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NRC Project 691

BWROG-00089
November 30, 2000

US Nuclear Regulatory Commission
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SUBJECT: ***BWR OWNERS' GROUP LICENSING TOPICAL REPORT NEDO-32991 "REGULATORY RELAXATION FOR BWR POST ACCIDENT SAMPLING STATIONS (PASS)" DATED OCTOBER 2000***

Attachment: NEDO-32991 "Regulatory Relaxation for BWR Post Accident Sampling Stations (PASS)" dated October 2000

The BWR Owners' Group (BWROG) has completed an evaluation regarding the benefits of Post Accident Sampling Stations (PASS) that are installed in U.S. BWRs. The result of this evaluation is that the installed Post Accident Sampling Stations (PASS) do not provide the benefits expected by the NRC when the requirements were imposed 20 years ago following the Three Mile Island Unit 2 accident. In-plant instruments and the associated analysis methods are as good as, or better than PASS for collecting and assimilating information to assess core damage following an accident. In addition, Emergency Operating Procedures and Severe Accident Management Guidance methods are based on decisions made using in-plant instrumentation. The BWROG is in the process of developing a revised BWR Core Damage Assessment procedure which will rely exclusively on plant instrumentation and known fuel release characteristics to assess core damage and make plant recovery decisions. The BWROG recommendation is to eliminate all PASS regulatory requirements. BWRs may, however, elect to maintain certain portions of the PASS for plant specific contingencies.

The BWROG has also considered the effect of removing the PASS System from a safety risk perspective. None of the BWR Probabilistic Risk Assessments (PRAs) rely upon or address PASS Systems and, therefore, quantitative risk assessments cannot be made. The risk insights based on review of normal operating, emergency, and severe accident conditions indicate that the existence or non-existence of PASS Systems would have no affect on core

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damage or large early release frequencies (LERF).

The BWROG requests that this Licensing Topical Report be reviewed in anticipation of use of the Consolidated Line Item Improvement Process (CLIIP) similar to the submittals described in the Federal Register on August 11, 2000 (Volume 65, number 156). After the NRC reviews and approves this Licensing Topical Report and prepares a model safety evaluation (SE) and associated no significant hazards consideration (NSHC), applicable changes will be incorporated into the Standard Technical Specifications in a manner that supports subsequent license amendment applications. Participating BWR owners to which these NRC-developed models apply will then request amendments confirming the applicability of the SE and the NSHC determination to their reactors. These amendments will also provide the requested plant-specific verifications and commitments. This approach minimizes the resource demand on both the licensee and the NRC for developing individualized justifications and for conducting regulatory reviews, respectively.

This letter has been endorsed by a substantial number of the members of the BWR Owners' Group; however, it should not be interpreted as a commitment of any individual member to a specific course of action. Each member must formally endorse the BWROG position in order for that position to become the member's position.

The BWR Owners' Group is looking forward to your timely review of this Licensing Topical Report. If you desire to discuss this information in more detail, please contact the General Electric Project Manager, TA Green at (408) 925-1308 or the undersigned.

Very truly yours,

J. M. Kenny

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BWR Owners' Group
Licensing Topical Report

Regulatory Relaxation for BWR Post-Accident Sampling Stations (PASS)

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ACRONYMS AND ABBREVIATIONS

ALARA	As Low As Reasonably Achievable
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners' Group
BWRVIP	Boiling Water Reactor Vessel and Internals Program
CDAG	Core Damage Assessment Guidline
CEOG	Combustion Engineering Owners' Group
CFR	Code of Federal Regulations
DNB	Departure from Nucleate Boiling
EOP	Emergency Operating Procedures
EPRI	Electric Power Research Institute
IGSCC	Intergranular Stress Corrosion Cracking
LERF	Large Early Release Frequency
LOCA	Loss-of-Coolant Accident
LTR	Licensing Topical Report
NRC	Nuclear Regulatory Commission
NSHC	No Significant Hazards Consideration
NSSS	Nuclear Steam Supply System
PASS	Post-Accident Sampling Station
PRA	Probabilistic Risk Assessment
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
RPS	Reactor Protection System
SAG	Severe Accident Guideline
SAR	Safety Analysis Report
SCC	Stress Corrosion Cracking
SE	Safety Evaluation
SER	Safety Evaluation Report
SLCS	Standby Liquid Control System
SGTS	Standby Gas Treatment System
TRM	Technical Requirements Manual
WOG	Westinghouse Owners' Group

1.0 INTRODUCTION

The BWR Owners' Group (BWROG) is in the process of preparing justifications to eliminate or declassify specific BWR system requirements where the cost to maintain and test the systems is excessive with respect to the safety benefit.

Accident response requirements [Ref. 1] specify that utilities must keep local, state and federal authorities informed of the plant status and the potential for activity release during any type of reactor incident. This information, which includes projected activity release rate and environmental dose estimates, must be provided on a timely basis and must be continually upgraded as conditions change. In addition, authorities must be notified within 15 minutes of declaring a site alert. These accident response requirements are beyond the capability of Post-Accident Sampling Sations (PASS) and associated analysis. Consequently, utilities typically estimate fuel damage from in-plant instrumentation such as containment radiation levels and containment hydrogen concentration in conjunction with knowledge of fuel failure and activity release as a function of reactor accident conditions. Most BWRs currently employ PASS as a means for confirmatory information only. This is in response to previous commitments to provide fuel damage assessment that is based on results of PASS analyses.

The current BWROG Core Damage Assessment Guideline (CDAG) is outdated. The BWROG is currently preparing a revised CDAG consistent with current accident source terms [Ref. 10] and relying exclusively on in-plant instruments and known fuel release characteristics.

The Westinghouse Owners' Group (WOG) and Combustion Engineering Owners' Group (CEOG) have been actively engaged in reducing or eliminating the PASS sampling and analysis requirements. In November 1996, Westinghouse transmitted WCAP-14696, "WOG Core Damage Assessment Guidance", to the NRC. This procedure utilizes installed plant instrumentation for real-time accident management decisions. In addition, calculational methods using known fuel behavior as a function of operational parameters are used to predict release conditions when plant instrumentation is insufficient. In September 1999, the NRC issued a safety evaluation of WCAP-14696, Rev. 1, stating this procedure was an acceptable basis for meeting the NUREG-0737 sampling and analysis requirements [Ref. 2].

In May/June 2000, the NRC issued Safety Evaluation Reports to the CEOG and WOG to facilitate elimination of the PASS requirements for the applicable PWRs. It should be noted that minimal plant-specific evaluations are required before Post-Accident Sampling Stations can be removed from these plants. The BWROG participating utilities will implement an I-131 site survey detection capability that will be applicable to all accident scenarios and

release points. This will provide an alternate means for dose projections. The net impact of this change will be a positive enhancement to plant safety for BWRs that currently do not have this capability. In addition, BWR owners will commit to maintain a contingency plan for obtaining and analyzing highly radioactive reactor coolant, suppression pool, and containment atmospheric samples.

