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August 6, 1982

In reply, please
refer to LAC-8468

DOCKET NO. 50-409

Director of Nuclear Reactor Regulation
ATTN: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: DAIRYLAND POWER COOPERATIVE
LA CROSSE BOILING WATER REACTOR (LACBWR)
PROVISIONAL OPERATING LICENSE NO. DPR-45
NUREG-0737, ITEM II.B.3, POST-ACCIDENT SAMPLING SYSTEM

REFERENCES: (1) NRC Letter, Crutchfield to Linder, dated June 30, 1982
(2) DPC Letter, Linder to Denton, LAC-6979, dated June 12, 1980

Gentlemen:

Dairyland Power has used NUREG-0737 as guidance in designing and installing the Post Accident Sampling System. Attachment 1 provides a brief description of the Reactor Liquid and Containment Building Atmospheric Sample Systems. Also included are drawings showing flow paths and major system components. Attachment 2 addresses the eleven criterions itemized in Reference 1.

If you have any questions regarding this submittal, please contact us.

Very truly yours,

DAIRYLAND POWER COOPERATIVE

Frank Linder

Frank Linder, General Manager

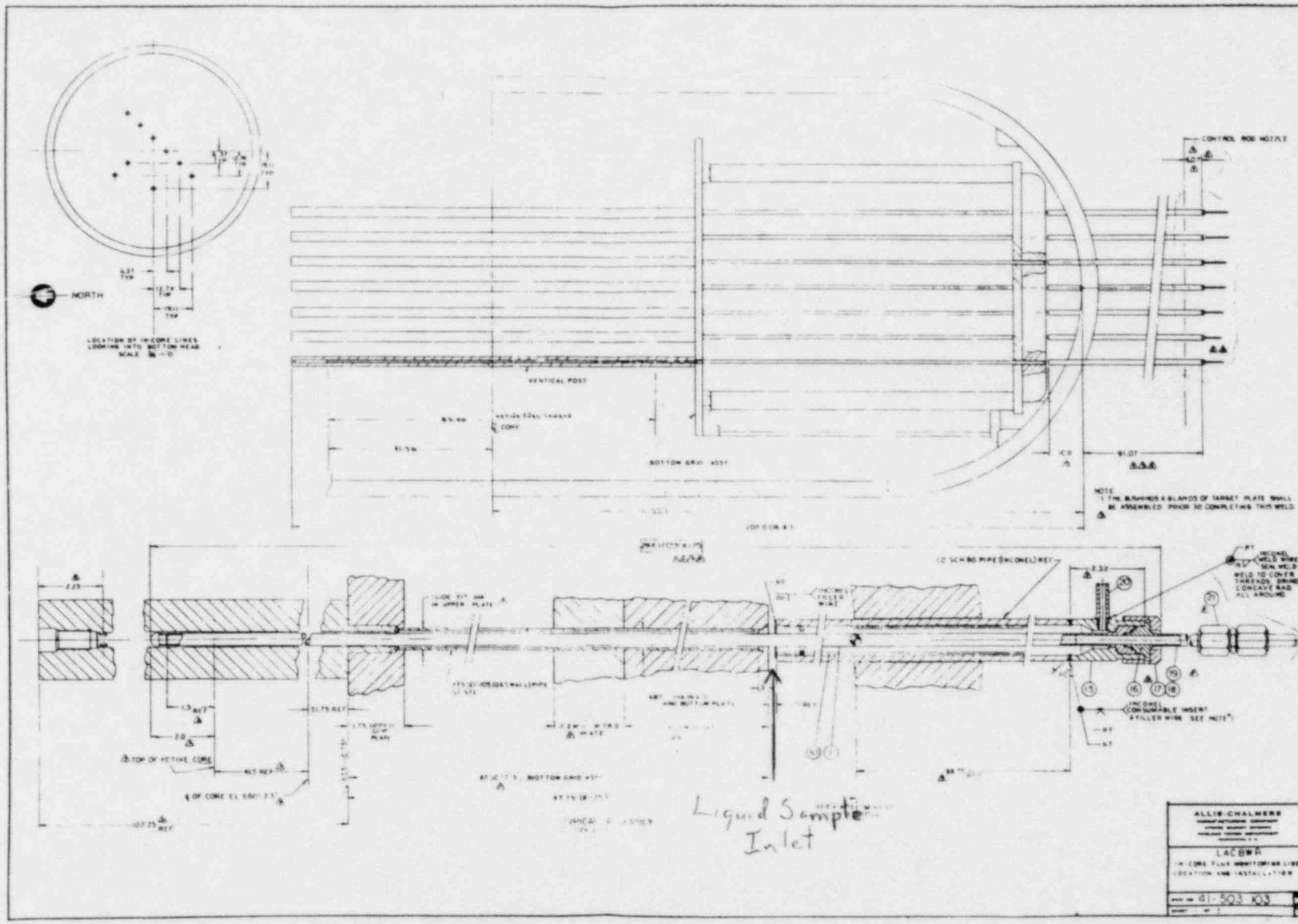
FL:RMB:eme

Enclosures

cc: J. G. Keppler, Regional Administrator, NRC-DRO III
NRC Resident Inspector

A046 Aperture Card Dist
Drawings To: BC

WP-2.5



IN-CORE FLUX MONITORING LINES LOCATION AND INSTALLATION

Enclosure 1

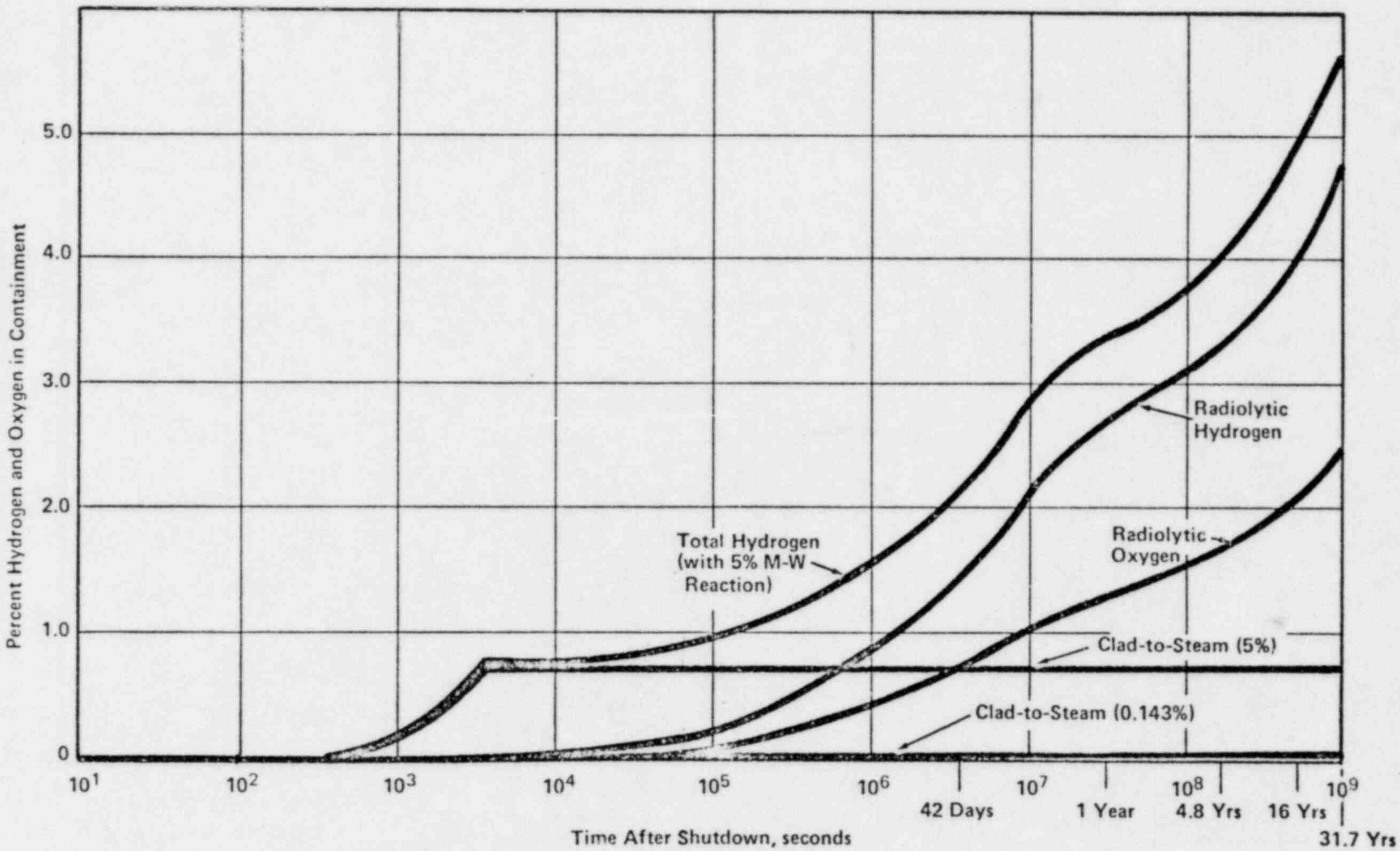


Fig. 1 — Containment Building – Hydrogen and Radiolytic Oxygen – Without Purge

Enclosure 2

SEE

APERTURE

CARDS

AVAILABILITY PDR CF HOLD

NUMBERS OF PAGES 2

ATTACHMENT 1

