ENCLOSURE 1

WCNOC PROPOSED POST ACCIDENT SAMPLING SYSTEM (PASS) FUNCTIONS REDUCTION

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I. OBJECTIVE:

The Post Accident Sampling System (PASS) installed at Wolf Creek Generating Station (WCGS) is designed to perform a multitude of sampling and analysis functions. These functions were designed and intended to be used in post accident scenarios and were put into place after the Three Mile Island Unit 2 (TMI-2) events. Those events were the catalyst for several regulatory guidance documents which culminated in the issuance of NUREG-0737, "Clarification of TMI Action Plan Requirements." However, in the years since NUREG-0737 was issued, a considerable amount of knowledge and operating experience has been gathered about core behavior and the role that a PASS would play in the different accident scenarios. From this knowledge and experience, a proposed revision to the PASS equipment and functions is being made. This revision, used in conjunction with Westinghouse WCAP-14696, "Westinghouse Owners Group Core Damage Assessment Guidance," would significantly change the functions for the system. The required samples and analysis would be reduced to a single sample and analysis point (i.e. RCS boron). All other samples and analyses would be eliminated based on their redundancy to other installed equipment and/or their not being needed in the mitigation of the various accident scenarios.

II. EXECUTIVE SUMMARY:

The post accident sampling functions discussed in WCAP-14986, "Westinghouse Owners Group Post Accident Sampling System Requirements: A Technical Basis," have been thoroughly reviewed for applicability and suitability at WCGS. The proposed revisions to the PASS functions would clearly benefit and enhance the safe and reliable operation of the facility. The revised functions would allow Wolf Creek Nuclear Operating Corporation (WCNOC) personnel to devote their attention and available resources to samples and analyses which would benefit personnel involved in accident analysis and mitigation. In no way does the reduction in samples and analyses jeopardize the health and safety of the public. The revised sample and analyses reflect the knowledge gained in the nuclear industry since the initial functions were promulgated in the aftermath of TMI-2.

A complete evaluation and justification for each PASS sample and analysis currently in use at WCGS is contained within the body of this report. This evaluation draws heavily upon the information contained within WCAP-14986 as well as the experience and knowledge contained within WCNOC.

III. CURRENT PASS SAMPLES AND ANALYSIS:

WCNOC currently follows the guidance stated in NUREG-0737 and Regulatory Guide 1.97, Rev. 2, "Instrumentation For Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Detailed below are the regulatory guidance and the method in which WCNOC currently fulfills the guidance.

A. REGULATORY GUIDANCE:

The guidance for a PASS is contained within NUREG-0737 and Regulatory Guide 1.97 Rev. 2. Detailed below are the applicable sections of the two regulatory documents:

NUREG-0737 Section II.B.3:

1. The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should

be 3 hours or less from the time a decision is made to take a sample.

2. The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the 3-hour time frame established above, quantification of the following:

   a) Certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases, iodine's and cesium's, and non-volatile isotopes)

   b) Hydrogen levels in the containment atmosphere

   c) Dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids

   d) Alternatively, have inline monitoring capabilities to perform all or part of the above analyses

3. Reactor coolant and containment atmosphere sampling during post accident conditions shall not require an isolated auxiliary system [e.g., the letdown system, reactor water cleanup system (RWCUS)] to be placed in operation in order to use the sampling system.

4. Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

5. The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions, the licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

6. The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 Rem whole body, 75 Rem extremities). [Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979 letter from H. R. Denton to all licensees).

7. The analysis of primary coolant samples for boron is required for PWR's. (Note that Revision 2 of Regulatory Guide 1.97, when issued, will likely specify the need for primary coolant boron analysis capability at BWR plants.)

8. If inline monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable.

Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident and at least one sample per week until the accident condition no longer exists.

9. The licensee's radiological and chemical sample analysis capability shall include provisions to:

   a) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guide 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 μCi/g to 10 Ci/g.

   b) Restrict background levels of radiation in the radiological and chemical analysis facility from sources, such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of ventilation system design which will control the presence of airborne radioactivity.

10. Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe the radiological and chemical status of the reactor coolant systems.

11. In the design of the post accident sampling and analysis capability, consideration should be given to the following items:

   a) 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, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post-accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

   b) The ventilation exhaust from the sampling station should be filtered with charcoal absorbers and high-efficiency particulate air (HEPA) filters.

Regulatory Guide 1.97, Rev. 2:

Table 2 of Regulatory Guide 1.97, Rev. 2 requires that the following accident sampling and analysis be maintained on site.

Primary Coolant and Sump:

- Gross Activity

- Gamma Spectrum
- Boron Content
- Chloride Content
- Dissolved Hydrogen or Total Gas
- Dissolved Oxygen
- pH

Containment Air:

- Hydrogen Content
- Oxygen Content
- Gamma Spectrum

For all of the samples identified above, the Regulatory Guide requires that the samples be taken and analyzed within 3 hours of the decision is made to sample, with the exception of chlorides which is required within 24 hours.

B. **WCNOC RESPONSE TO THE REGULATORY GUIDANCE:**

Detailed below are the applicable sections of the USAR which describe WCNOC's response to the guidance. USAR Appendix 3A and Table 7A are applicable to Regulatory Guide 1.97, Rev. 2, while USAR Section 18.2.3.2 applies to NUREG-0737. In addition, Technical Specification 6.8.4.d contains requirements pertaining to PASS.

- USAR Section 3A:

  REGULATORY GUIDE 1.97 REVISION 2 DATED 12/80

  Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident

  DISCUSSION:

  The recommendations of this regulatory guide are discussed in Appendix 7A.

- USAR Appendix 7A:

REGULATORY GUIDE 1.97 TABLE 2 RECOMMENDATIONS				
VARIABLE IDENT. NO	VARIABLE	RANGE	*CATEGORY	PURPOSE
E.6.1	Primary Coolant	Grab Sample	$3^{5,10}$	Release assessment, verification analysis
E.6.1.1	Gross Activity	10 µCi/ml to 10 Ci/ml		
E.6.1.2	Gamma Spectrum	(Isotopic Analysis)		
E.6.1.3	Boron Content	0 to 6,000 ppm		
E.6.1.4	Chloride Content	0 to 20 ppm		

REGULATORY GUIDE 1.97 TABLE 2 RECOMMENDATIONS				
VARIABLE IDENT. NO	VARIABLE	RANGE	❶CATEGORY	PURPOSE
E.6.1.5	Dissolved Hydrogen or Total Gas	0 to 2,000 cc(STP)/kg		
E.6.1.6	Dissolved Oxygen	0 to 20 ppm		
E.6.1.7	pH	1 to 13		
B.1.3	RCS Soluble Boron Concentration	0 - 6,000 ppm	3	Verification
C.1.3	Analysis of Primary Coolant (Gamma Spectrum)	10 µCi/gm to 10 Ci/gm or TID-14844 source term in coolant volume	3[5]	Detail analysis, accomplishment of mitigation, verification, long-term surveillance
E.6.3	Containment Air	Grab Sample		Release assessment, verification analysis
E.6.3.2	Oxygen Content	0 to 30 percent		Release assessment, verification analysis
E.6.3.3	Gamma Spectrum	(Isotopic Analysis)		Release assessment, verification analysis
E.6.2	Sump	Grab sample	3[5,18]	Release assessment, verification analysis
E.6.2.1	Gross Activity	10 µCi/ml to 10 Ci/ml	3	
E.6.2.2	Gamma Spectrum	(isotopic analysis)	3	
E.6.2.3	Boron Content	0-6,000 ppm	3	
E.6.2.4	Chloride Content	0-20 ppm	3	
E.6.2.5	pH	1 to 13	3	
C.3.2	Containment Hydrogen Concentration	0 to 10% (capable of operating from 10 psia to maximum operating pressure)	1	Detection of potential for breach, accomplishment of mitigation, long term surveillance
E.6.3.1	Hydrogen Content	0 to 10%	3	Release assessment, verification analysis

Notes:

❶ The Category classifications (i.e. 1, 2 or 3) and additional notes (i.e. 5 or 18) are in accordance with Regulatory Guide 1.97, Rev. 2.