The BWROG requests that this Licensing Topical Report be processed in anticipation of use of the Consolidated Line Item Improvement Process (CLIIP) similar to the submittals described in the Federal Register on August 11, 2000 (Vol. 65, No. 156). After the NRC reviews and approves this Licensing Topical Report and prepares a model Safety Evaluation (SE) and associated No Significant Hazards Consideration (NSHC), applicable changes will be incorporated into the Standard Technical Specifications in a manner that supports subsequent license amendment applications. Participating BWR owners to which these NRC-developed models apply will then request amendments confirming the applicability of the SE and the NSHC determination to their reactors. These amendments will also provide the requested plant-specific verifications and commitments. This approach minimizes the resource demand on both the licensee and the NRC for developing individualized justifications and for conducting regulatory reviews, respectively.

2.0 EXECUTIVE SUMMARY

The results of a BWROG evaluation confirm that BWR Post-Accident Sampling Stations (PASS) do not provide the benefits expected by the NRC when the requirements were imposed 20 years ago following the Three Mile Island Unit 2 accident. In-plant instruments, in conjunction with analysis methods based on known fuel release characteristics, are as good as or better than PASS for collecting and assimilating information to assess core damage following an accident. In addition, BWR emergency and severe accident response strategies are based on use of available instrumentation and do not require use of PASS. The BWROG is in the process of developing a revised BWR Core Damage Assessment Guideline that relies exclusively on in-plant instrumentation and known fuel release characteristics to assess core damage and make the required accident mitigation decisions. The BWROG recommendation is to eliminate all PASS regulatory requirements. BWRs may, however, elect to maintain certain portions of the PASS for plant-specific contingencies.

The BWROG has also considered the effect of removing the PASS from a safety risk perspective. None of the BWR Probabilistic Risk Assessments (PRAs) relies upon or addresses the PASS and, therefore, quantitative risk assessments cannot be made. The risk insights based on review of normal operating, emergency, and severe accident conditions indicate that the existence or non-existence of the PASS would have no effect on core damage or large early release frequencies (LERF).

In order to provide an alternate means for dose projections, the BWROG participating utilities will implement an I-131 site survey detection capability that will be applicable to all accident scenarios and release points. This commitment will be met by use of existing effluent monitoring systems or through the analysis of samples obtained by portable sampling equipment. The net impact of this change will be a positive enhancement to plant safety for BWRs that currently do not have this capability. In addition, BWR owners will commit to maintain a contingency plan for obtaining and analyzing highly radioactive reactor coolant, suppression pool, and containment atmospheric samples.

3.0 POST-ACCIDENT SAMPLING SYSTEM REGULATORY REQUIREMENTS

As part of the TMI Action Plan Requirements, the NRC instituted requirements for post-accident instrumentation and for post-accident sampling and analysis. These requirements were promulgated through Regulatory Guide 1.97 [Ref. 3] and through NUREG-0737 [Ref. 4] and its various clarification letters. The primary purpose of the sampling and analysis requirements is to provide information regarding the extent of reactor core damage under all accident conditions. Utilities were also required to establish procedures for determining the extent of core damage based on the results of coolant, sump (suppression pool), and containment atmosphere sampling and analysis. It was required that the analyses needed to assess the core damage be completed within three hours of the decision to take a sample. Based on a clarification provided by the NRC, the extent of core damage was to be assessed in terms of the following matrix:

Degree of Degradation	Minor (<10%)	Intermediate (10%-50%)	Major (>50%)
1. No Fuel Damage	--	--	--
2. Cladding Failure	X	X	X
3. Fuel Overheat	X	X	X
4. Fuel Melt	X	X	X

As recommended by the NRC, there are four general classes of damage and three degrees of damage within each of the classes except for the "No Fuel Damage" class. Consequently, there are a total of 10 damage assessment categories. The conditions of more than one category could exist simultaneously. The objective of the final core damage assessment is to narrow down, to the maximum extent possible, those categories which apply to the actual in-plant situation.

In June 1983, the BWR Owners' Group submitted a generic guideline to the NRC [Ref. 5] for assessing the extent of core damage. This submittal included GE report NEDO-22215, "Procedure for the Determination of the Extent of Core Damage Under Accident Conditions", [Ref. 6] plus an attachment titled "Integration of Other Plant Parameters into Core Damage Estimate". The other plant parameters included containment atmosphere hydrogen measurement and containment atmosphere radiation measurement. The generic radiological guideline was based primarily on WASH-1400 [Ref. 7]. This guideline provides methods for estimating the extent of core damage by comparing the observed activity release to the maximum expected activity release. The maximum expected activity release is based

on the Regulatory Guide 1.3 [Ref. 8] source terms (100% release of noble gases, 50% release of iodines with 50% of the iodine release being volatilized, and 1% release of non-volatile species). Most BWR utilities adopted the guideline as a basis for their site-specific core damage assessment procedure.

Since the 1979 TMI accident, however, there have been extensive studies of fuel behavior and fission product release and transport under accident conditions. Among other things, this has resulted in a new set of NRC-approved accident source terms [Ref. 10]. A detailed discussion of specific regulatory requirements, implementation issues, and proposed or accepted relaxation of these requirements is contained in the following subsections.

3.1 NUREG-0737

NUREG-0737, published in November 1980 [Ref. 4], consolidated the TMI-related action items that had been approved by the Nuclear Regulatory Commission (NRC) for implementation. The document included clarifications to many of the previously issued requirements, and a caveat that additional requirements would be forthcoming.

The post-accident primary system sampling and analysis requirements are covered in Section II.B.3 of NUREG-0737 and are summarized in Table 3-1 of this Licensing Topical Report. The accident monitoring instrumentation and gaseous effluent sample and analysis requirements are covered in Section II.F.1. These sections of NUREG-0737, including the 11 clarification items, were applicable to all plants. The sampling systems were to be designed to handle a Regulatory Guide 1.3 [Ref. 8] source term; namely, a 100% release of the core inventory of noble gases plus a 50% release of the core inventory of iodines to the primary containment with half of this iodine being volatilized. Unless indicated otherwise, analyses are to be completed within three hours of the decision to take a sample. Licensees were to provide the NRC with a detailed description of their post-accident sample and analysis systems. This description was to include P&IDs and summary descriptions (or copies) of the procedures for sample collection, sample transfer or transport, and sample analysis.

Section II.B.3

Primary System Sampling and Analysis: In 1982 the NRC sent a letter to operating reactors describing the criteria by which the staff would conduct a post-implementation review of post accident sample systems [Ref. 11]. These Criteria Guidelines restate the 11 NUREG-0737 primary system sampling and analysis requirements and the 11 clarification items. In the Guidelines, the original 11 clarification items are identified as "Criterion" and a new, detailed and prescriptive "Clarification" is given for each "Criterion". Most significantly, these clarifications specified analysis sensitivities and accuracy, and defined a standard chemical and radiation test matrix for evaluating the capabilities of the proposed analysis procedures.

Section II.F.1

Containment Hydrogen Monitor: A continuous indication of the hydrogen concentration in the containment atmosphere shall be provided in the control room. Originally, it was stated that the range should be 0 to 10% hydrogen under both positive and negative ambient pressure. It was noted that Regulatory Guide 1.97, Rev. 2, had not been finalized, but the section of Regulatory Guide 1.97 regarding the hydrogen monitors had been appended to the NUREG-0737 clarification letter [Ref. 11] and was to be considered as the new NUREG-0737 requirement.

