lines (see also specific Comment #3).

March 25, 1999 WOG/NRC Meeting Discussion:

WOG stated that there are two issues: (1) plugging of lines due to aerosol production from core concrete interaction, and (2) plateout in sample lines. NRR stated that it is further evaluating this issue considering guidance provided in Regulatory Guide 1.21.

RESPONSE:

At the time that the regulatory requirements related to post accident sampling capabilities were developed, the state of knowledge of severe accidents recognized the potential for a number of concerns that could impact the reliability of the samples and/or their analysis results. These concerns were limited to those specified in Clarification 11 to NUREG-0737 Item II.B.3, which states that the PASS should have "Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment." The known concerns did not include the potential for sample line plugging by deposition of aerosol material in the samples lines.

Based on current knowledge of severe accidents, plugging of sample lines by dense concentrations of aerosols in the containment would be another consideration. The phenomena and model are described by Morewitz in "Leakage of Aerosols from Containment Buildings" Health Physics, Vol. 42, No. 2, pp. 195-207. This is also discussed in the EPRI Technical Basis Report for Severe Accident Management, TR-101869, pg. 3-64 of Volume 1 and Appendix CC of Volume 2. Based on the EPRI work, the potential for aerosol plugging is also included as a potential negative impact at various places in the WOG Severe Accident Management Guidance.

The potential for plugging of sample lines is primarily during periods of extensive aerosol generation, for example during core concrete interactions without a water cover. These conditions are well beyond the point in the accident progression where sampling would be useful to assess the state of the core, since the core damaged is already extensive. For example, the response to NRC RAI #32 on the Wolf Creek IPE Submittal, indicates that the total amount of aerosol in the containment prior to core concrete interactions is less than 0.1 cubic meter, based on the entire core inventory of iodine and cesium. Thus, it can be reasonably assumed that sample lines would not plug in this time frame of an accident. Core damage assessment, EALs and PARs after reactor vessel failure can easily be made based on plant parameters obtained from fixed in-plant instrumentation and sampling would not be useful for validation or refinement.

With respect to the other issues identified in Clarification 11 to NUREG-0737, generically, the post accident sampling system design reduces/minimizes losses during sampling. Further, the current uses of PASS results recognize the uncertainties associated with fission product behavior that cannot be eliminated through PASS design. For example, if the radionuclide is an aerosol, there is no way to completely eliminate mechanical deposition in sample lines. As a result, the current WOG PASS methodology requires comparison of core damage estimates made from a number of different radionuclides and provides guidance for resolving any differences that may occur (as for example through deposition of aerosol radionulcides in sample lines).

Therefore, we conclude that on a generic basis, the current PASS systems meet the regulatory requirements and the intent of the regulatory requirements.

C. Specific Comment - 3. Page 8, Paragraph 2 Comment - Plugging of Sample Lines

WCAP-14986 states that the recent analytical studies have indicated that dense aerosol concentration in the RCS and containment could plug sample lines. Please provide the referred studies. Clarification 11 to NUREG-0737 Item II.B.3 states that the PASS should have "Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment." Please provide further information on why the provision put in place to meet this NUREG-0737 item are not adequate.

March 25, 1999 WOG/NRC Meeting Discussion:

WOG stated that plugging of sample lines may occur due to aerosol produced from core concrete interactions after the core has gone ex-vessel.

RESPONSE:

See response to Item B, directly above.

D. Specific Comment - 5. Page 9, Paragraph 2 Comment - EALs for Reactivity Excursions

WCAP-14986 states that radiological analysis of plant fluids was not useful for emergency action level classification. It is not clear what spectrum of accidents were evaluated that lead to this conclusion. The criteria included in the EAL scheme equates to approximately 2 - 5% clad damage. At these levels of clad damage, it is not clear that other parameters would prompt classification. Paragraph 3 on page 16 describes classification of a reactivity excursion event. It seems a reactor coolant sample may be the only indication of clad damage for this event.

March 25, 1999 WOG/NRC Meeting Discussion:

Due to time constraints, this item was not discussed in detail at the meeting.

RESPONSE:

The first sentence of the text from WCAP-14986-P referenced in this question may be misleading. The PASS information is not timely for emergency classification. Emergency classification and protective actions are expected to be evaluated within a short period of time after identifying a negative trend on plant parameters/conditions. The process of obtaining and analyzing a sample is generally outside the time frame required for emergency classification and protective action decisions and has therefore been characterized in WCAP-14986-P as "not useful".

Further, as described on page 9 of WCAP-14986-P, a full range of accidents was considered in the assessment that led to the conclusion that radiological analyses of plant fluids was not required to make appropriate Emergency Action Level (EAL) classifications. With respect to reactivity excursion accidents typically described in the Safety Analysis report, only the uncontrolled rod withdrawal from full power and the control rod ejection accidents are postulated to potentially result in fuel rod cladding damage. In the case of the rod ejection accident, the RCS also depressurizes and the appropriate EAL classification can be made based on containment radiation levels and loss of the RCS barrier. For the uncontrolled rod withdrawal event, there is no loss of reactor coolant to the containment. In this case, the reactor is tripped and shutdown by the nuclear instrumentation. The first indication of fuel rod failures would be by means of alarms on the letdown radiation monitor or the area radiation monitors in the auxiliary building in the vicinity of the letdown system, caused by shine from the letdown system piping. Also, shine from reactor coolant piping would result in increased readings from area radiation monitors in the containment. According to the NEI EALs in NEI-97-03, Rev. 3 (reference 7 in WCAP-14986-P), the unexpected increase in plant radiation levels would trigger an Unusual Event EAL classification. This is the same level that high radiation levels in a reactor coolant system sample from analysis of PASS samples would trigger. Thus, for the analyzed reactivity excursion accidents, the proper EAL classification would be made based on instrumentation indications rather than analysis of PASS samples.