WCGS DESIGN PROVISIONS									
VARIABLE IDENT. NO.	VARIABLE	RANGE	SENSOR/TRANSMITTER		CONTROL ROOM				PLANT COMPUTER
					INDICATOR		RECORDER		
			IDENT. NO	CL. 1E	PANEL	CL. 1E	PANEL	CL. 1E	
E.6.1.1	Gross Activity	Refer to Section 18.2.3.2	SJ-145	N	SJ-082	N			PASS
E.6.1.2 E.6.3.3	Gamma Spectrum								
E.6.1.3	Boron Content								
E.6.3.2	Oxygen Content								
E.6.1.4	Chloride Content (2)		SJ-145	N	SJ-082	N	-	-	Y
E.6.1.5	Dissolved Hydrogen		SJ-145	N	SJ-082	N	-	-	Y
E.6.1.6	Dissolved Oxygen		SJ-145	N	SJ-082	N	-	-	Y
E.6.1.7	pH		SJ-145	N	SJ-082	N	-	-	Y

Notes:

1. The WCGS design includes an inline post accident sampling system which meets the stated guidance. Refer to Section 18.2.3.2 for details on the system design provisions. Samples are obtained from redundant sample points with Class 1E isolation valves for the containment atmosphere, the containment recirculation sumps, and the reactor coolant.
2. Chloride analysis is completed within 4 days of the sample being taken as discussed in NUREG-0737. Refer to Section 18.2.3.1. The analysis can be performed on site if dose rates allow, or by an off-site facility contracted to provide results within four days.

E.6.2	Sump Grab Sample Containment Recirculation	See data sheet 13.1							
	ECCS Pump Room Sumps	Not required							

	Auxiliary Building Sumps	Not required							

Notes:

1. The containment recirculation sumps are sampled by the inline sampling system described on data sheet 13.1 and in Section 18.2.3.2.
2. The ECCS pump room and auxiliary building sumps are provided with Class 1E level indication and operate as described in Section 9.3.3. Process and effluent monitors provide indication of any airborne activity in these sumps since they are directly vented to the auxiliary building normal exhaust system.
3. Sump sampling for the ECCS pump rooms and auxiliary building is considered unnecessary. The Class 1E level indication will detect any accumulated leakage, and the isolation valves will prevent its discharge from the auxiliary building. Should the leakage be from a line that contains fluid from the recirculation sump, the recirculation sump sample will provide the recommended analyses, since the fluid is from the same source.
4. The analysis can be performed on site if dose rates allow, or by an off-site facility contracted to provide results within four days.

C.3.2	Containment Hydrogen Concentration	0-10%	AT-10	Y	020	Y	020	Y	NPIS
E.6.3.1			AT-19	Y	020	Y	-	-	NPIS

Notes:

1. The hydrogen analyzers are described in Section 6.2.5 and shown on Figure 6.2.5-1
2. The hydrogen analyzers meet all of the stated guidance. Refer to Section 18.2.12.2 for a comparison with NUREG-0737 guidance. The analyzers will operate properly within the recommended containment pressure ranges.
3. The hydrogen concentration is a Type A variable and is used for initiating the Hydrogen Recombiners when hydrogen is detected. Should the need arise, the recombiners could be started following load sequencing operations should the core or primary systems indicate a potential for hydrogen generation rates above any current design bases.
4. As stated in Section 7A.3.2.d, diverse variables need only be performance grade and not Class 1E. The post accident sampling system will provide the capability to sample the containment atmosphere following an event. Refer to data sheet 13.1.

- USAR Section 18.2.3.2:

The WCGS design provides an in-line monitoring system. The radionuclide analysis may be performed by either the in-line analyzer contained within SJ-145 or by taking a grab sample and analyzing the sample on-site using other appropriate instruments. The system has the capability to sample from the reactor coolant system hot legs 1 and 3, containment recirculation sumps, and the containment atmosphere. The sampling locations in the containment atmosphere are the same as those for the containment hydrogen monitor, as described in (USAR) Section 6.2.5.2.2.3 and shown on (USAR) Figure 6.2.5-1.

The analyses to be performed by the system are listed in the table below.

TABLE II.B.3-1	
ANALYSES FOR THE POSTACCIDENT SAMPLING SYSTEM	
Liquids	**Ranges**
Radioisotopic identification	10^{-3} - 10^{7} µCi/cc
Boron	0 - 6,000 ppm
pH	1 - 13
Hydrogen	0 - 2,000 cc (STP)/kg
Oxygen	0 - 20 ppm
Chloride(1)	0 - 20 ppm
Conductivity	0.1 - 1,000 µ mhos
Gases	
Radioisotopic identification	10^{-7} - 10^{5} Ci/cc
Oxygen	0 - 50 wt %
Hydrogen(2)	0 - 10 Volume Percent

Notes:
(1) The chloride analysis can be performed on-site if dose rates allow, or by an off-site facility contracted to provide results within four days. The instrument used to perform the chloride analysis on site is not part of the Post Accident Sampling System (SJ-145).
(2) The containment hydrogen monitor is not part of the Post Accident Sampling System Cabinet (SJ-145).

The in-line monitoring system is normally isolated; however, it could be manually initiated and operated after an accident.

Due to the use of remote in-line monitoring equipment, personnel exposures are minimized. The expected dose is calculated to be between 450 and 1500 mRem for taking a liquid sample. This dose meets the guidance of NUREG-0737, Section II.B.3 and Regulatory Guide 1.97. Provisions have been included for providing both diluted and undiluted grab

samples of the reactor coolant, containment atmosphere, and the recirculation sump. The grab samples are shielded and the system is designed to minimize personnel exposure while obtaining grab samples, if they are needed.

The lines in the sample panel are flushed or purged after each analysis and all the sampled fluids are returned to the containment. Since the sample panel is located in the auxiliary building, any leakage from the system is filtered through the charcoal absorber and HEPA filters of the auxiliary building emergency exhaust system (see Section 9.4.3).

The system includes a sample control panel, CRT, and printer located in the counting room. A read-only printer is also located in the control room.

The nuclear sampling system P&ID (10466-M-12SJ04) is listed in Table 1.7-2.

Accessibility of the auxiliary building to obtain a grab sample was addressed in a detailed shielding study (See Section 18.2.2).