Containment High-Range Radiation Monitors: A minimum of two in-containment radiation monitors with a maximum range of 10^8 rads/hr (gamma plus beta, 10^7 rads/hr gamma only) is required. The monitors must be physically separated and located such as to view a large segment of the containment atmosphere. For a BWR Mark III, two such monitor systems are required to be installed in both the drywell and containment.

Noble Gas Effluent Monitors: Gross activity, noble gas effluent monitors are required at all gaseous activity release points. Some of the basic requirements are:

- Monitors range of activity from normal operation to that of the design basis accident.
- Provides continuous monitoring of the high-level post accident noble gas activity release.
- Multiple, overlapping range detectors may be used to cover the full range of activity.

Iodine and Particulate Effluent Samplers: Continuous post-accident iodine and particulate gaseous samplers are required at all gaseous activity release points whenever exhaust flow occurs.

3.2 REGULATORY GUIDE 1.97

Regulatory Guide 1.97 Revision 2, published in December 1980 [Ref. 12], is concerned primarily with the instrumentation requirements for assessing the status of the plant and environs during and following an accident. It also incorporated the post-accident sampling and analysis program of NUREG-0737 and, therefore, specified the liquid and atmospheric analyses to be performed, and a concentration range for these analyses.

Revision 3 to Regulatory Guide 1.97 was published in May 1983 (Reference 3). There were minor changes in the chemistry and radiological requirements, and most of the changes were in the footnotes to the requirements. The most significant changes were that fresh water plants had 96 hours from the time of primary coolant sampling in which to perform a chloride analysis; and primary coolant dissolved oxygen need not be determined for the first 30 days post-accident, provided the chloride concentration was less than 0.15 ppm. If the chloride

concentration increases to 0.15 ppm, oxygen should be determined within 2 hours. A paragraph was added to Section 1.4 of the Regulatory Guide describing the qualification requirements for the instrumentation:

> "In general, Category 1 provides for full qualification, redundancy, and continuous real-time display and requires on-site (standby) power. Category 2 provides for qualification but is less stringent in that it does not (of itself) include seismic qualification, redundancy, or continuous display and requires only a high-reliability power source (not necessarily standby power). Category 3 is the least stringent. It provides for high-quality commercial-grade equipment that requires only off-site power."

A list of "Design and Qualification Criteria for Instrumentation" was then provided to detail the requirements for each of the three categories. All of the post-accident sampling requirements which the BWROG is proposing to eliminate are classified as Category 3.

Table 3-2 of this LTR compares the Regulatory Guide 1.97 analyses requirements with those given in the NUREG-0737 Evaluation Criteria Guideline letter [Ref. 11]. (There are no analysis ranges in the NUREG-0737 document itself.) Except for boron analysis, where it is suspected that the Criteria Guidelines upper limit is in terms of ppm boric acid (not boron), the requirements are very similar. The most significant difference is that the NUREG-0737 evaluation criteria letter set limits other than zero for the analysis sensitivity. The NRC also raised the required level of sensitivity and lowered the accuracy requirements for dissolved oxygen and hydrogen. Regarding gaseous effluent monitoring and analysis, the only difference is that the Regulatory Guide specifies minimum as well as maximum levels of activity. Both documents specify 100 μCi/cc upper limits for iodine activity.

3.3 TECHNICAL SPECIFICATION REQUIREMENTS

On November 1, 1983, the NRC published Generic Letter No. 83-36, "NUREG-0737 Technical Specifications" [Ref. 13]. Enclosure 1 of this letter, under item (2), Post-Accident Sampling (II.B.3), states:

> "Licensees should ensure that their plant has the capability to obtain and analyze reactor coolant and containment atmosphere samples under accident conditions. An administrative program should be established, implemented and maintained to ensure this capability. The program should include:
>
> a) training of personnel,
>
> b procedures for sampling and analysis, and
>
> c) provisions for maintenance of sampling and analysis equipment.

It is acceptable to the Staff, if the licensee elects to reference this program in the administrative controls section of the Technical Specifications and include a detailed description of the program in the plant operation manuals. A copy of the program should be readily available to the operating staff during accident and transient conditions."

The Standard Improved Technical Specifications (NUREG-1433, NUREG-1434) contain similar non-specific PASS requirements. The PASS system is addressed in Section 5.5 (Programs and Manuals) and in the Tech Spec Bases B 3.3.3.1 (Post-Accident Monitoring Instrumentation Action D1 references PASS as a justification to allow both hydrogen monitors to be out of service for 72 hours).

3.4 MODIFICATIONS TO NUREG-0737 AND RG 1.97

As a result of discussions between the NRC and GE/BWROG, several modifications have been made in the NUREG-0737 and Regulatory Guide 1.97 requirements, including (1) primary coolant dissolved gas measurements, (2) containment atmosphere iodine analysis and (3) elimination of containment sump sampling. Individual plants have also obtained changes in the requirements.

3.4.1 Relaxation of Primary Coolant Dissolved Gas Measurements

On January 6, 1984, GE issued a letter to the PASS Owners' Design Group stating that at a December 12, 1983 meeting, the NRC agreed to relax the dissolved hydrogen requirements [Ref. 14].

Reference 15 documents the agreement reached between the NRC and GE at the December 12, 1983 meeting. The total dissolved gas was to be determined by measuring the pressure rise on expanding a known volume of liquid sample into a partially evacuated chamber. The accuracy was stated to be at least ±50% for dissolved gases between 25 cc/kg and 50 cc/kg and at least ±30% for dissolved gas concentrations greater than 50 cc/kg.

In July 1984, the NRC issued a letter [Ref. 16] summarizing a May 4, 1984 meeting with GE/BWROG. This letter stated that the NRC agreed with the position that the upper limit for total dissolved gas was 400 cc/kg and that the dissolved gas grab sample capability would be sufficient to monitor dissolved oxygen with an accuracy of at least ±60% at 4-8 ppm and at least ±30% at 8-20 ppm.

3.4.2 Elimination of Containment Sump Sampling Requirement

In July 1984, the NRC agreed (Reference 16) with the position that sampling the suppression pool satisfied the intent of the Regulatory Guide 1.97 requirement to provide containment sump sampling capability.

3.4.3 Post-Accident Sampling of Radionuclides in Containment Atmosphere

In 1987 in response to a GE inquiry for clarification of the NUREG-0737 requirement for radiological analysis of the containment atmosphere, the NRC stated [Ref. 17]: "The selection of which specific nuclides are to be measured is the licensee's responsibility." Some utilities have used this as justification for eliminating containment atmosphere particulate and iodine sampling, since the results are not used in their core damage assessment procedures.

3.4.4 Detection of Cladding Breach

Regulatory Guide 1.97 requires monitoring of the radioactivity concentration or radiation level in circulating primary coolant in order to detect a breach in the fuel cladding. The BWROG position is that the intent of this requirement has been satisfied by the combination of:

- Offgas pretreatment radiation monitor
- Main steamline radiation monitors
- Containment area radiation monitor
- Containment hydrogen monitor
- Post-accident sample station (manual sample analysis)

3.5 PROPOSED REGULATORY RELAXATIONS

In 1984, the NRC initiated a program to review light water reactor regulatory requirements to determine if some of the requirements could be relaxed or eliminated without compromising public health and safety. Pacific Northwest Laboratory was commissioned to evaluate the benefits of possible modifications, including post-accident sampling and analysis. The significant conclusions were [Ref. 9]:

- With the possible exception of boron analysis, the timing of PASS sample results had a marginal or negligible effect on public risks due to reactor accidents.