The Reactor Coolant Post-Accident Sampling System (Drawing 84-51-001) obtains primary coolant from an incore flux monitoring flushing connection. The sample passes through an inside containment solenoid operated isolation valve, a sample heat exchanger, through the containment shell and external solenoid operated isolation valve. The sample pressure is reduced as it passes through a motor operated throttle valve and is diluted with demineralized water as it flows through the sample cylinder, or its bypass valve and is directed to the waste tanks or through an external containment solenoid operated isolation valve and internal isolation check valve, which returns the sample flow to the containment building basement.

The Containment Atmosphere Post-Accident Sampling System (Drawing 84-51-0^2) consists of a vacuum pump which takes a suction on the containment atmosphere at the 714' level. The atmosphere sample is drawn through two solenoid operated isolation valves, a heat exchanger, and moisture trap. Then the sample is discharged to the two in-parallel hydrogen analyzers with preset flowmeters; then either through a bypass line or a remote sample cylinder and back to the containment atmosphere through two solenoid operated isolation valves.

ATTACHMENT 2

NRC CRITERION 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.

DPC RESPONSE:

The reactor coolant P.A.S.S. sample cylinder is located in the Feedwater Heater area, elevation 640' which is located about 30 m from the LACBWR Radioanalytical Chemistry Laboratory where the sample preparation and analysis would be performed. The Radioanalytical Chemistry Laboratory is located on the 640' elevation on the North side of the Turbine Building. The reactor coolant post-accident sample container consists of a 10cc stainless steel cylinder which enables it to be removed