Additional details on the post accident sampling system design, and post-accident reactor core damage assessment, were submitted to the NRC by SNUPPS letters dated February 4, 1983, April 24, 1984 and June 21, 1984.

18.2.3.3 Conclusion

The post accident sampling system design for WCGS meets the recommendations of Item II.B.3 of NUREG-0737.

- Technical Specification 6.8.4.d:
  - Post-accident Sampling:
    - O A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodine's and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:
      - * Training of personnel,
      - * Procedures for sampling and analysis, and
      - * Provisions for maintenance of sampling and analysis equipment.

In addition, the requirements of the Technical Specifications are addressed by various programs within WCGS (e.g. CHS SJ-145 and AP 16C-001).

IV. PROPOSED PASS SAMPLES:

Detailed below in the following sections is a discussion of each sample and analysis that is performed by WCGS and the resulting logic for their elimination or retention.

A. Reactor Coolant Dissolved Gases & Reactor Coolant Hydrogen:

WCNOC submitted to the NRC a request (i.e. WCNOC letter WO 98-0047 dated May 11, 1998) for approval to delete the inline monitoring

and grab sample capability for RCS dissolved hydrogen from USAR Section 18.2.3. A complete evaluation and analysis was included as justification for the elimination of the capability to perform a dissolved total gas (e.g. hydrogen) sample and analysis.

The NRC response (WCNOC Incoming Letter No. 98-01418 dated September 4, 1998) indicated that prior approval of the NRC to delete the inline monitoring and grab sample capability for RCS dissolved hydrogen was not required, based on the result of the Unreviewed Safety Question Determination (USQD 59 98-0071) performed by WCNOC for this issue. Based on the NRC's response and the USQD evaluation, an USAR Change Request was initiated to delete the inline monitoring and grab sample capability for RCS dissolved hydrogen.

B. **Reactor Coolant Oxygen**:

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

There are no accident management or emergency planning functions that require identification of the reactor coolant oxygen content. The requirement for reactor coolant oxygen sampling and analysis is tied to preventing chloride induced stress corrosion cracking of stainless steel piping, 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.

Therefore, based on the discussion above, it can be concluded that there is no need to maintain a post accident system capability to sample and analyze the RCS oxygen.

C. **Reactor Coolant Chlorides**:

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

Chlorides can be introduced into the reactor coolant system in four different ways:

- recirculation of water from the containment sump (design basis emergency core cooling recirculation), or
- refilling the refueling water storage tank (RWST) with non-demineralized water (e.g., brackish river water) to continue injection of water into the reactor coolant system per EMG C-11, "Loss of Emergency Coolant Recirculation," or
- backfilling the RCS through a ruptured steam generator tube per EMG ES-31, "Post SGTR Cooldown Using Backfill," or
- SAM SAG-03, "Inject Into RCS."

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 containment sump water level. Such increases would 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 RCS via emergency core cooling system (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 break 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.

Since WCGS is located at a fresh water site, the potential concentration of chlorides that can be introduced into the reactor coolant system is generally quite low. Additionally, if the pH of the water in the RCS can be estimated and adjusted, there is no need to know the exact chloride concentrations in the water.

Therefore, based on the discussions above, no reasons exist for not deleting this post accident sampling function. For long term monitoring, capabilities will be retained (but not within the plant licensing basis) for obtaining undiluted and/or diluted reactor coolant samples.

D. **Reactor Coolant pH**:

The purpose of sampling the RCS 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 iodine's for design basis accidents or accidents in which emergency core cooling is operational in the recirculation mode.

There are no accident management or emergency planning functions that inquire about the reactor coolant pH. However, as described previously in the section on reactor coolant chlorides, there is a need 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 EMG (EMG C-11, "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 and reactor coolant chlorides, this post accident sampling function should be deleted. For long term monitoring, capabilities will be retained (but not within the plant licensing basis) for obtaining undiluted and/or diluted reactor coolant samples to assess the corrosiveness of the water.

E. **Reactor Coolant Boron**:

The purpose of sampling the RCS for boron is to assure that there is adequate shutdown margin of boron in the reactor coolant system to enable cold shutdown to be achieved.

Need for Reactor Coolant Boron PASS Capability:

The post accident sampling capability to measure the reactor coolant boron is a NUREG-0737 and a Regulatory Guide 1.97 function.

In addition, the capability to measure boron in the reactor coolant system supports the WCGS EMGs. A review of the EMGs has identified at least ten different EMG procedures where the control room personnel are directed to verify adequate shutdown margin exists. However, this is also performed without explicitly directing the personnel to obtain a RCS boron sample. In all of the procedures except one (EMG ES-02, "Reactor Trip Response"), 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. The EMGs provide a "fail-safe" for this condition (proceeding with subsequent EMG steps if boron sample analysis is not available) in EMG F-0, "Critical Safety Function Status Trees (CSFST)" CSF F-01, 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 EMG FR-S2, "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 EMG ES-02, "Reactor Trip Response." In this procedure, the inability to obtain a reactor coolant boron sample to verify adequate shutdown margin would result in stopping further recovery actions until adequate shutdown margin can be verified.

Based on the above assessments of the need for post accident sampling capabilities for reactor coolant boron, this post accident sampling function is not required to support mitigation of core damage accident conditions. However, this post accident sampling function does support the recovery planning for placing the plant in a long term safe, stable state following an accident. Therefore, this post accident capability should be retained.

1. Post Accident Sampling Timing and Accuracy:

Post accident sampling of reactor coolant for boron is not required for accident mitigation. Therefore, there is little need to take such a sample during the transient portion of an accident. Recognizing that the primary purpose of reactor coolant boron is to assure that the plant is in a long term safe, stable condition, the initial reactor coolant boron sample should be obtained as soon as possible after stable core and reactor coolant system conditions exist. In terms of a numerical requirement, this

should be specified to be within eight (8) hours after stable core conditions are achieved.

The existing required accuracy for the boron measurement is ± 5% of the measured value. However, since the RCS boron concentration required to assure a subcritical condition is generally greater than 1500 ppm, the greatest accuracy should be required only for boron concentrations above 1500 ppm. It is recommended that the minimum accuracy (including sampling and dilution uncertainties) be set at 10 % at a 1 sigma uncertainty for values above 1500 ppm. For values below 1500 ppm, an accuracy of 20% of 1500 ppm, or 300 ppm at a 1 sigma uncertainty level, should be sufficient to plan recovery actions. The value of 20% was chosen since it is outside the range required for safe stable conditions and a greater accuracy would mean that samples in this range would be more accurate than those above 1500 ppm which is the range of interest.

F. **Reactor Coolant Conductivity**:

WCNOC currently possess the capability of taking and analyzing a reactor coolant sample for conductivity using the PASS. However, there is no clear reason or documentation regarding the purpose of this sample or analysis.

The post accident sampling capability to measure the conductivity of the reactor coolant is not specified in NUREG-0737 or Regulatory Guide 1.97, Rev. 2. In addition, there are no accident management or emergency planning functions that require the RCS conductivity content. Therefore, elimination of the requirement to take and analyze a reactor coolant sample in a post accident condition for conductivity will not change the ability of WCNOC personnel to bring the plant to a safe, stable state following an accident.