- Review of the benefits of radiological analysis of coolant samples during the accident management, emergency response, and plant recovery phases of an accident indicated that the information may not be available in time for accident management and emergency response. Eliminating the analysis requirements during the accident management and emergency response phases would have a marginal effect on the risks to the public. This conclusion was due, in part, to the fact that other indicators of core damage such as containment hydrogen concentration, containment radiation levels, and potential source terms are more readily available.

- PASS information was found to be useful in planning plant recovery actions. However, the benefits of PASS are primarily related to reductions in occupational exposure rather than protecting the public from the consequences of an accident.

Because of the need to assess the core status and activity release potential in a timely manner, utilities developed accident assessment procedures that do not rely on taking samples. These procedures are based on fuel failure modes and activity release as a function of fuel temperature in addition to in-plant instrumentation, such as containment radiation level and containment hydrogen monitors. This information, coupled with environmental radioactivity measurements, allows accurate prediction of the release potential. It is also noted that the PASS sampling and analysis processes have major shortcomings with respect to the ability to determine the extent of core damage:

- Post-accident sampling and analysis cannot satisfy the time requirements for evaluating core conditions and predicting activity releases.
- Post-accident sampling and analysis is of no benefit during transient conditions.
- PASS sampling during the accident management and emergency response phases of an accident is an unnecessary diversion of control room personnel who must open and close the requisite isolation valves and otherwise monitor the sampling activities. This sampling is also an unnecessary burden to the health physics organization who must grant access to the sampling areas and maintain cognizance of the sampling operation.
- Since there is no need for samples during the management and emergency response phases of an accident, sampling imposes an unnecessary exposure to sampling personnel in addition to the risk of contamination due to spills or leakage.
- The PASS is designed for sampling coolant that contains soluble impurities. If the severe accident involves suspended material such as insulation debris, it may be impossible to obtain a representative sample and there is a possibility of plugged sample lines. There also are concerns related to selective deposition in the sample lines.
- These shortcomings support the recommendation that PASS sampling and analysis be eliminated as regulatory requirements as discussed and justified in Section 4.

Table 3-1
NUREG-0737 ANALYSIS REQUIREMENTS

Analysis	Range	Accuracy	Reason
Primary Coolant			
Gross Activity and Gamma Spectroscopy	1 μCi/g to 10 Ci/g	Within factor of 2	Estimate Core Degradation
Boron	1000 to 6000 ppm	+5%	Verify Shutdown Margin
	<1000 ppm	±50 ppm	
Chloride	0.5 to 20 ppm	±10%	Coolant Corrosion Potential
	<0.5 ppm	±0.05 ppm	
pH (see Note 1)	5 to 9	±0.3 pH units	Coolant Corrosion Potential
	<5 to >9	±0.5 pH units	
Total Dissolved Gas or Dissolved Hydrogen	50 to 2000 cc/kg	±20%	Estimate Core Degradation And Coolant Corrosion Potential
	<50 cc/kg	±5.0 cc/kg	
Dissolved Oxygen (see Note 2)	0.5 to 20.0 ppm	±10%	Coolant Corrosion Potential
	<0.5 ppm	±0.05 ppm	

Table 3-1 (Continued)
NUREG-0737 ANALYSIS REQUIREMENTS

Analysis	Range	Accuracy	Reason
Containment Atmosphere			
Gamma Spectroscopy	Not Specified (see Note 3)	Not Specified	Estimate Core Degradation
Hydrogen	Not Specified	Not Specified	Estimate Core Degradation and Explosion Potential
Iodines and Particulates	Not Specified (see note 3)	Not Specified	Estimate Core Degradation

NOTES:

1. pH not specifically addressed in NUREG-0737, however, range and accuracy requirements appear in criteria guideline letter.

2. NUREG-0737 states that "measuring the O_2 concentration is recommended, but is not mandatory".

3. 100 μCi/g specified as design basis for shielding.

Table 3-2

COMPARISON OF RG 1.97 AND NUREG-0737 ANALYSIS REQUIREMENTS

Analysis	Regulatory Guide 1.97 Requirements	NUREG-0737 Criteria Guidelines Letter Range & Accuracy
Primary Coolant and Sump		
Gross Activity	10 μCi/ml to 10 Ci/ml, Rev. 2 1 μCi/ml to 10 Ci/ml, Rev. 3	Not required
Gamma Spectrum	10 μCi/ml to 10 Ci/ml, Rev. 2 and 3	Accurate within a factor of 2 over the RG 1.97 range
Boron	0 to 1000 ppm	1000 to 6000 ppm ± 5%[a] <1000 ppm ± 50 ppm
Chloride	0 to 20 ppm	0.5 to 20 ppm ± 10% <0.5 ppm ± 0.05 ppm
pH	1 to 13	5 to 9, ± 0.3 pH <5 and >9, ± 0.5 pH
Dissolved H_2 or Total Gas	0 to 2000 cc(STP)/kg	50 to 2000 cc/kg ± 20%[b,c] <50 cc/kg ± 5.0 cc/kg
Dissolved O_2	0 to 20 ppm	0.5 to 20.0 ppm ± 10%,[d] <0.5 ppm ± 0.05 ppm
Containment Atmosphere		
Hydrogen Content	0 to 10 Volume. % 0 to 30 Volume %, inerted containment	References R.G. 1.97 requirements
Oxygen Content	0 to 30 Volume. %	Analysis not required
Gamma Spectrum	Isotopic Analysis	Accurate within a factor of 2
Radiation Monitors		
Primary Containment	1 R/hour to 10^7 R/hour, Rev. 2, 3	1 R/hour to 10^7 R/hour, gamma only or 1 rad/hour to 10^8 Rad/hour, $\gamma + \beta$[e]
Secondary Containment		
Mark 1 and 2	0.1 R/hour to 10^4 R/hour	Not Required

Table 3-2 (continued)
COMPARISON OF RG 1.97 AND NUREG-0737 ANALYSIS REQUIREMENTS

NOTES:

1. The guideline upper limit on boron concentration is so much higher than the Regulatory Guide 1.97 requirement, it is suspected that the guideline concentration is in terms of ppm boric acid (6000 ppm boric acid corresponds to 1050 ppm boron).

2. An accuracy of ±10% is desirable between 50 and 2000 cc/kg, but ±20% can be acceptable.

3. In 1984, the NRC apparently agreed [Ref. 16] with GE's position [Ref. 15] that for total dissolved gas or hydrogen analysis an accuracy of ±50% at 25 to 50 cc/kg and ±30% above 50 cc/kg was adequate. Presumably, 25 cc/kg is the minimum required level of measurement.

4. In 1984, the NRC agreed [Ref. 16] with GE's position [Ref. 15] that oxygen with an accuracy of at least ±30% at 8-20 ppm and ±60% at 4-8 ppm was acceptable. Presumably, 4 ppm is the minimum required level of measurement.