once the sample is collected and the reactor coolant P.A.S.S. is valved out of service. Depending upon the radiation levels on contact and at 1 foot from the cylinder, the cylinder could be moved by tongs directly, or moved in a shielded mobile lead pig to the lab, and placed behind lead bricks inside the lab hood, where an aliquot could be drawn from the cylinder, diluted in water and a second aliquot drawn for radioisotopic analysis.

Reference 2 summarizes the results of a plant shielding study performed by Nuclear Energy Services, Inc., and preliminary results from a supplmental analysis indicate that cumulative personnel whole body dose during reactor coolant sampling operations will be less than the criteria of GDC 19 (Appendix A, 10CFR50) as recommended in the clarification section for Item II.B.3 of NUREG-0737. As assumed in Reference 2, it is estimated that the sampling time would be 10 minutes, handling the sample cylinder at 1 foot would be 12.5 minutes, transit time would be 5 minutes and radioisotopic analysis time would be 10 minutes. The total time estimate for sampling and analysis is approximately 1 hour. The supplemental analysis is considering special sample handling techniques, such as the use of extension tongs, a portable lead sample transfer pig, additional laboratory hood lead shielding and sample dilution techniques. These techniques will be used to minimize personnel exposure during sampling and analysis during potential emergencies. An Emergency Operating Procedure for the Reactor Coolant P.A.S.S. has been written and implemented, and an emergency sampling and analysis procedure shall be implemented prior to completion of the refueling outage.