G. **Reactor Coolant Radionuclides**:

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

The capability to measure radionuclides in the RCS supports the Emergency Action Level classification in the WCGS Radiological Emergency Response Plan, regardless of whether the NUREG-0654 or the NUMARC/NESP-007 classification scheme is used. One of the criteria for declaring an emergency is the identification of greater than 300 µCi/cc of equivalent I-131 in the RCS. 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. Other criteria, from NUMARC/NESP-007, for escalating an accident condition to an Alert or higher condition, include:

- high core exit thermocouple indication, or
- low reactor vessel water level indication, or
- high containment radiation level indication, or
- loss of RCS subcooling, or

- a safety injection signal, or
- indication of a failure to achieve subcriticality following a reactor trip.

Considering the accidents that could result in core damage, 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. However, deletion of this post accident sampling function requires that the Emergency Action Level criteria based on reactor coolant activity also be deleted. While it is recommended that this function be deleted as a post accident sampling system requirement, the capability to obtain a reactor coolant sample and analyze it for radioactivity should be retained (but not within the plant licensing basis) in order to assist in planning long term recovery activities.

H. **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.

1. Need for Containment Hydrogen PASS Capability:

The post accident sampling capability to measure the reactor coolant radionuclide content is a NUREG-0737 and Regulatory Guide 1.97 function. However, 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 are now installed in PWR containment's.

The capability to measure hydrogen concentration in the containment supports several accident management functions. The EMGs require knowledge of the containment hydrogen concentration for decisions regarding the use of the hydrogen recombiners (EMG FR-Z1, "Response To High Containment Pressure") and the reactor coolant head vent (EMG FR-I3, "Response To Voids In Reactor Vessel"). The Severe Accident Management Guidelines (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 or analysis of PASS samples is not available (i.e., by assuming a pre-determined bounding amount of hydrogen generation), it is not a long term substitute for measuring the actual containment hydrogen concentration.

Based on the SAMG philosophy of maintaining alternative diverse methods of obtaining information related to the containment hydrogen concentration, the optimal situation would be to continue the capability to measure containment hydrogen by both the on-line containment hydrogen monitor and the sampling of containment atmosphere. However, the present 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. However, sampling is a viable alternative, albeit with a time lag, and could be included in the SAMG list of alternatives.

Based on the above assessments and requirements, at least one means of obtaining a measurement of the containment hydrogen concentration is required. As long as the appropriate timing and accuracy needs outlined below can be met, the containment on-line hydrogen monitor would be an acceptable method for obtaining containment hydrogen concentration information.

2. Post Accident Sampling Timing and Accuracy:

An indication of the containment hydrogen concentration is required within about thirty (30) minutes of the onset of core damage (i.e., within 30 minutes of the time the core exit thermocouples exceed 1200°F). As discussed earlier, for an accident in which the reactor coolant system integrity is impaired (e.g. a LOCA) the containment hydrogen concentration can increase from near zero to a flammable state within 30 minutes. For non-LOCA core damage accidents, the increase is much slower since the only pathway from the reactor coolant system to the containment is via the pressurizer safety or relief valves. However, in the non-LOCA cases, the containment hydrogen concentration can increase from a non-flammable state to a flammable state in a matter of 5 to 10 minutes if the reactor coolant system integrity is lost after core damage occurs. Severe accident phenomena such as creep failure of the RCS hot leg or reactor vessel bottom head failure could result in such a condition.

Thus, there needs to be the capability to measure containment hydrogen concentration within 30 minutes after the onset of core damage. If the on-line containment hydrogen monitor is used to measure the containment hydrogen concentration, the capability exists for nearly continuous measurement. If sampling and analysis of containment atmosphere samples is used to measure the containment hydrogen concentration, there needs to be the capability to measure the containment hydrogen concentration at intervals of at least every 10 to 15 minutes after the initial measurement. In the case where sampling is used to measure containment hydrogen concentration, the plant SAMG needs to be enhanced:

a) to caution the user that the hydrogen indications are lagging real time, and

b) to provide a method to diagnose the possible rapid increases in containment hydrogen concentrations

due to severe accident phenomena from other plant parameters.

The accuracy of the on-line containment hydrogen monitor at WCGS is ± 1% hydrogen concentration. The ERG setpoints for containment hydrogen concentrations include instrument error and a ± 1% value is acceptable for carrying out the ERG instructions. In the case of the SAMG, the setpoints and computational aids were developed with margins to account for instrument error of ± 1% in measuring containment hydrogen. In the SAMG, the conversion of the generic WOG SAMG to a plant specific version does not include the development of setpoints that include instrument error. This approach was taken due to the uncertainties in instrument error under core damage accident conditions. The margins in the setpoints and computational aids were considered to be sufficient to cover possible instrument error under core damage accident conditions while still permitting the appropriate actions to be taken. [Note that if very large uncertainties in instrument performance are quantified in the diagnosis and evaluation of core damage accidents and possible recovery actions, the most appropriate actions for the actual plant conditions may never be considered.] Thus, a ± 1% analysis error is also appropriate for containment hydrogen measurements obtained from containment gas samples.

In the case of containment hydrogen, the sampling accuracy is highly influenced by the variations in the local hydrogen concentrations at the sample inlet inside containment, which in turn are dependent on a number of factors related to the accident sequence and recovery actions that have been implemented. Therefore, it is not possible to designate a requirement for sampling accuracy for containment hydrogen. Since the containment hydrogen is only used in decision-making related to the use of the severe accident management guidance, the variations in local containment hydrogen concentrations and how they influence the containment hydrogen indications (either on-line hydrogen monitor indications or the results of analysis of containment hydrogen samples using the post accident monitoring system capabilities) used for decision-making may be included in the SAMG training.

I. **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. However, the post accident sampling capability to measure the containment oxygen concentration is not a NUREG-0737 nor a Regulatory Guide 1.97 function for PWRs.

The capability to measure oxygen concentration in the containment does not support any accident management or emergency planning functions. Finally, based on the lack of regulatory requirements and the fact that the sample and analysis of oxygen concentration is not used for accident management or emergency planning functions, this sample and analysis can be deleted.

J. **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 capability to measure the radionuclide content of the containment atmosphere supports the 1984 WOG Core Damage Assessment Methodology and some of the offsite dose assessment procedures (e.g. AP 07B-003, "OffSite Dose Calculation Manual") in the Radiological Emergency Response Plan (RERP). 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 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 containment atmosphere would not be timely. Also, considering the behavior of fission products, 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 1996 WOG Core Damage Assessment Guidance, i.e. WCAP-14696, "Westinghouse Owners Group 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 RCS, 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 EMGs 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. However, the capability to sample containment atmosphere for radionuclide content after the plant has been returned to a controlled stable condition should be retained (but not within the plant licensing basis) to assist in long term recovery planning.

K. **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.

As discussed in Section J, "Containment Airborne Radioactive Samples," above, the capability to measure the radionuclide content of the containment atmosphere supports the 1984 WOG Core Damage Assessment Methodology and some of the offsite dose assessment procedures 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 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.

L. **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.

1. Need for Containment Sump pH PASS Capability:

The post accident sampling capability to measure the containment sump pH is not a NUREG-0737 item. However,

Regulatory Guide 1.97 requires sampling and analysis of containment sump pH.