5. NUREG-0737 originally required 10^8 rads/hour, but stated a beta monitor that would withstand the primary containment environmental conditions only was unavailable and a 10^7 R/hour gamma monitor would be acceptable.

4.0 JUSTIFICATION FOR ELIMINATION OF REQUIREMENTS FOR POST-ACCIDENT SAMPLING SYSTEM

As was the case for PWR Owners' Groups, the BWROG has determined that in-plant instruments and the associated methodology are as good as, if not better than, the PASS for collecting and assimilating information needed to assess core damage following an accident. In addition, BWR emergency and severe accident response strategies are based on use of available instrumentation and do not require use of the PASS. Based on current emergency and severe accident response strategies and guidelines, it has been determined that the PASS provides no benefit to the plant staff in assessing an unfolding event. The BWROG is in the process of replacing the existing BWR Core Damage Assessment Guideline (CDAG) to rely exclusively on in-plant instruments and known fuel release characteristics. Following implementation of the revised CDAG in early 2001, use of PASS information will no longer be recommended to assess the radiological impact of unfolding events.

The BWROG has evaluated each PASS sampling and analysis requirement on a sample-by-sample basis. Recommendations to delete PASS requirements are summarized along with non-PASS alternatives that can be employed. The general philosophy for elimination of PASS sampling from the plant design basis is that (1) these samples are not required in the EOP/SAG decision-making process, (2) the emergency preparedness requirements of 10CFR50.47 may be adequately met without using the PASS, (3) while the PASS results in a more direct measurement, it does not necessarily result in improved prediction capability, (4) reliance on the PASS can result in poor emergency planning decisions and unnecessary radiation exposure, and (5) the PASS requires significant plant resources, and decisions to take PASS samples may conflict with more pressing needs associated with accident mitigation.

The following provides an itemized discussion of the NUREG-0737 and related PASS requirements and a technical basis for their elimination. The discussion for the reactor coolant sample analysis capabilities is also applicable for suppression pool samples.

(1) Reactor Coolant Dissolved Gases and Reactor Coolant Hydrogen

Purpose:

NUREG-0737 requires the determination of either total dissolved gas or dissolved hydrogen in the reactor coolant. The primary purpose is to identify the potential for void formation in the vessel head from dissolved gases upon depressurization and to determine the contribution to the total hydrogen generated from the metal-water reaction with Zircaloy. A secondary benefit is to evaluate the coolant corrosion potential.

Recommendation:

Delete requirement.

Justification:

BWR reactor water level systems provide continuous indication of reactor water level, which will serve to demonstrate that non-condensable gas does not accumulate in the head of the reactor pressure vessel. In addition, the BWR vessel depressurization process (via postulated pipe break or safety-relief valve operation) will flush the primary system of dissolved gases [Ref. 15]. For BWRs, greater than 95% of the hydrogen generated is rapidly transported to the containment regardless of pressure [Ref. 15]. The containment hydrogen/oxygen monitors are employed to estimate core degradation from the reaction of water with the Zircaloy. NUREG/CR-4330 [Ref. 9] suggests that this requirement could be eliminated under these conditions.

Knowledge of the total dissolved gas concentration and hydrogen gas concentration can be employed to infer coolant corrosion potential (oxygen concentration), but this analysis is not necessary to estimate core degradation or for the mitigation of severe accidents.

(2) Reactor Coolant Oxygen

Purpose:

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

Recommendation:

Delete Requirement.

Justification:

There are no accident mitigation or emergency planning functions that require identification of the reactor coolant oxygen content. The requirement for reactor coolant oxygen sampling and analysis is tied to assessing the potential for chloride-induced stress corrosion cracking of stainless steel piping and components. For the high-pressure conditions that exist prior to vessel depressurization, a bounding analysis shows that the maximum oxygen concentration will be less that 0.4 ppm [Ref. 15]. For low pressure conditions, the oxygen concentration can be determined from the containment oxygen concentration. In 1984, the NRC agreed to delete requirements for reactor coolant dissolved oxygen at BWRs.

(3) Reactor Coolant Chlorides

Purpose:

The purpose of sampling for chlorides is to assure that chloride-induced Intergranular Stress Corrosion Cracking (IGSCC) of stainless steel piping and components will not occur in the long term.

Recommendation:

Delete requirement.

Justification:

There are no accident mitigation or emergency planning functions that require identification of the reactor coolant chloride content. The NRC has recognized that the potential for high concentrations of chlorides in the Reactor Coolant System is a strong function of the plant design and the water source for the ultimate heat sink. This is evidenced by the requirements in NUREG-0737 for the time at which the first sample for chlorides must be taken. For fresh water plants and plants with brackish or salt water with more than one barrier between the containment and the ultimate sink, the initial chloride sample is not required for 96 hours (4 days). For brackish and salt water plants with only one barrier between the potential source of chlorides and the containment, the first chloride sample is required in 24 hours.

Stress corrosion cracking (SCC) is a function of temperature, chloride, oxygen concentration, pH, and stress. Because there is no removal mechanism for chloride in post-accident scenarios, the chloride concentration can be bounded by conductivity readings early in the accident, in conjunction with knowledge of the plant's cooling water source impurities and evaluation of critical accident parameters.

(4) Reactor Coolant pH

Purpose:

The purpose of sampling the reactor coolant for pH is to assure that chloride-induced SCC of stainless steel piping and components will not occur in the long term and to assure that radioiodine species are retained in the water. Requirements to measure pH are included in RG 1.97 and the NUREG-0737 post-implementation guidelines.

Recommendation:

Delete requirements.

Justification:

For BWRs, it has been demonstrated that the reactor water and suppression pool pH will remain above 7.0 following loss-of-coolant accidents (LOCAs), which assures that the iodine will be retained in the coolant (see Appendix). Under these conditions, there is no benefit to

monitor this parameter. Note that for BWR large and intermediate break LOCAs, the composition of the reactor coolant and suppression pool water would be expected to be essentially the same. It is also noted that the generic BWR Severe Accident Guidelines require injection of the Standby Liquid Control System (SLCS) sodium pentaborate solution upon entry into the guideline. The addition of this solution acts as a buffer to further assure that the pH will remain basic and preclude iodine re-evolution.

(5) Reactor Coolant Boron

Purpose:

For BWRs, the normal method to verify shutdown margin is to assure that all rods are fully inserted. Boron is not employed in the reactor coolant of BWRs for normal reactivity control, as is the case for PWRs. Requirements to measure boron are included in RG 1.97 and NUREG-0737.

Recommendation:

Delete requirement.

Justification:

For BWRs, a concentrated boron solution can be injected into the reactor pressure vessel should the control rods fail to insert. The plant licensing basis does not, however, require this Anticipated Transient Without Scram (ATWS) event to be coupled to an event where major fuel degradation occurs. Also, for large LOCA events, the boron would be rapidly transported to the suppression pool.