The Containment Building atmospheric P.A.S.S. sample cylinder is also located in the Feedwater Heater area. The Containment Building atmospheric sample container consists of a 300cc stainless steel cylinder designed to collect a combination of particulate, radioiodines and noble gases. The cylinder would be analyzed directly on a Ge(Li) detector at extended geometries if radiation levels were sufficiently low. If necessary, shielding can be placed around the cylinder or a smaller aliquot of gases may be taken from the 300cc cylinder in order to reduce detector dead time to accomplish isotopic analysis. It is estimated that the time for sampling and analysis for Containment Building Atmospheric P.A.S.S. would be approximately 1 hour.

An Emergency Operating Procedure for the Containment Atmospheric P.A.S.S. has been implemented. An Emergency Preparedness Procedure, EPP-6, describing methodology for post-accident sampling sample handling and analyses shall be written and implemented prior to completion of the refueling outage.

NRC CRITERION 2:

The licensee shall establish an onsite radiological and chemical analysis capability to provide, within three-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; iodines and cesiums, 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) Alternately, have inline monitoring capabilities to perform all or part of the above analyses.

DPC RESPONSE:

(a) Provisions to estimate the extent of core damage based on sample radionuclide concentrations for both the reactor coolant P.A.S.S. and containment atmospheric P.A.S.S. are being developed, and will be included in the EPP-6 revision. Containment Building atmospheric specific activities will be related to High Range Containment Building ARM readings which are already related to percent core degradation as listed in the Emergency Action Levels found in EPP-1, which are as follows:

HRCB Area Radiation Monitor Reading	Approximate Fuel Degradation	Approximate Containment Noble Gas Activity
10^1-10^2 R/hr	0%	0.04-0.4 μCi/cc
10^3-10^4 R/hr	≈ 1%	4-40 μCi/cc
10^4-10^5 R/hr	≈ 10%	40-400 μCi/cc
10^5-10^6 R/hr	≈ 100%	400-4000 μCi/cc

Specific radionuclide concentrations related to core damage will be determined based upon core inventory values from the LACBWR Safeguards Report, extrapolated values from GE-BWR Radiation Source Report No. 22A2703R, Rev. 5, EPA-520/1-75-001, NUREG-0771 or the Rogovin Report after cross comparison between these reports is completed (see Table 1).

TABLE 1

ESTIMATED CORE ACTIVITY AT TO:

CORE ACTIVITY @ TO (CURIES)

RADIONUCLIDE	BASED ON LACBWR SAFEGUARDS TABLE 14-8	EXTRAPOLATEDΔΔ FROM GE-BWR RADIATION SOURCE REPORT	EXTRAPOLATEDΔΔ FROM EPA-520/1-75-001	EXTRAPOLATEDΔΔ FROM NUREG-0771	EXTRAPOLATEDΔΔ FROM IAEA TECH. REPORT NO.152
Total Noble Gases	3.61×10^7	5.94×10^7	1.89×10^7	1.72×10^7	9.2×10^6
Total Radio-Iodines	1.77×10^7	6.77×10^7	3.60×10^7	3.84×10^7	**
I-131	2.08×10^6	4.5×10^6	4.25×10^6	4.55×10^6	4.16×10^6
I-133	4.59×10^6	9.1×10^6	8.50×10^6	9.15×10^6	6.09×10^6
Total Particulates and Semi Volatiles	3.08×10^5*	5.45×10^8	**	2.22×10^8	8.9×10^7
Sr-89	6.38×10^4	3.96×10^6		5.05×10^6	6.30×10^6
Sr-90	3.84×10^3	3.80×10^5		2.0×10^5	2.4×10^5
Cs-134	3.00×10^3Δ	3.3×10^5		3.8×10^5	**
Cs-137	5.00×10^3Δ	5.4×10^5		2.6×10^5	1.8×10^5
Barium-140	**	7.43×10^6		8.6×10^6	8.5×10^6
Ru-103	**	7.26×10^6		5.9×10^6	5.1×10^6
Ru-106	**	2.64×10^6		1.35×10^6	3.6×10^5
Zr-95	**	7.43×10^6		8.05×10^6	8.12×10^6
Nb-95	**	7.43×10^6		8.05×10^6	7.95×10^6
Ce-141	7.46×10^4	7.43×10^6		8.05×10^6	7.89×10^6
Ce-144	4.96×10^4	5.78×10^6		4.55×10^6	4.4×10^6
Pr-144	3.27×10^4	5.78×10^6		**	4.4×10^6
Np-239	**	**		9.0×10^7	**
Y-90	3.91×10^3	3.96×10^3		2.1×10^5	2.4×10^5