Following an accident, the pH of the containment sump water is dependent on a large number of factors, including:

a) the amount of reactor coolant and accumulator water accumulated in the containment sump,

b) the operation of the containment spray system (i.e., the spray additive tank injection),

c) the amount of RWST water accumulated in the sump, and

d) 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).

2. Post Accident Sampling Timing and Accuracy:

Post accident sampling of the containment sump water for pH is not required immediately following an accident. A discussion within Westinghouse Nuclear Safety Advisory Letter (NSAL-93-016) identified that the containment sump water pH must be within the allowable range within 48 hours after the accident occurs to minimize the potential for chloride induced stress corrosion cracking of stainless steel piping.

Therefore, analysis of containment sump samples are not required in the first 24 hours after the initiation of the accident. A minimum required time interval for obtaining the first sample should therefore be set at 24 hours after the accident initiation. In addition, any pH sample analysis should be ± 0.3 units between a pH of 5.0 and 9.0 and ± 0.5 units for all other values. This accuracy requirement will be consistent with accuracy statements from an industry accepted group (i.e. American Nuclear Society).

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. Therefore, the capability to sample and analyze containment sump pH is not necessary. However, the capability to sample the containment sump for pH should be retained (but not within the plant licensing basis) to enable the plant operators the ability to ensure that chloride stress corrosion will not be a factor in any long term recovery effort.

M. **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 item.

Following an accident at WCGS, the chloride concentration of the containment sump water is dependent on the source of any additional water (beyond the original RWST volume) that has been injected either:

a) into the RCS (e.g., RWST refill with fire water from EMG C-11 "Loss of Emergency Recirculation") or

b) from the SAMG at SAG-3/SAG-6), or into the containment (e.g., SAMG at SAG-4)

The potential for chloride induced stress corrosion cracking of stainless steel piping is only a concern when the sump pH is acidic (i.e., the pH is less than 7). In addition, the potential for significant quantities of untreated water that may contain high levels of chlorides in the containment sump can be easily detected by a combination of:

1. monitoring the containment sump level and comparing the indicated level to the expected value based on the quantities of water that have been injected into either the RCS or the containment, and

2. predicting the chemical content of the sump water based on the chemical content of the water sources injected into the containment.

In the case where discrepancies between the indicated and predicted values cannot be resolved, the pH of the sump water can be adjusted to a neutral or basic state (i.e., pH of 7 or higher), thereby eliminating the chloride induced stress corrosion cracking concern.

Based on the above assessments of the need for post accident sampling capabilities for containment sump chlorides, this post accident sampling function is considered to be unnecessary.

N. **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 item. However, Regulatory Guide 1.97 requires sampling and analysis of containment sump boron.

There are no existing EMG or SAMG bases 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. 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 a return to criticality if the sump water is used for emergency core cooling recirculation.

Therefore, the only scenario where the containment sump boron concentration could be at a level where a return to criticality may be a concern when the water is used for emergency core cooling recirculation is when unborated water is added to the containment. These scenarios involve either:

1. the intentional injection of unborated water to the reactor coolant system or containment, or

2. 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.

V. CONCLUSIONS AND SUMMARY:

A. Conclusions:

The discussions contained throughout this report concerning all of the samples and analyses performed by WCNOC in fulfillment of its regulatory guidance for post accident sampling have been performed after a careful and thorough review of WCAP-14986, "Westinghouse Owners Group Post Accident Sampling System Requirements: A Technical Basis." These reviews and evaluations clearly show that the majority of the samples and analysis performed do not aid emergency response personnel or plant personnel in any accident control functions. On the contrary, these samples and analyses divert attention from more critical and telling sample information results. Therefore, the samples and analysis that are proposed to be retained are:

1. RCS boron, and
2. Containment hydrogen

B. Summary:

Table 1 summarizes the WCNOC proposed PASS recommendations in conjunction with Westinghouse WCAP-14986. A comparison matrix identifying the corresponding WCNOC procedures for the procedures referenced throughout WCAP-14986 is provided as Attachment A.

Table 1 - Summary of Proposed Post Accident Sample System Functions

Sample Point/Analysis	Guidance			Recommendation	Comments
	•Regulatory	•Accident Management	•Emergency Planning		
RCS					
Dissolved Gases	0737 / 1.97	EMG	N/A	Delete from PASS	Original function is tied to RCS natural circulation cooling concerns
Hydrogen	0737 / 1.97	N/A	N/A	Delete from PASS	0737 function is only as an alternative to total dissolved gas
Oxygen	1.97	N/A	N/A	Delete from PASS	Original function is tied to RCS chlorides
pH	1.97	N/A	N/A	Delete from PASS	
Chlorides	0737 / 1.97	N/A	N/A	Delete from PASS	
Boron	**0737 / 1.97**	**EMG**	**N/A**	**Retain in PASS**	**Extend sample and analysis timing to 8 hours**
Conductivity	N/A	N/A	N/A	Delete from PASS	No guidance exist for this sample and analysis
Radionuclides	0737 / 1.97	N/A	EAL	Delete from PASS	Requires NRC approval of WCAP-14696 which must be implemented in conjunction with WCAP-14986.

Table 1 - Summary of Proposed Post Accident Sample System Requirements (continued)

Sample Point/Analysis	Requirement			Recommendation	Comments
	❶Regulatory	❷Accident Management	❸Emergency Planning		
Containment Atmosphere					
Hydrogen	**0737 / 1.97**	**EMG / SAM**	**EAL**	**Retain analysis, but not as part of PASS**	**Sample and analysis will be fulfilled by Containment Hydrogen analyzers**
Oxygen	N/A	N/A	N/A	Delete from PASS	
Radionuclides	0737 / 1.97	SAM	EAL	Delete from PASS	
Containment Sump					
pH	1.97	EMG	N/A	Delete from PASS	
Chlorides	1.97	N/A	N/A	Delete from PASS	
Boron	1.97	N/A	N/A	Delete from PASS	
Radionuclides	1.97	EMG	EAL	Delete from PASS	Requires change to EMG FR-Z2, "Response To Containment Flooding"

Notes:

❶ 0737 / 1.97 refers to NUREG-0737 and Regulatory Guide 1.97, Rev. 2, respectively

❷ EMG refers to one of the EMG procedures. For the specific procedure, refer to the applicable sample section within the report body

SAM refers to one of the SAM guidelines. For the specific guideline, refer to the applicable sample section within the report or to the SAM guidelines used as Reference material.