ATWS with SLCS injection of boron may result in minor cladding damage, and the PASS could be employed to determine the concentration of boron in the reactor coolant. However, under these conditions (minor fuel clad damage), there are other methods to determine the effectiveness/concentration of the injected boron. These include:

- SLCS tank level
- Neutron Monitoring System
- Sampling of reactor coolant from normal reactor building sample sink

6) Reactor Coolant Conductivity

Purpose:

Conductivity measurements typically are used to confirm other analyses. For example, ionic species (e.g., boron, chloride) contribute to solution conductivity. If an imbalance exists between the measured conductivity and that expected based on the concentration of the analyzed species, it indicates either an error in the analyses or the presence of additional ionic

species which were not included in the analysis matrix. There is no requirement in NUREG-0737 or RG 1.97, Rev. 2, to measure the conductivity of the reactor coolant.

Recommendation:

Delete analysis.

Justification:

There are no accident mitigation or emergency planning functions that require knowledge of the RCS conductivity.

(7) Reactor Coolant Radioisotopes

Purpose:

The purpose of sampling the reactor coolant for radioisotopes is to provide data as input to the existing CDAG. Applicable requirements are included in RG 1.97 and NUREG-0737.

Recommendation:

Delete analysis.

Justification:

The revised BWROG CDAG will provide alternate methods using in-plant instrumentation to assess core damage (principally water level history, hydrogen concentrations, and containment radiation levels).

For badly damaged core conditions, there is very little value for radioisotopic assessments. Since significant core uncovery has occurred, significant quantities of radioisotopes would have left the Reactor Coolant System (RCS) or would have plated out in regions away from the sample point and would not necessarily be dissolved in the RCS sampled coolant. This level of core damage may be adequately assessed via evaluation of containment hydrogen and containment radiation levels along with event-specific information available from plant-related SAG observations. For reactor cores with minimal core damage, the existing reactor building sample stations can be employed to obtain samples.

Eliminating the sampling of RCS radioisotopes will not impact the ability of the plant to manage the accident or effect appropriate emergency response. With regard to the EOPs, isotopic analysis is not required prior to entering shutdown cooling. The assessment of core damage may be accomplished via methods that do not rely on the reactor coolant radioisotopic analysis. The existence of Core Damage Assessment Guidelines that are sufficient to make appropriate operational decisions in the absence of such an analysis supports the deletion of this requirement.

(8) Containment Atmosphere Hydrogen

Purpose:

The purpose of sampling the containment atmosphere for hydrogen concentration is to provide a means of assessing core damage and to monitor the potential formation of a combustible atmosphere in containment. Analysis of containment hydrogen concentration is a requirement of NUREG-0737 and RG 1.97, Revisions 2 and 3.

Recommendation:

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

Justification:

NUREG-0737 requires the capability to quantify the hydrogen levels in containment. Also, if in-line monitoring is employed to meet this PASS requirement, there is an additional requirement of having the capability of obtaining a backup grab sample.

Containment hydrogen is best determined through the use of the in-line hydrogen monitors installed in BWR plants. These in-line monitors satisfy the requirements of RG 1.97 and provide real-time data that can assist the operators in assessing core damage long before a grab sample could be obtained and analyzed. These monitors provide an indication of the potential hydrogen combustion threat in the presence of oxygen. The redundant trains of these monitors obviate the need for a backup grab sample and the intent of the PASS requirement is met.

(9) Containment Atmosphere Oxygen

Purpose:

The purpose of sampling the containment atmosphere for oxygen concentration is to provide an indication, along with the hydrogen analysis, of the potential for the formation of a combustible atmosphere in containment. There is no requirement for PASS capability to measure the containment oxygen concentration in NUREG-0737; however, this analysis is a requirement of RG 1.97.

Recommendation:

Maintain capability to monitor oxygen in the containment atmosphere. Delete PASS sampling requirement.

Justification:

Containment oxygen is best determined through the use of the current in-line oxygen monitors installed in BWR plants. These redundant in-line monitors satisfy the requirements of RG 1.97 and provide real-time data that can assist the operators in assessing core damage

long before a grab sample could be obtained and analyzed. These monitors provide indication of the potential oxygen combustion threat in the presence of hydrogen.

(10) Containment Airborne Radioisotopic Samples

Purpose:

The purpose of sampling the containment for radionuclide content is to enable offsite dose assessments to be made for post accident containment leakage and failure conditions. The current BWR Core Damage Assessment Guidelines employ these radionuclide concentrations to estimate the extent of core damage. These analyses are required by RG 1.97 and NUREG-0737.

Recommendation:

Delete requirements.

Justification:

When NUREG-0737 was originally drafted in the early 1980s, it was believed that the most accurate assessment of offsite dose would result from using the containment airborne radionuclide estimates obtained from sample analysis. However, considering the behavior of fission products, it is now apparent that the sample results would not be very accurate. In the early 1980s it was believed that the airborne iodine will be primarily volatile elemental iodine gas or organic iodine compounds.

Following extensive government funded research over the past two decades, the expectation of the source term constituents changed from primarily volatile to overwhelmingly particulate [Ref. 10]. This has an important implication in the ability of a remote system to accurately assess airborne iodine concentrations. For example, for core damage accidents, a significant portion of the volatile and non-volatile fission products would be expected to deposit on RCS internal surfaces and would not be released to the containment. Therefore, the assessment of core damage based on the containment radionuclides could be severely distorted. In addition, severe accident analyses have found that when the containment is depressurized (as in a containment pressure boundary failure or an intentional release through a containment vent), a significant fraction of the fission products previously deposited on internal surfaces of the RCS 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.

Information provided with respect to the containment gas samples is also suspect due to issues associated with (1) obtaining data from a "truly representative" sample point, (2) plateout of cesium iodines (CsI) in the sample lines, and (3) time delays associated with

obtaining, processing and interpreting the sample during non-stable phases of the accident. Item 3 refers to issues associated with transient generation of the fission products within the core and its subsequent redistribution, as well as the impact of containment conditions (spraying, operation of fan coolers, containment leaks, etc.) on the fission product inventory.

In the case of potential containment failure or containment venting, the use of containment atmosphere samples would underpredict the actual releases as the containment pressure is reduced, due to re-evolution of aerosol fission products from surfaces within the containment, as well as transport of fission products in the RCS. Severe accident analyses, such as those summarized in the EPRI Severe Accident Management Guidance Technical Basis Report, show that the aerosol fission product inventory in the containment increases when the containment is depressurized [Ref. 18].

The revised BWROG Core Damage Assessment Guideline relies exclusively on in-plant instrumentation and known fuel release characteristics to assess core damage, and containment radiation is a key input to this evaluation. It is recommended that the radioisotopic sampling capability provided by the PASS be replaced by I-131 site survey detection capability. Site survey capability is applicable to all accident scenarios and release points and provides a realistic means for dose projections. The net impact of this change will be a positive enhancement to plant safety for BWRs that currently do not have this capability.

Partial relief of this requirement was previously granted by the NRC by allowing deletion of the heat tracing requirement on the sample lines if radioiodines are not used for core damage assessment. Since alternative means exist for the assessment of core damage that do not rely on containment radionuclide analysis, this requirement may be deleted.

Note that while heat tracing may impact the revolatilization of the deposited elemental iodines, it will not affect the deposition of particulate CsI (the expected dominant chemical form). This compound has a vaporization temperature of approximately 1280°F.

