REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8304150498 DOC.DATE: 83/04/14 NOTARIZED: NO DOCKET #
FACIL:50-361 San Onofre Nuclear Station, Unit 2, Southern Californ 05000361
50-362 San Onofre Nuclear Station, Unit 3, Southern Californ 05000362
AUTH.NAME AUTHOR AFFILIATION
DIETCH,R. Southern California Edison Co.
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Forwards results of post-accident sampling sys demonstration test in support of 830224 application for Amends 19 & 5 to Licenses NPF-10 & NPF-15, respectively.

DISTRIBUTION CODE: B001S COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 62
TITLE: Licensing Submittal: PSAR/FSAR Amdts & Related Correspondence

NOTES:J Hanchett 1cy PDR Documents, ELD Chandler 1cy. 05000361
NRR Scaletti 1cy.
J Hanchett 1cy PDR Documents, ELD Chandler 1cy. 05000362

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	NRR/DL/ADL	1	0	NRR LB3 BC	1	0
	NRR LB3 LA	1	0	ROOD,H. 01	1	1
INTERNAL:	ELD/HDS2	1	0	IE FILE	1	1
	IE/DEPER/EPB 36	3	3	IE/DEPER/IRB 35	1	1
	IE/DEQA/QAB 21	1	1	NRR/DE/AEAB	1	0
	NRR/DE/CEB 11	1	1	NRR/DE/EHEB	1	1
	NRR/DE/EQB 13	2	2	NRR/DE/GB 28	2	2
	NRR/DE/HGEB 30	1	1	NRR/DE/MEB 18	1	1
	NRR/DE/MTEB 17	1	1	NRR/DE/SAB 24	1	1
	NRR/DE/SGEB 25	2	2	NRR/DHFS/HFEB40	1	1
	NRR/DHFS/LQB 32	1	1	NRR/DL/SSPB	1	0
	NRR/DSI/AEB 26	1	1	NRR/DSI/ASB	1	1
	NRR/DSI/CPB 10	1	1	NRR/DSI/CSB 09	1	1
	NRR/DSI/ICSB 16	1	1	NRR/DSI/METB 12	1	1
	NRR/DSI/PSB 19	1	1	NRR/DSI/RAB 22	1	1
	NRR/DSI/RSB 23	1	1	REG FILE 04	1	1
	RGN5	3	3	RM/DDAMI/MIB	1	0
EXTERNAL:	ACRS 41	6	6	BNL(AMDTS ONLY)	1	1
	DMB/DSS (AMDTS)	1	1	FEMA-REP DIV 39	1	1
	LPDR 03	1	1	NRC PDR 02	1	1
	NSIC 05	1	1	NTIS	1	1
NOTES:		3	3			

TOTAL NUMBER OF COPIES REQUIRED: LTTR 57 ENCL 50

Southern California Edison Company

P. O. BOX 800

2244 WALNUT GROVE AVENUE

ROSEMEAD, CALIFORNIA 91770

ROBERT DIETCH
VICE PRESIDENT

TELEPHONE
213-572-4144

April 14, 1983

H. R. Denton
Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket Nos. 50-361 and 50-362
San Onofre Nuclear Generating Station
Units 2 and 3

References: A) Letter from K. P. Baskin (SCE) to H. R. Denton (NRC) dated February 24, 1983

B) Letter from K. P. Baskin (SCE) to H. R. Denton (NRC) dated March 4, 1983

C) Letter from K. P. Baskin (SCE) to George W. Knighton (NRC) dated November 30, 1982

Reference A transmitted Amendment Application No. 19 to Operating License No. NPF-10 and Amendment Application No. 5 to Operating License No. NPF-15 for San Onofre Nuclear Generating Station, Units 2 and 3 (SONGS 2&3). These amendments request schedular relief from the Nuclear Regulatory Commission (NRC) to make the Post-Accident Sampling System (PASS) operable and to implement the Post-Accident Sampling Program. Reference B provided additional information to the NRC to clarify and supplement the requested license amendments.

Subsequently, on April 4-7, 1983, Messrs. Wm. Gammill, C. Willis, F. Witt and H. Rood met with Southern California Edison (SCE) and contractor personnel at SONGS. At that time information was provided to the NRC on the PASS hardware status, alternative post-accident sampling capability and the post-accident sampling program including procedures and training. In addition, a full demonstration of PASS was conducted for the NRR representatives as well as for Messrs. G. Yuhas and A. Chaffee from NRC Region V. The purpose of this letter is to provide the NRC with the results of the PASS Demonstration Test and with information discussed in the various meetings at SONGS.