* 1% of core inventory
Δ Estimated
** Not Listed
ΔΔ Extrapolated = $\frac{165\ Mw_t}{\text{Report Reactor Mwt}}$ x Activity

(b) Two hydrogen analysers with remote readout in the Control Room have been installed in the Containment Building Atmospheric Post-Accident Sampling System. This allows LACBWR to determine the hydrogen concentration within the Containment Building. The ability to obtain a grab sample exists. However, a method to determine the hydrogen concentration in the laboratory has not been established. Study SS-1180 conducted by Gulf Nuclear Fuels Company indicates when using stainless steel clad fuel that a hydrogen problem is insignificant. LACBWR uses stainless steel clad fuel. According to this study, a 4% hydrogen concentration would not be reached for 4.8 years after a LOCA, thus giving LACBWR sufficient time to perform laboratory analysis on or offsite. Included as Enclosure 2 is a graph displaying hydrogen buildup in the LACBWR Containment Building. Nuclear Regulatory Guide 1.7 also considers the hydrogen generation difference between zircaloy and stainless steel cladded fuel. This guide is written for plants using zircaloy cladded fuel and indicates all plants using stainless steel cladded fuel will be considered on an individual basis.

(c) Due to the fact LACBWR is a boiling water reactor, the amount of gases contained in the coolant is minimal. During plant operation a continuous degasification of the primary coolant is maintained by the air ejectors. Degasification of the primary coolant during hot shutdown is possible through the shutdown condenser. Because of these aspects, total gas analysis of the primary coolant has not been considered.

Chloride concentration in the primary coolant is performed routinely in LACBWR's Radioanalytical Chemistry Laboratory. The resultant Cl^- concentration is less than 0.02 ppm. Therefore, the chloride concentration can be determined during a potential accident condition using the existing analytical procedure while following guidelines for handling potentially highly radioactive samples.

A boron analysis is also performed routinely. Procedures exist which allow LACBWR personnel to determine the boron concentration in liquid samples. These procedures can be used in conjunction with guidelines for handling potentially highly radioactive samples.

(d) The only inline monitoring capability provided is for hydrogen level in the containment building atmospheric sample. Hydrogen sampling is addressed in DPC Response 2(b).

NRC CRITERION 3:

Reactor coolant and containment building atmospheric 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).

DPC RESPONSE:

The reactor coolant and containment building atmospheric sampling systems do not require an isolated auxiliary system to perform their sampling function. Both sample systems use heat exchangers cooled by component coolant water. The component cooling water system is not automatically isolated for any accident condition and is available to provide the necessary cooling function.

All solenoid valves used in these two systems that are required to operate for obtaining a sample are environmentally qualified and will be reviewed for addition to the preventative maintenance program. The reactor liquid sample system utilizes one motor operated metering valve located in the pipe tunnel. To assure liquid sampling capabilities, the motor operated valve will be administratively maintained in a partially open position that would permit liquid flow through the valve if the valve motor failed. During a postulated accident higher levels of radiation (5×10^4 R for a 100 day cumulative dose) may be experienced, however, continued operation of this valve is expected. All containment isolation valves are environmentally qualified with controls in the control room and administratively controlled to be in the shut position except during sampling or testing activities.

NRC CRITERION 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 H_2 concentration is recommended, but is not mandatory.

DPC RESPONSE:

As discussed in Criterion 2(c), studies conclude that hydrogen and oxygen generation in a plant using stainless steel cladded fuel is insignificant. Oxygen analysis is routinely performed on the primary coolant with an average resultant O_2 concentration of $\leq$ 0.1 ppm.

NRC CRITERION 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.