5.0 POST ACCIDENT SAMPLING SYSTEM ANNUAL COST BURDEN

The BWROG surveyed the participating utilities to determine the cost burden of maintaining BWR Post-Accident Sampling Systems. These costs vary significantly based on the type of PASS system employed. For a typical BWR with a standard GE PASS system, the approximate cost burdens are as follows:

1. Corrective and preventative maintenance	$5000
2. Personnel training and emergency planning drills	$12000
3. Surveillance testing	$10000
4. Off-site sample analysis facility	$12000
Total	**$39000**
Total BWR fleet cost based on net present value analysis for 34 plants over 20 years	**$13,000,000**

For BWRs that have PASS systems which contain on-line analysis capabilities, the annual cost burdens are several times higher than the above stated typical cost burdens.

6.0 CONCLUSIONS AND RECOMMENDATIONS

The results of a BWROG evaluation confirm that BWR Post-Accident Sampling Stations (PASS) do not provide the benefits expected by the NRC when the requirements were imposed 20 years ago following the Three Mile Island Unit 2 accident. All BWR emergency and severe accident response strategies can be implemented using in-plant instrumentation without use of PASS. Because of the need to assess the core status in a timely manner, utilities have developed accident assessment procedures that do not rely on taking samples. In-plant instrumentation and the associated analysis methods based on known fuel release characteristics will provide the timely information required to assess core damage that is needed to provide guidance to the plant staff for the mitigation of severe accidents. This information is required early in the accident scenario, and the information derived from in-plant instrumentation and known fuel release characteristics is as good as or better than information currently provided by PASS several hours after event initiation. PWR Owners' Groups have previously documented similar findings.

The BWROG has carefully evaluated PASS regulatory requirements on an individual sample analysis basis. The BWROG has determined that (1) these samples are not required in the EOP/SAMG decision-making process, (2) the emergency preparedness requirement of 10CFR50.47 may be adequately met without using PASS, (3) while the PASS results in a more direct measurement, it does not necessarily result in improved predictive capability, (4) reliance on PASS can result in poor emergency planning decisions and unnecessary radiation exposure, and (5) PASS requires significant plant resources; and decisions to take PASS samples may conflict with more pressing needs associated with accident mitigation.

The BWROG has also determined that the annual cost burden for a typical BWR Post-Accident Sampling System is approximately $40,000 and that the safety benefits of this equipment do not justify the required expenditure. This represents a BWR fleet cost of $13,000,000, based on a net present value analysis for 34 plants over 20 years with an 8% discount rate. For BWRs that have on-line analysis capabilities, the annual cost burdens are several times higher.

The BWROG has also considered the effect of removing PASS from a safety risk perspective. None of the BWR Probabilistic Risk Assessments (PRAs) relies upon or addresses PASS and, therefore, quantitative risk assessments cannot be made. The risk insights based on review of normal operating, emergency, and severe accident conditions indicate that the existence or non-existence of PASS would have no effect on core damage or large early release frequencies (LERF). The BWROG has therefore concluded that PASS can be removed without significantly affecting plant safety and, thus, recommends that all PASS

regulatory requirements be eliminated. The BWROG also recommends that the participating utilities take the following actions prior to removal of PASS:

- Implement an I-131 site survey detection capability that will provide an alternate means for dose projections. It is noted that most BWRs currently have this capability.
- Evaluate and revise (if required) plant-specific EOPs and SAG to assure conformance to the revised BWR Core Damage Assessment Guideline (CDAG) (available First Quarter 2001). This revised CDAG relies exclusively on in-plant instrumentation to provide assessment of fuel damage.
- Develop contingency plans for obtaining and analyzing highly radioactive reactor coolant, suppression pool, and containment atmospheric samples.

7.0 REFERENCES

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APPENDIX A
POST-ACCIDENT PH AFFECT ON IODINE PARTITIONING

A 1. BACKGROUND

Section 5.2 of NUREG-1465 (Reference 1) reports that the re-evolution of iodine can impact the plant radiological analyses if the suppression pool pH is below a value of 7.0. Specifically, for those BWRs that credit the long-term retention of iodine in the suppression pool via sprays or pool scrubbing, NUREG-1465 suggests that the maintenance of a pH at or above a level of 7.0 should be demonstrated. Since BWRs generally do not have a requirement to control post-accident suppression pool pH, evaluation of pool pH transient is needed to demonstrate that the level of pH is 7.0 or above to preclude re-evolution of elemental iodine.

This appendix briefly describes the principal acidic and basic materials in containment that affect the post-accident pH level in the BWR suppression pool. Results of evaluation performed for the Grand Gulf and Perry plants are also summarized.

A 2. POST-ACCIDENT POOL PH LEVEL

Suppression pool pH is determined by the relative concentration of H^+ and OH^- ions in the pool. NUREG/CR-5950 [Ref. 2] includes a general discussion of acids and bases in containment. The principal contributors to the post-accident production of acids for transport to the suppression pool following loss-of-coolant accidents are hydriodic acid (HI), nitric acid (HNO_3), and hydrochloric acid (HCl). The most important post-accident production of basic material for transport to the suppression pool is cesium hydroxide (CsOH). In addition, chemical additives can also be used as pH buffers to control the pH level in the suppression pool. Injection of the SLCS sodium pentaborate solution can provide this function to further assure that the pH remains basic.

A 2.1 Acidic Materials in Containment

A 2.1.1 Hydriodic Acid Production

Table 3.12 of NUREG-1465 indicates that 5% of the core iodine inventory is discharged during the gap release phase, while an additional 25% is discharged during the early in-vessel phase. The iodine existing in the Reactor Coolant System post-accident, consistent with Section 5.4 of NUREG-1465, will be composed of no more than 5% I and HI. Therefore, hydriodic acid (HI) generation can be conservatively assumed to be 5% of the post-accident iodine release.

A 2.1.2 Nitric Acid Production

Nitric acid (HNO_3) is produced by the irradiation of water and air. As reported in NUREG/CR-5950, the experimental result of radiation G value for nitric acid formation based on radiation absorption by water is 0.007 molecules/100 eV. This is equivalent to 7.3E-6 mol HNO_3/L per Mrad of radiation dose in the suppression pool.

A 2.1.3 Hydrochloric Acid Production

The radiolysis of chlorine-bearing electrical cable insulation and jackets will result in the production of hydrochloric acid (HCl) vapor. The predominant contributor is jackets made of Hypalon, a chlorosulfonated polyethylene. As reported in NUREG/CR-5950, the experimental result of radiation G-value (molecules/100 eV) for hydrochloric acid formation based on radiation absorption by Hypalon is 2.115 molecules/100 eV. This is equivalent to 2.192E-6 mol HCl/g per Mrad of radiation dose. Both the gamma and beta dose in the drywell and containment (wetwell) should be considered in the calculation of hydrochloric acid production. The radiation dose for the cable jacket is the product of (1) drywell/wetwell dose, (2) absorption fraction in the jacket, and (3) the ratio of average radiation flux in the jacket to the incident flux. For the gamma radiation, the flux ratio is 1.0 and the absorption fraction is about 0.12 for bounding ½-inch insulation. For the beta radiation, the short range of the beta particles makes the absorption fraction nearly 1.0, while the flux ratio for a typical 45-mil jacket is about 0.18.