B001

PASS Demonstration

The PASS Demonstration was conducted on April 5, 1983. The demonstration consisted of:

1) obtaining a reactor coolant system (RCS) sample and analyzing the sample for Boron, pH, O_2, H_2, total gas and radionuclides.

2) obtaining and analyzing a containment atmosphere sample for H_2 and radionuclides.

3) obtaining a diluted grab sample.

Approved SONGS procedures were followed in conducting the demonstration. No changes to the procedures were necessary during the demonstration. In addition, during the demonstration the NRC conducted a thorough inspection of the PASS equipment and found no leaks. The results of the demonstration test are indicated in tabular form on Enclosure 1.

As a matter of information, discussions were held regarding the surveillance accuracies of the PASS and their compliance with NRC guidance. Information on the PASS surveillance accuracies and SCE's interpretation of NRC guidance is tabulated in Enclosure 2.

PAS Program

To ensure that the PASS is operable when needed, a complex and detailed PAS Program is being developed in parallel with the PASS. The program at SONGS has taken additional steps beyond NRC regulatory requirements. It includes a precise definition of PASS operability. In addition, it includes the identification of alternative capabilities for obtaining PASS parameters that can be utilized to obtain information to assess core damage in the event that portions of PASS are inoperable.

In the development of the PAS Program, 40 procedures have been identified that address the PAS Program, operation of PASS for sampling and analyzing RCS and containment atmosphere, maintenance of PASS and surveillances to demonstrate the operability of the PASS and specify alternative methods of parameter data gathering in the event a portion of the PASS fails a routine surveillance. Of the 40 procedures, only 9 remain to be finalized and issued. Enclosure 3 is a memorandum dated April 4, 1983 that summarizes the Program and provides the status of each of the procedures. The final aspect of the PAS Program is the Training Program as described in Reference B.

SONGS Procedure S0123-III-8.1, "Post-Accident Sampling System Routine Surveillances" describes the process and frequency of routine surveillances necessary to ensure operability of the system. SCE was asked by NRC staff to include in this letter its definition of PASS operability and the basis for that definition. Accordingly, Enclosure 4 is responsive to that request.

Enclosure 1 to Reference B provided a discussion of alternative post-accident sampling capability. Enclosure 5 to this letter provides information requested by NRC staff to supplement the previously submitted information on alternative capabilities.

The NRC staff requested that the Core Damage Assessment Procedure be in place before SONGS 2 is returned to power. As noted in Enclosure 2 to Reference B and Enclosure 3 to this letter, the Core Damage Assessment Procedure is being developed and targeted for completion by May 30, 1983. The SONGS 2&3 Core Damage Assessment Procedure is based upon the Draft Interim Procedure Guidelines for Core Damage Assessment prepared by Combustion Engineering, Inc. (CE) for the CE Owners Group. The interim guidelines are for a core damage assessment based on radiological analysis of samples obtained from the reactor coolant, containment building sump, and the containment building atmosphere. The interim guidelines provided by CE are being made plant specific for SONGS 2 and 3. The latest draft of the SONGS 2 and 3 procedure is enclosed for your information as Enclosure 6. It could be used by SCE corporate support personnel in its present form for assessment of core damage at SONGS 2 and 3 if an accident were to occur. It is currently in the process of being reformatted to make the procedure consistent with the other SCE procedures and being reviewed by all appropriate SCE organizations. It is expected that the SONGS 2 and 3 interim Core Damage Assessment Procedure will be issued by May 30, 1983. SCE considers that the procedure in its present form meets the NRC staff's intent that SONGS 2, prior to return to power, have the capability to use PASS information for core damage assessment.

The second phase of the CE procedure guidelines will determine core damage assessment based on a comprehensive evaluation of data on plant condition. CE has not yet completed the comprehensive guidelines. As such a target date of February 15, 1984 has been established by SCE for issuance of the comprehensive Core Damage Assessment Procedure.

NRC criteria for post-accident sampling requires that 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. As demonstrated to NRC staff on April 5, 1983, SONGS 2 and 3 complies with that criteria. As described in Reference C, the key to that compliance is agreement that prior to the decision to take a sample certain activities shall be completed to insure adequate preparation and handling of the radioactive sample media. The activities include:

1. Assemble chemistry personnel with implementation of health physics requirements including anti-contaminant clothing and personnel dosimetry.