DPC RESPONSE:

Routine chloride analysis is performed and procedures exist which can be utilized during a postulated accident condition following guidelines for handling potentially highly radioactive samples. This analysis could be performed within 96 hours using dilution if necessary to minimize radiation levels. This dilution could be up to 1:500 to meet the 10 ppm criteria. Due to the limited volume of the post-accident liquid sample cylinder (10 ml) an undiluted sample analysis for Cl^- within 30 days is not possible due to storage and handling of the necessary sample volume. A sample with a dilution ratio of 1:3 could be performed within 30 days. This would significantly increase the accuracy of analysis from that of an initial sample dilution. If the radiation levels from the 10 ml reactor coolant sample is sufficiently low a larger sample cylinder is available to obtain the necessary sample volume for an undiluted Cl^- analysis.

NRC CRITERION 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).

DPC RESPONSE:

Reference 2 summarizes the results of a plant shielding study performed by Nuclear Energy Services, Inc., and preliminary results from a supplmental analysis indicate that cumulative personnel whole body dose during reactor coolant sampling operations will be less than the criteria of GDC 19 (Appendix A, 10CFR50) as recommended in the clarification section for Item II.B.3 of NUREG-0737. As assumed in Reference 2, it is estimated that the sampling time would be 10 minutes, handling the sample cylinder at 1 foot would be 12.5 minutes, transit time would be 5 minutes and radioisotopic analysis time would be 10 minutes. The total time estimate for sampling and analysis is approximately 1 hour. The supplemental analysis is considering special sample handling techniques, such as the use of extension tongs, a portable lead sample transfer pig, additional laboratory hood lead shielding and sample dilution techniques. These techniques will be used to minimize personnel exposure during sampling and analysis during potential emergencies. An Emergency Operating Procedure for the Reactor Coolant P.A.S.S. has been written and implemented, and an emergency sampling and analysis procedure shall be implemented prior to completion of the refueling outage.

NRC CRITERION 7:

The analysis of primary coolant samples for boron is required for PWRs. (Note that Rev. 2 of Regulatory Guide 1.97 specifies the need for primary coolant boron analysis capability at BWR plants).

DPC RESPONSE:

Procedures for determining the boron concentrations in either saturated boron solution (Boron Inject Solution) or in a liquid sample are currently part of the LACBWR Health & Safety Procedures. These procedures can be used in conjunction with guidelines for handling a potentially highly radioactive sample to obtain the boron concentration if injected into the primary coolant.

NRC CRITERION 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 anlayzing 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.

DPC RESPONSE:

As indicated in Criterion 2(b), the only inline monitoring devices in the post-accident sampling system are the two hydrogen analyzers located in the containment building atmospheric sampling system. DPC response to Criterion 2(b) of this letter addresses the need for grab sample analysis of H_2.

NRC CRITERION 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 1µ Ci/g to 10 Ci/g.

(b) Restrict the 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 a ventilation system design which will control the presence of airborne radioactivity.

DPC RESPONSE:

(a) Radionuclides in both the primary coolant P.A.S.S. and containment atmospheric P.A.S.S. will be identified and quantified by one of two Ge(Li) detectors which are coupled to a computer-based multichannel analyzer located in the Radioanalytical Chemistry Lab. The current isotopic libraries for normal routine liquid and gaseous samples are adequate to identify and quantify radionuclides expected in post-accident samples.

The specific activities in both samples were discussed in our response to Criterion 2(a).

Provisions have been made for in line dilution of the primary coolant sample, and a compressed gas system can be used to dilute the containment atmospheric sample, if necessary, thus reducing the sample cylinder's specific activities. In addition, lab dilution for liquid sample will be possible thus enabling measurement of specific activities within the recommended range of 10^{-6} to 10 Ci/ml.

(b) Samples from both reactor coolant P.A.S.S. and containment atmospheric P.A.S.S. will be initially prepared for analysis in the laboratory sample hood, which is located in a separate room from the detector system. The Ge(Li) detectors are also located in large lead and steel shields. In addition, after sample aliquots have been removed from P.A.S.S. sample cylinders, the P.A.S.S. cylinders containing residual activity will be removed from the lab prior to sample analysis. All of these methodologies will reduce background radiation levels and therefore will enable low Ge(Li) deadtime and sample analyses with an error less than a factor of 2. Residual (background) radiation levels will have no effect on H_2, Cl^- or Boron analyses.