A 2.2 Basic Materials in Containment

A 2.2.1 Cesium Hydroxide Production

Table 3.12 in NUREG-1465 indicates that 5% of the core cesium inventory is discharged during the gap release phase, while an additional 20% is discharged during the early in-vessel phase. Table 3.12 in NUREG-1465 also indicates that 5% of the core iodine inventory is discharged during the gap release phase, while an additional 25% is discharged during the early in-vessel phase. The iodine existing in the Reactor Coolant System (RCS), consistent with Section 5.4 of NUREG-1465, will be composed of at least 95% cesium iodide (CsI). The cesium that is not in the chemical form of CsI is assumed to exit the RCS in the form of cesium hydroxide (CsOH) and this material will be deposited into the suppression pool. This is the most important basic material governing post-accident pH that is introduced into the suppression pool.

A 2.2.2 Chemical Additives for pH Control

Chemical additives can be used as pH buffers to control the post-accident pH level in the suppression pool following loss-of-coolant accidents. The measure of pH buffer capacity, defined as the increment of strong base per change in pH due to increment of the base, is related to the equilibrium constant for dissociation of the weak acid or weak base. As

reported in NUREG/CR-5950, the pH buffer materials for use in containment must be borate or phosphate because other potential chemical additives would not have the desirable pH range and chemical stability. Injection of the BWR SLCS sodium pentaborate solution can provide this function to further assure that the pH remains basic.

A 3. GRAND GULF EVALUATION

Suppression pool pH analyses were performed at Grand Gulf Nuclear Station with the methodology documented in Reference 3 and a detailed analysis utilizing this methodology as documented in Reference 4. The following assumptions were used in the analysis:

- Initial suppression pool pH is conservatively assumed to be 5.3.
- Total pool volume is 4.841E6 liters.
- Core inventory is 2400 moles cesium and 325 moles iodine, including the stable Cs-133 and I-127.
- Total mass of exposed cable jacket and insulation in the drywell is conservatively estimated to be 874 lb with free air drop (i.e., not routed in cable trays and fully exposed to beta radiation) and 874 lb routed in trays. For the containment, the mass is 1,561 lb and 14,049 lb, respectively.
- Table A-1 shows the results of the Grand Gulf analysis. Both cesium hydroxide and hydriodic acid productions follow the post-accident cesium and iodine release profile. That is, gap release phase is initiated 121 seconds into the accident for a duration of 30 minutes, followed by a 90-minute early in-vessel release phase. Therefore, both the cesium hydroxide and the hydriodic acid concentrations stay at a constant level after 2.03 hours. This also means that the pool pH decreases after 2.03 hours, as no additional cesium hydroxide is produced. The results in Table A-1 clearly indicate that the post-accident pH level in the suppression pool stays above 7.0.

Table A-1
Grand Gulf Post-Accident Pool pH and Acid/Base Concentrations (moles/l)

Time (hr)	HI	HNO_3	Drywell HCl	Containment HCl	Total $[H^+]$	CsOH	pH
0.0336	0	0	0	0	5.01E-6	0	5.3
0.5336	1.68E-7	0	8.83E-7	8.51E-7	6.91E-6	5.15E-5	9.2
2.0336	1.01E-6	6.60E-7	2.36E-6	2.60E-6	1.16E-5	1.05E-4	10.0
24	1.01E-6	5.73E-6	6.73E-6	8.58E-6	2.71E-5	1.05E-4	9.9
720	1.01E-6	4.25E-5	2.02E-5	2.90E-5	9.77E-5	1.05E-4	8.9

A 4. PERRY EVALUATION

In the license amendment submittal to NRC for the Perry Nuclear Power Plant, First Energy has proposed to use the existing Standby Liquid Control System (SLCS) for controlling and maintaining long-term suppression pool water pH levels at 7.0 or above following the postulated Design Basis Accident (DBA). The SLCS is a safety-related system and designed as a Seismic Category 1 system. The system is manually initiated from the main control room to pump a boron neutron absorber solution into the reactor. The SLCS contains 5236 pounds of sodium pentaborate, which acts as a pH buffer.

The NRC has performed an evaluation of post-accident suppression pool pH for Perry to verify the licensee's conclusion [Ref. 5]. The Perry analysis considered the following factors:

- The addition of sodium pentaborate into the pool.
- Hydrochloric acid generated from the electrical cable degradation.
- Cesium hydroxide formed from the fission products released from the core.
- Nitric acid produced by irradiation of water and air in the containment.

Table A-2 shows the results of the evaluation. Both the cesium hydroxide and the hydriodic acid amounts stay at a constant level after 2 hours, which is the end of the gap release and early in-vessel release. The pool pH level peaks at a value of 8.63, 2 hours into the accident,

and then gradually decreases to a value of 8.48 after 30 days. The results in Table A-2 clearly indicate that the post-accident pH level in the suppression pool stays above 7.0.

Table A-2
Perry Post-Accident Pool pH and Acid/Base Amounts (Moles)

Time (hr)	HI	HNO_3	HCl	CsOH	pH
1	2.0	2.2	8.4	407	8.61
2	4.6	8.3	39	868	8.63
24	4.6	91	425	868	8.60
720	4.6	503	1745	868	8.48

A 5. CONCLUSIONS

For the post-accident suppression pool pH level, the two most significant contributors are the cesium hydroxide released from the core cesium inventory and the hydrochloric acid generated from the radiolytic decomposition of chlorine-bearing electrical cable insulation/jacket. Thus, prediction of pool pH level is highly dependent upon good estimates of these two compounds.

Results of suppression pool pH evaluations performed for the Grand Gulf and Perry plants conclude that the post-accident pool pH stays above a value of 7.0, with (for Perry) or without (for Grand Gulf) the use of the SLCS to inject pH buffers. Thus, re-evolution of elemental iodine from the suppression pool is minimal. This conclusion does not necessarily apply directly to other BWRs without evaluating plant-specific variations in pool size, amount of chlorine-bearing electrical cable insulation/jacket, and other factors that affect the generation of acids/bases in the pool. It should be noted, however, that a similar analysis performed for a typical BWR/4 to support a power uprate analysis also confirmed that the suppression pool pH will remain above 7.0 for the duration of the accident. This analysis is consistent with the DBA source terms (up to and including the early in-vessel release phase) as specified in NUREG-1465 and does not reach any conclusions for severe accident releases where extensive ex-vessel releases associated with core-concrete interactions are considered.

A 6. REFERENCES

1. NUREG-1465, Accident Source Terms for Light-Water Nuclear Power Plants, February 1995.

2. NUREG/CR-5950, *Iodine Evolution and pH Control*, November 1992.

3. Grand Gulf Nuclear Station Engineering Report GGNS-98-0039, Rev. 1, *Suppression Pool pH and Iodine Re-evolution Methodology*, dated 05/06/99.

4. Grand Gulf Nuclear Station Calculation XC-Q1111-98013, *Suppression Pool pH Analysis*, dated 02/19/99.

5. U.S. NRC Safety Evaluation Related to Amendment No. 103 to Facility Operating License No. NPF-58, FirstEnergy Nuclear Operating Company, Perry Nuclear Power Plant Unit 1, Docket No. 50-440, March 26, 1999.

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