2. Dispatch personnel to the PASS Sample Laboratory employing post accident precautions.

3. Verify a safe atmosphere in the PASS Sample Laboratory which is considered to be an enclosed space.

4. Conduct health physics survey of the PASS Sample Laboratory.

5. Energize and verify the PASS status.

6. Select, by procedure, the initial sample point and sample media.

The time to perform the above tasks is estimated to take about 4 hours. Another prerequisite is that two PASS operators are available to conduct the above prerequisite activities and perform the post-accident sampling and analysis. The training program discussed in Reference B provides in the short term a sufficient number of qualified SCE personnel to assure that at least two could be called to report in a timely manner following an accident. SCE does not consider it necessary to have personnel qualified to operate PASS on shift at all times or on-call able to report within one hour. The basis for this position is that PASS is not a system to be immediately utilized. Most accident scenarios develop over a period of hours. A PASS sample cannot be taken until the dynamics of an accident stabilize. If the reactor chemistry is not stable, an accurate analysis of the sample will not be obtained. During this period of time, operators are not without data to assist in the assessment of the accident. For example, alternate capabilities, such as containment hydrogen monitors, high range in containment area radiation monitors, subcooled margin monitor and core exit thermocouples, are available as discussed in Enclosure 5. The interim SONGS 2 and 3 Core Damage Assessment Procedure is being revised to additionally provide for an assessment of core damage utilizing data available to the operators exclusive of the PASS.

The PASS demonstration and meetings conducted April 4-7 are further evidence of the willingness and determination of SCE to do all that is necessary to achieve a reliable PASS that will perform its intended function when called upon following an accident. Therefore, SCE considers it most appropriate to issue the license amendments requested in Reference A.

If you have any questions on this matter, please call me.

Very truly yours,

Robert Dietch

Enclosures

cc: J. B. Martin, Regional Administrator, Region V
H. Rood (To be opened by addressee only)

RESULTS OF PASS DEMONSTRATION
CONDUCTED APRIL 5, 1983

PARAMETER	PASS	CHEM LAB	MEASURED DEVIATION	TOLERANCE
RCS pH	5.9	5.89	0.01pH Unit	$\pm$0.3pH Unit
RCS Boron	1850ppm	1755ppm	+ 5.4%	$\pm$5%
RCS Total Gas	170cc/kg	188cc/kg	-9.6%	$\pm$20%
RCS O_2	0.15%*	0.0%	0*	None Specified
RCS H_2	15.28 cc/kg**	16.0 cc/kg	0.72 cc/kg**	$\pm$5cc/kg
RCS Radio-nuclides	Calibration Source Used	N/A	N/A	N/A
Containment H_2	0.0%	0.0%	0	$\pm$20%
Containment Radionuclides	Calibration Source Used	N/A	N/A	N/A

* Data listed is from second sample, 0.15% is the amount of O_2 present in the nuclear service water used for flushing.

**Data listed is from second sample. The most likely cause of the discrepancy in the H_2 analysis is the time difference between PASS and Chem. Lab samples (approximately 3 hours) and the probability that equilibrium H_2 conditions did not exist in the RCS.

DLC:7770

SURVEILLANCE CRITERIA

The attached table provides the specific surveillance criteria proposed by SCE for implementation of the NRC requirements on accuracies of the instrumentation employed in the Post-Accident Sampling System. The surveillance criteria stated in this table agree directly with the NRC guidance for instrument accuracies. A basis is stated for the case of each instrument minimum sensitivity. The subject surveillance criteria will be implemented into the definition of Plant Accident Sampling System through established station procedures.

PASS Function	Proposed Surveillance Criteria	Basis
pH	± 0.3 pH units between pH 5 to 9 ± 0.5 pH units in all other ranges	Defined by NRC guidance.
Boron	±5 percent of the comparative laboratory measurement but no less than ± 100 ppm.	The range for boron measurement is 0 to 6000 ppm. The important value is the boron concentration required for reactor shutdown which is about 1800 ppm. Five percent of this value is approximately 100 ppm. Therefore, the value 100 ppm was selected as the minimum sensitivity and is consistent with steady state boron concentrations under normal operations.
Total dissolved gas	± 20 percent of the comparative laboratory measurement but no less than ± 100 cc/kg.	The range for this measurement is 0 to 2000 cc/kg. A mid-range value which may be anticipated under post accident degraded core conditions is 500 cc/kg. Twenty percent of this value is 100 cc/kg. Therefore, the value 100 cc/kg is employed as the minimum sensitivity. The value of 100 cc/kg is consistent with values of total gas found under normal operating conditions. It is not anticipated that the plant will operate below 50 cc/kg.
Oxygen gas	No surveillance required	This instrument is recommended but not required by NUREG-0737. No surveillance will be implemented in terms of the definition of PASS operability. However, standard maintenance practices will be employed to keep this instrument operable.