❸ ODCM refers to AP 07B-003, Rev. 1, "Offsite Dose Calculation Manual"

EAL refers to EP 01-2.1-1, Rev. 11, "Emergency Action Levels"

VI. REFERENCES:

A. USQD 59 98-0071

B. RCMS ID # 98-052

C. PIR 98-0276

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

E. Reg. Guide 1.97 Rev. 2, "Instrumentation For Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident"

F. Westinghouse Owners Group WCAP-14696, "Westinghouse Owners Group Core Damage Assessment Guidance"

G. Westinghouse Owners Group WCAP-14986, "Post Accident Sampling System Requirements: A Technical Basis"

H. WCNOC letter # WO 98-0047

I. WCNOC-5A, Rev. 2, "Wolf Creek Generating Station Core Damage Assessment Methodology"

J. AP 06-003, Rev. 0, "Severe Accident Management"

K. AP 07B-003, Rev. 1, "Offsite Dose Calculation Manual"

L. EPP 01-2.4, Rev. 7, "Core Damage Assessment Methodology"

M. EPP 01-2.1, Rev. 18, "Emergency Classification"

N. EP 01-2.1-1, Rev. 11, "Emergency Action Levels"

O. EMG F-0, Rev. 12, Critical Safety Function Status Trees (CSFST)

P. EMG E-1, Rev. 11, "Loss of Reactor or Secondary Coolant"

Q. EMG ES-03, Rev. 10, "SI Termination"

R. EMG ES-04, Rev. 8, "Natural Circulation Cooldown"

S. EMG ES-05, Rev. 8, "Natural Circulation Cooldown With Steam Void In Vessel (Without RVLIS)"

T. EMG ES-06, Rev. 8, "Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS)"

U. EMG ES-11, Rev. 12, "Post LOCA Cooldown and Depressurization"

V. BD-EMG E-1, Rev. 2, "Loss of Reactor or Secondary Coolant"

W. EMG FR-C1, Rev. 10, "Response to Inadequate Core Cooling"

X. EMG FR-I3, Rev. 8, "Response to Voids in Reactor Vessel"

Y. SAM CA-01, Rev. 0, "Severe Accident Management Guideline Computational Aids"

Z. SAM DFC-01, Rev. 0, "Diagnostic Flow Chart"

AA. SAM SACRG-01, Rev. 0, "Severe Accident Control Room Guideline Initial Response"

BB. SAM SACRG-02, Rev. 0, "SACRG For Transients After TSC Is Functional"

CC. SAM SAEG-01, Rev. 0, "TSC Long Term Monitoring"

DD. SAM SAEG-02, rev. 0, "SAMG Termination"

EE. SAM SAG-01, Rev. 0, "Inject Into The Steam Generators"

FF. SAM SAG-02, Rev. 0, "Depressurize The RCS"

GG. SAM SAG-03, Rev. 0, "Inject Into RCS"

HH. SAM SAG-04, Rev. 0, "Inject Into Containment"

II. SAM SAG-05, Rev. 0, "Reduce Fission Product Releases"

JJ. SAM SAG-06, Rev. 0, "Control Containment Conditions"

KK. SAM SAG-07, Rev. 0, "Reducing Containment Hydrogen"

LL. SAM SAG-08, Rev. 0, "Flood Containment"

MM. SAM SCST-01, Rev. 0, "Severe Challenge Status Trees"

NN. SAM SCG-01, Rev. 0, "Reducing Fission Product Releases"

OO. SAM SCG-02, Rev. 0, "Depressurize The Containment"

PP. SAM SCG-03, Rev. 0, "Controlling Hydrogen Flammability"

QQ. SAM SCG-04, Rev. 0, "Controlling Containment Vacuum"

RR. Radiological Emergency Response Plan, Rev. 58

1. section 3.1.2
2. section 3.3.1.3
3. section 5.2.4

SS. USAR Section 6.2

TT. USAR sections 18.2.1 through 18.2.1.3

UU. NUREG-0881, "SER Related to Operation of Wolf Creek Generating Station"

VV. Technical Specification 3.3.3.6

WW. Technical Specification 3.6.4.1

XX. Technical Specification 3.6.4.2

YY. Technical Specification 3.4.11

ZZ. Technical Specification 6.8.4.d

AAA. Technical Specification Sections 3.0 and 4.0 Bases

BBB. Severe Accident Management Guidance, June, 1994

CCC. E-03BB30, "RCS Heat Vent Valves"

DDD. M-12BB04, "P&ID Reactor Coolant System"

EEE. M-13BB17, "SM PP ISO REAC COOL SYS REAC BLDG"

FFF. J-352-00088, "IM Normal & Post-Acdt Sampl Sys Vol-1"

ATTACHMENT A TO ENCLOSURE 1

COMPARISON MATRIX BETWEEN WCAP-14986 PROCEDURES AND WCNOC PROCEDURES

WOG Emergency Response Guide (ERG)	Corresponding WCNOC Procedure
• E-1, Loss of Reactor or Secondary Coolant	• EMG E-1, Rev. 11, "Loss of Reactor or Secondary Coolant"
step 12(c), "Obtain samples [enter plant specific list]"	step 24, "Obtain Samples: ..." Note: Chemistry may take the samples by using either the: O Nuclear Sample System, or O Post Accident Sampling System, or O Post Accident Monitoring System
step 16, "Determine if reactor vessel head should be vented; consult plant Engineering staff"	step 32, "Determine if reactor ..."
step 19, "Evaluate long term plant status: consult plant Engineering staff"	step 36, "Evaluate long term plant status: ..."
• E-2, "Faulted Steam Generator Isolation"	• EMG E-2, Rev. 8, "Faulted Steam Generator Isolation"
step 6, "Check secondary radiation"	step 11, "Determine Secondary Radiation Levels: ..."
step 6(a), "Request periodic activity samples of all SGs; [enter plant specific means]"	step 11(g), "Direct Chemistry to sample all S/G's for activity."
• E-3, "Steam Generator Tube Rupture"	• EMG E-3, Rev. 11, "Steam Generator Tube Rupture"
step 2, "Identify Ruptured S/G(s): High radiation from any S/G sample"	step 5, "Identify Ruptured S/Gs: * ... * ... * High radiation from any S/G steamline radiation monitor * ... "
• ES-0.2, "Natural Circulation Cooldown"	• EMG ES-04, Rev. 8, "Natural Circulation Cooldown" • EMG ES-05, Rev. 8, Natural Circulation Cooldown With Steam Void In Vessel (Without RVLIS) • EMG ES-06, Rev. 8, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS)

COMPARISON MATRIX BETWEEN WCAP-14986 PROCEDURES AND WCNOC PROCEDURES

WOG Emergency Response Guide (ERG)	Corresponding WCNOC Procedure
step 3, "Verify Cold Shutdown Boron Concentration (in the RCS) by Sampling	O EMG ES-04, step 5, "Verify Cold Shutdown Boron Concentration by Sampling: ..." O EMG ES-05, "Caution" prior to step 1 requires steps 1 - 14 of EMG ES-04 be completed before performing any of EMG ES-05. O EMG ES-06, "Caution" prior to step 1 requires steps 1 - 14 of EMG ES-04 be completed before performing any of EMG ES-06.
• ES-1.2, "Post LOCA Cooldown and Depressurization"	• EMG ES-11, Rev. 12, "Post LOCA Cooldown and Depressurization"
Note prior to step 7, "Shutdown margin should be monitored during RCS cooldown."	Note prior to step 13. "Shutdown margin shall be monitored during RCS cooldown."
step 21, "Verify adequate shutdown margin"	step 36, "Verify Adequate Shutdown Margin: ..."
step 21(a), "Sample RCS"	step 36(a), "Direct Chemistry to sample RCS for boron."
step 32, "Evaluate long term plant status: consult plant Engineering staff"	step 49, "Evaluate Long Term Plant Status: a. ... b. "Consult plant engineering staff"
• ES-3.1, "Post SGTR Cooldown Using Backfill"	• EMG ES-31, Rev. 9, "Post-SGTR Cooldown Using Backfill"
step 3, "Verify adequate shutdown margin"	step 3, "Verify Adequate Shutdown Margin: ..."
step 3(a), "Sample ruptured S/G(s)"	step 3(a), "Direct Chemistry to sample RCS and ruptured S/G(s) for boron."
step 3(b), "Sample RCS"	step 3(a), "Direct Chemistry to sample RCS and ruptured S/G(s) for boron."
step 12, "Evaluate long term plant status: consult plant Engineering staff"	step 15, "Evaluate Long Term Plant Status: a. ... b. "Consult plant engineering staff"
• ES-3.2, "Post SGTR Cooldown Using Blowdown"	• EMG ES-32, Rev. 9, "Post SGTR Cooldown Using Blowdown"