E. Specific Comment - 6. Page 11, Paragraph 5 - Heat tracing for sample lines

WCAP-14986 state[s] that the NRC approved deletion of the requirement for heat tracing of sample lines. The NRC stated in that a licensee that utilizes iodine in its core damage assessment procedures must include appropriate design consideration to ensure representative sampling. It is not clear how xenon and/or krypton isotopic analyses can be used to ascertain the degree and <u>type</u> of core damage.

March 25, 1999 WOG/NRC Meeting Discussion:

Due to time constraints, this item was not discussed in detail at the meeting.

RESPONSE:

The proposed revision to the WOG Core Damage Assessment Methodology, as described in WCAP-14696, does not require analysis of PASS samples for radioactive iodines. Therefore, according to the staff's Safety Evaluation Report on the Combustion Engineering Owners Group (CEOG) submittal in CEN-415, the deletion of heat tracing from samples lines is permitted. As further described in WCAP-14696 (Core Damage Assessment) and the subsequent WOG response to RAI#1 for WCAP-14696 (Ref. 1), noble gases may be the only radionuclide that is reasonable to evaluate if validation of core damage assessments made using the WCAP-14696 guidance is sought by the plant engineering staff.

F. Specific Comment - 9. Page 43, Section 4.3.2 - Accuracy of Source Term Data

WCAP-14986 provides information regarding the use of source term information to make dose assessment. This section discusses various ways that the source term can be obtained, including use of precalculated values (e.g., from design basis or PRA sources) or use of containment high range area radiation monitor, but does not provide information on the relative accuracy of the different means of obtaining source term data.

March 25, 1999 WOG/NRC Meeting Discussion:

WOG stated that it did not attempt to quantify uncertainty in the different source term data. The topical report discusses the issues inherent in assessing the uncertainties in a source term that might be derived from PASS samples.

RESPONSE:

The sensitivity of offsite dose levels to the source term quantities of noble gases, volatiles and non-volatiles was first studied in detail in the Seabrook Emergency Planning studies in the late 1980's (Ref. PLG-0432, PLG-0465 and PLG-0550). In these studies, the 24 hour equivalent whole body doses to an individual in the "plume pathway" Emergency Planning Zone was the controlling dose measure of offsite consequences. Using a baseline source term similar to that in NUREG-1465, the Seabrook studies concluded that additional reductions in particulate source term (all nuclides other than noble gases, iodine and cesiums) would not significantly reduce doses or the distance at which the knee in the CCDF occurs because doses are primarily the result of plume shine doses from noble gases in the plume (pg. 5-7 of PLG-0432). A review of the bases for this conclusion shows that the converse of this conclusion would also be true; namely that reasonable increases (e.g., less than a factor of 10) in the particulate source term would not significantly increase doses or the distance at which the knee in the CCDF occurs.

This insight is confirmed in the recent severe accident offsite dose analyses for the Westinghouse AP600 design (Ref.AP600 PRA Section 49). The 24 hour doses are of the same order of magnitude for all of the source term categories with similar noble gas releases, irrespective of the quantities of other radionuclides released. For example, in Table 49-3, the mean 24 hour dose for the CFI source term is 66% of the that for the BP source term where in Table 40-2, the ratio of the noble source terms is 0.72, the ratio of the iodine and cesium source terms is nearly 1.0 and the ratio of all other radionuclides is at least less than 0.1.

Additionally, information provided in WCAP-14969, related to the revised WOG Core Damage Assessment methodology, shows that the containment radiation levels do not change in direct proportion to the amount of volatile and non-volatile radionuclides assumed to be airborne in the containment. In particular, Figure 4 of WCAP-14696 shows that there is less than a factor of 3 difference in the containment radiation level for the case of no volatiles and nonvolatiles in the containment atmosphere (curve labeled 100% noble gas) vs. the case with 50% of the volatiles and non-volatiles released to the RCS being airborne in the containment.

This level of uncertainty in the offsite dose projections derived from a source term based on the containment radiation monitor would not be expected to result in variations in the recommended protective action recommendations for protection of the health and safety of the public. Further, this level of uncertainty is likely to be less than the uncertainties inherent in estimating offsite doses from radionuclide samples of plant fluids.

G. Specific Comment - 11. Page 44, Section 4.3.3 - Reactivity Excursion Events

WCAP-14986 discusses an assessment performed to identify possible accident sequences where the RCS fission product inventories can approach EAL trigger values. Please provide details of the assessment. In particular, provide information on how reactivity excursion events and events involving the potential for clad rupture due to flooding with cold water were evaluated.

March 25, 1999 WOG/NRC Meeting Discussion:

Due to time constraints, this item was not discussed in detail at the meeting.

RESPONSE:

See WOG response to Item D in this submittal.

References:

Westinghouse Owners Group Letter, OG-99-040, L. F. Liberatori, Jr. to Document Control Desk, "Response to NRC Request for Additional Information on WCAP-14696, "Westinghouse Owners Group Core Damage Assessment Guidance," Non-Proprietary," April 28, 1999.