NRC CRITERION 10:

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

DPC RESPONSE:

Gross Activity, Gamma Spectrum

The accuracy of instrumentation used to quantify gross activity and should individual radionuclide specific activities should remain accurate within a factor of two across the entire range. The Internal Proportional counters used for gross alpha and beta activity analyses should have an overall counting error of about 25-30%.

The Ge(Li)-MCA System should have an overall error of about 10-15%.

Boron

As in Criterion 7, boron analysis is not necessary in a BWR unless it is injected. Normal analysis is performed under existing LACBWR Health and Safety Procedures. The existing procedure for boron in a liquid sample has a minimum detectable concentration of 0.2 μg with a relative standard deviation of 22.8% and a relative error of 0%.

Chloride

Routine analysis of the chloride concentration in the primary coolant indicates a concentration of less than .02 ppm. The procedures that exist for chloride analysis have a minimum sensitivity of ± .02 ppm.

Hydrogen or Total

As indicated in Criterion 2(b), hydrogen and total gas capabilities at LACBWR are limited. The studies referenced in this criterion also indicate a significant time frame available to establish criteria for these analysis should the necessity arise.

Oxygen

Although this analysis is routinely performed, it could not be accomplished during a potential accident condition due to the possibility of high coolant activity and the volume of coolant necessary to perform the analysis. The reference material in Criterion 2(b) also indicates that a necessity to perform this analysis is minimal and the time frame to establish methods and procedures for the analysis is of significant length.

pH

Normal coolant pH is performed on an Orion 701 pH meter and would be available during a potential accident condition. This instrument has an accuracy of ± .1 pH unit and is buffer checked daily.

NRC CRITERION 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 building atmospheric 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.*

DPC RESPONSE:

(a) The routing of the containment building atmospheric and reactor liquid sample system lines were chosen to minimize the impact on habitable areas after a postulated accident. Portions of these systems outside containment can be purged either by compressed air, nitrogen or demineralized water to reduce any residual radiation level in the sample line. The need, extent, and specific method to be used to flush these lines would be made if the system was used following an accident.

The sample intake point for the containment building atmospheric system is located high (Elev.714') in the containment building to provide a representative sample and minimize the possibility of fouling the intake with loose material. The Reactor Liquid Sample System draws its sample from the bottom of the reactor vessel. An outboard in-core flux monitoring flushing line which protrudes several inches above the reactor vessel bottom provides a 1/4 inch opening in which a sample of the vessel water can be obtained (Enclosure 1). This suction configuration and in-line strainer will minimize the possibility of flow blockage.

Proper handling and disposal of samples will be in accordance with standard LACBWR radiological safety practices and specific post-accident sampling procedures.

Both post-accident sampling systems are designed with dual containment isolation valves. The containment building atmospheric sample system has environmentally qualified solenoids with one inside and outside the containment on both the suction and return lines. The reactor liquid sample system uses two environmentally qualified solenoids on the inlet line with one being located on each side of the containment. The return line has a check valve inside containment and an environmentally qualified solenoid outside containment. Liquid sample return will normally be sent back to containment, but can also be diverted to the waste storage tanks located in the turbine building basement. Attachment 1 includes drawings of the sample systems. In the event of a ruptured sample line, adequate isolation is provided to minimize the volume released outside containment.

All post-accident sampling system containment isolation solenoid valves are administratively controlled in the shut position. They are normally closed valves that require power to open. Remote control of these valves is provided at a common panel located in the Control Room.

(b) The ventilation exhaust from the sampling station (grab sample area) is common to the normal plant ventilation exhaust. A negative pressure is maintained on the turbine building by one or two of the 35,000 cfm rated stack fans. The flow path frcm the sample station would proceed through the pipe tunnel, a normally unoccupied space, and up the plant stack. The ventilation exhaust is not filtered at LACBWR. In the event of a post-accident sample being required, the individual will be required to wear protective clothing which will include respiratory protective equipment.