COMPARISON MATRIX BETWEEN WCAP-14986 PROCEDURES AND WCNOC PROCEDURES

WOG Emergency Response Guide (ERG)	Corresponding WCNOC Procedure
step 3, "Verify adequate shutdown margin	step 3, "Verify Adequate Shutdown Margin"
step 3(a), "Sample ruptured SGs"	step 3(a), "Direct Chemistry to sample RCS and ruptured S/G(s) for boron."
step 3(b), "Sample RCS"	step 3(a), "Direct Chemistry to sample RCS and ruptured S/G(s) for boron."
step 16, "Evaluate long term plant status: consult plant Engineering staff"	step 19, "Evaluate Long Term Plant Status: a. ... b. "Consult plant engineering staff"
• EMG ES-3.3, Rev. 1B, "Post-SGTR Cooldown Using Steam Dump"	• EMG ES-33, Rev. 9, "Post SGTR Cooldown Using Steam Dump"
step 3, "Verify adequate shutdown margin"	step 5, "Verify Adequate Shutdown Margin"
steps 3(a) and (b), "Sample Ruptured S/G" and "Sample RCS"	step 5(a), "Direct Chemistry to sample RCS and ruptured S/G(s) for boron."
Step 16, "Evaluate Long Term Plant Status a. Maintain Cold Shutdown Conditions b. Consult Plant Engineering Staff	step 21, "Evaluate Long Term Plant Status: a. ... b. Consult plant engineering staff"
• ECA-0.1, "Loss of All AC Power Recovery Without SI Required"	• EMG CS-01, Rev. 10, "Loss of All AC Power Recovery Without SI Required"
step 17, "Verify adequate shutdown margin"	step 28, "Verify Adequate Shutdown Margin: ..."
step 17(a), "Sample RCS"	step 28(b), "Direct Chemistry to sample RCS for boron."
• ECA-2.1, "Uncontrolled Depressurization of Steam Generators"	• EMG C-21, Rev. 9, "Uncontrolled Depressurization of All Steam Generators"
step 43, "Evaluate long term plant status: consult plant Engineering staff"	step 61, "Evaluate Long Term Plant Status: a. ... b. Consult plant engineering staff"

COMPARISON MATRIX BETWEEN WCAP-14986 PROCEDURES AND WCNOC PROCEDURES

WOG Emergency Response Guide (ERG)	Corresponding WCNOC Procedure
• ECA-3.1, "SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired"	• EMG C-31, Rev. 13, "SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired"
step 25, "Verify adequate shutdown margin"	O Note prior to step 21 requires that Shutdown Margin be monitored during RCS cooldown O step 45, "Verify Adequate Shutdown Margin: ..."
step 25(a), "Sample ruptured SG(s)"	step 15, "Obtain Samples: O ... O ... O ... O Steam Generators"
step 25(b) "Sample RCS"	step 15, "Obtain Samples: O RCS loops O ... O ... O ..."
step 38, "Evaluate long term plant status: consult plant Engineering staff"	step 60, "Evaluate Long Term Plant Status: a. ... b. Consult plant engineering staff"
• ECA-3.2, "SGTR With Loss of Reactor Coolant - Saturated Recovery Desired"	• EMG C-32, Rev. 13, "SGTR With Loss of Reactor Coolant - Saturated Recovery Desired"
step 19, "Verify adequate shutdown margin"	• Note prior to step 8 requires that Shutdown Margin be monitored during RCS cooldown • step 34, "Verify Adequate Shutdown Margin: ..."
step 19(a), "Sample ruptured SG(s)"	"Notes" prior to step 1 requires steps 1 - 22 of EMG C-31 be completed before performing any of EMG C-32. Sampling of S/G's is performed in step 15 of EMG C-31.
step 19(b), "Sample RCS"	"Notes" prior to step 1 requires steps 1 - 22 of EMG C-31 be completed before performing any of EMG C-32. Sampling of the RCS is performed in step 15 of EMG C-31.

COMPARISON MATRIX BETWEEN WCAP-14986 PROCEDURES AND WCNOC PROCEDURES

WOG Emergency Response Guide (ERG)	Corresponding WCNOC Procedure
step 32, "Evaluate long term plant status: consult plant Engineering staff"	step 49, "Evaluate Long Term Plant Status: a. ... b. Consult plant engineering staff"
• ECA-3.3, "SGTR Without Pressurizer Pressure Control"	• EMG C-33, Rev. 9, "SGTR Without Pressurizer Pressure Control"
step 23, "Verify adequate shutdown margin"	step 36, "Verify Adequate Shutdown Margin: ..."
step 23(a), "Sample ruptured SG(s)"	step 36(a), "Direct Chemistry to sample RCS and ruptured S/G's for boron."
step 23(b), "Sample RCS"	step 36(a), "Direct Chemistry to sample RCS and ruptured S/G's for boron."
step 37, "Evaluate long term plant status: consult plant Engineering staff"	step 52, "Evaluate Long Term Plant Status: a. ... b. Consult plant engineering staff"
• FR-C.1, "Response to Inadequate Core Cooling"	• EMG FR-C1, Rev. 10, "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"	Note prior to step 9, "This procedure shall be continued while obtaining containment hydrogen sample."
step 8(a), "Obtain hydrogen concentration [enter plant specific means]"	step 9, "Obtain Containment Hydrogen Concentration: ..."
step 8(b), "RNO if containment hydrogen concentration is greater than 6 percent volume), consult plant Engineering staff for additional recovery actions."	O step 9(b), "Hydrogen concentration - LESS THAN 4% IN DRY AIR" O step 9(b) Response Not Obtained: "Perform the following: 1) Notify plant engineering staff of hydrogen concentration inside containment. 2) Periodically obtain a hydrogen concentration measurement. 3) OBSERVE CAUTION PRIOR TO STEP 10 and go to step 10."

COMPARISON MATRIX BETWEEN WCAP-14986 PROCEDURES AND WCNOC PROCEDURES

WOG Emergency Response Guide (ERG)	Corresponding WCNOC Procedure
step 8(c), "RNO if containment hydrogen concentration is greater than 0.5 percent volume), turn on hydrogen recombiner system."	O step 9(c), "Hydrogen concentration - LESS THAN 0.8% IN DRY AIR" O step 9(c) Response Not Obtained: "Turn on hydrogen recombiners using SYS GS-120, POST LOCA CONTAINMENT HYDROGEN RECOMBINER OPERATION."
• FR-H.3, "Response to Steam Generator Overfill"	• EMG FR-H3, Rev. 7, "Response to Steam Generator High Level"
step 7, "Check affected SG(s) radiation [enter plant specific means]"	step 11, "Determine Secondary Radiation Levels: ..."
• FR-I.3, "Response to Voids in Reactor Vessel"	• EMG FR-I3, Rev. 8, "Response To Voids In Reactor Vessel"
step 12, "Obtain containment hydrogen concentration measurement: [enter plant specific means]"	step 22, "Obtain Containment Hydrogen Concentration Measurement: ..."
• FR-Z.1, "Response to Containment High Pressure"	• EMG FR-Z1, Rev. 8, "Response to High Containment Pressure"
step 7, "Check hydrogen Concentration"	step 14, "Check Containment Hydrogen Concentration: ..."
step 7(a), "Obtain a current [containment] hydrogen measurement"	step 14(a), "Hydrogen analyzers - IN OPERATION" ..."
step 7(b), RNO (if containment hydrogen concentration is greater than 6 volume percent), "Consult plant engineering staff for additional recovery actions."	Step 14(b), "Hydrogen concentration - LESS THAN 4.0% IN DRY AIR" RNO - Perform the following: 1) Notify plant engineering staff of hydrogen concentration inside containment. 2) Periodically obtain a hydrogen concentration measurement. 3) ..."
step 7(c), RNO (if containment hydrogen concentration is greater than 0.5 volume percent), "Turn on Hydrogen Recombiner System"	Step 14(c), "Hydrogen concentration - LESS THAN 0.8% IN DRY AIR" RNO - Turn on hydrogen recombiners using SYS GS-120, "POST LOCA CONTAINMENT HYDROGEN RECOMBINER OPERATION"
step 9, "Periodically obtain a Hydrogen Concentration Measurement"	step 14, "Check Containment Hydrogen Concentration: ..."

COMPARISON MATRIX BETWEEN WCAP-14986 PROCEDURES AND WCNOC PROCEDURES

WOG Emergency Response Guide (ERG)		Corresponding WCNOC Procedure
• FR-Z.2, "Response To Containment Flooding"		• EMG FR-Z2, Rev. 6, "Response To Containment Flooding"
step 2, "Check containment sump activity level [enter plant specific means]"		step 3, "Sample containment sumps: ..."

ENCLOSURE 2

EVALUATION OF THE PROPOSED CHANGES TO THE WCGS CORE DAMAGE ASSESSMENT GUIDANCE

Discussion

The fundamental need for a method to estimate the type and degree of core damage stems from the need to provide timely recommendations with respect to offsite radiological protective measures that should be implemented to protect the health and safety of the public. By letter KMLNRC 84-155 dated August 31, 1984, WCNOC submitted a Core Damage Assessment Methodology (CDAM) that was based upon a generic Westinghouse methodology previously approved by the NRC in NUREG-0881 Supplement 5, "Safety Evaluation Report Related to the Operation of Wolf Creek Generating Station, Unit No. 1," Item II.B.3, "Post Accident Sampling System."

Since 1984 much has been learned about the progression of core damage accidents, the plant behavior during these types of accidents, and the projected consequences of the accidents. By letter OG-96-098 dated November 22, 1996, the Westinghouse Owner's Group transmitted an updated CDAM (WCAP-14696, "Westinghouse Owner's Group Core Damage Assessment Guidance, July 1996") to the NRC.

The development of plant specific core damage assessment guidance based on the information contained in WCAP-14696 will result in the capability to make an assessment of core damage based on current knowledge and understanding of core damage accidents and fission product behavior.

Development of Wolf Creek Core Damage Assessment Guidance

WCAP-14696 was followed as written during the development of the Wolf Creek Core Damage Assessment Guidance (CDAG).

Evaluation

The 1984 Westinghouse Owners Group (WOG) CDAG was based on the knowledge of the progression of core damage accidents and the fission product behavior that was available prior to 1984. Thus the WOG CDAM placed nearly equal reliance on the measurement of specific fission products isotopes (including xenon, krypton, iodine, cesium, tellurium, strontium, and barium). The methodology used the auxiliary indicators (containment area radiation monitors, core exit thermocouples, containment hydrogen monitors, and reactor vessel water level) as a confirmation that core damage had occurred, but not to provide a useful estimate of the degree of core damage.

WCAP-14696 found that the interpretation of the results of analyses of the radionuclide inventories in various plant fluids, using the PASS, does not provide reliable or timely information for estimating the degree of core damage while the accident is in progress. Additionally, it was found that only knowledge of the noble gas inventory in the containment atmosphere provides any basis for making core damage estimates after the recovery from the accident has been completed (i.e., after the core has been returned to a safe, cooled, stable state). Further, since the only need for estimating core damage after recovery from a core damage accident has been completed is for in-plant purposes, the development of a core damage assessment methodology based on analysis of samples of radioactive fluids is not required to meet the intent of regulatory needs. Although the estimation of core damage based on analysis of radioactivity in plant fluids was part of the NUREG-0737 items, current knowledge of core damage accidents provides evidence that this is no longer necessary.

The WCAP-14696 generic guidance is meant to replace the 1984 version as the basis for Wolf Creek plant-specific core damage assessment. It takes into account all of the current information on fission product behavior and the progression of a core damage accident. The new CDAG relies solely on fixed in-plant instrumentation for diagnosing the existence of core damage and estimating the amount of core damage. It is applicable to transient situations where core degradation is progressing, as well as to quasi-steady state conditions after recovery from a core damage accident has been successfully completed.

The technical investigations that were performed during the development of the new WOG CDAG lead to the conclusion that several of the original bases for core damage assessment items in NUREG-0737, Criteria II.B.3, cannot be justified based on the current understanding of core damage accidents and fission product behavior. The portions of the NUREG-0737 criteria that have been concluded to be obsolete are:

1) The definition of the types of core damage - NUREG-0737 described core damage as one of four categories: no damage, cladding failure, fuel overheating and fuel melting. The investigation performed during the development of the generic WOG CDAG lead to the conclusion that core melting, as a separate category, does not have to be diagnosed in order to provide offsite radiological protection recommendations. The fission products that are important for determining the severity of the offsite radiological risks are released from the fuel during the fuel overheating phase of core damage. Additionally, the generic WOG Severe Accident Management Guidance determined that the diagnosis of core melting was not required to evaluate and implement in-plant recovery strategies. Thus, the new WOG CDAG does not provide a means for a separate diagnosis of core melting; the diagnosis of a core overtemperature condition is adequate.

2) Sampling and Analysis of Fission Products - NUREG-0737 guidance includes the capability to sample reactor coolant and containment for noble gases, iodines, cesiums and other nonvolatiles within two hours. The investigations carried out during the development of the generic WOG CDAG determined that the results of samples would not be timely for the purposes of making offsite radiological protection recommendations. Additionally, the results of analyses of samples of plant fluids would not provide any clarification of the type and degree of core damage compared to that obtained using fixed in-plant instrumentation. Thus, the need for post accident sampling of plant fluids for the determination of the type and degree of core damage is not required by the new WOG Core Damage Assessment Guideline.

Conclusion

Based on the above evaluation, using the WCAP-14696 methodology for core damage assessment will improve the ability of WCNOC to protect the health and safety of the public.