Hydrogen gas	$\pm$ 20 percent of the comparative laboratory measurement but no less than a minimum sensitivity of 25 cc/kg.	The range of this measurement is 0 to 2000 cc/kg which is 100 percent of the range of total gas. The minimum sensitivity of 25 cc/kg is employed because this value is the minimum required for RCS hydrogen concentration under normal plant operations. It is not anticipated that the plant will operate below this value.
Containment Atmosphere Hydrogen	$\pm$ 20 percent of the comparative laboratory measurement but no less than a minimum sensitivity of 1 percent hydrogen concentration.	No specific NRC criterion is defined for the accurary of the containment hydrogen measurement. The accuracy stated is consistent with the reactor coolant hydrogen measurement. The minimum sensitivity is below the lower limit for flamability of a hydrogen gas mixture in air.
Gross Activity Gamma Spectrum	Accurate within a factor of two across the entire range of 10 Ci/ml to 10 Ci/ml.	Defined by NRC guidance.

DLC:7770

Enclosure 3

POST-ACCIDENT SAMPLING SYSTEM
SUMMARY OF PROCEDURES & PROGRAM

April 4, 1983

I. PROCEDURES

A total of 40 procedures have been identified as components of the PASS Program. Of the total, 9 remain to be finalized and issued. The target date for the issuance of all Station documents is May 1, 1983. Two important areas have significant work remaining: procedure(s) for the assessment of the extent of core damage and obtaining, handling and analysis of radioactive iodine and particulate samples.

II. PROGRAM

With the exception of the items noted above, the PASS Procedure Program has been substantially completed. The procedures encompass the programmatic requirements of the current license condition and Technical Specification 6.8.4.d for personnel training, equipment operation and maintenance. Additional depth has been added to the Program by making provision for a definition of system operability, routine system surveillances and the implementation of alternative sampling methods. The essence of the Program is contained in four procedures:

A. SO123-G-19, "Post-Accident Sampling System Program." Describes the Program in general terms and assigns programmatic support responsibilities to the various organizations.

B. SO123-III-8.0, "Post-Accident Sampling Program and Analytical Requirements." Assigns responsiblity within the primary organization (Chemistry Department), describes the sampling requirements and establishes the prerequisites of the sampling process.

C. SO123-III-8.1, "Post-Accident Sampling System Routine Surveillances." Describes the process and frequency of routine surveillances necessary to ensure operability of the system. The system will be considered operable when the conditions of the procedure are met, including actions to be taken in the event of component failure.

D. SO123-III-8.8, "Alternate Methods of Post-Accident Parameter Sampling." Describes in detail the alternate methods of post-accident parameter sampling available for use in the event of a PASS component failure.

An additional document describing the usage of all input data in evaluating the extent of damage to the core under accident conditions is under preparation of Nuclear Engineering. An interim version will be available for use by May 30, 1983. The final version, which will incorporate both generic and plant specific input from Combustion Engineering, is scheduled for completion by February 15, 1984.

POST-ACCIDENT SAMPLING PROGRAM PROCEDURES

Status as of April 4, 1983

A. CHEMISTRY PROCEDURES

Item	Procedure Number	Description	Revision In Place	Revision Forecast
1.	S0123-III-8.0	PASS Sampling Program and Analytical Requirements	Rev. 2 3/29/83	Rev. 3 5/1/83
2.	S0123-III-8.1	PASS Routine Surveillances	Rev. 0 3/29/83	Rev. 1 5/1/83
3.	S0123-III-8.2.23	Startup and Fill of PASS	Rev. 4 4/4/83	Not Anticipated
4.	S0123-III-8.3.23	Sampling Procedures and In-Line Analysis for PASS	Rev. 4 4/4/83	Not Anticipated
5.	S0123-III-8.4.23	Purging and Refilling of the PASS	Rev. 4 4/4/83	Not Anticipated
6.	S0123-III-8.5.23	Chemistry Calibration Procedure for PASS	Rev. 1 3/9/83	Not Anticipated
7.	S0123-III-8.6.23	Access to the PASS During Accident Conditions	Rev. 0 12/3/82	Not Anticipated
8.	S0123-III-8.7	Operation and Calibration of PASS Spectrometer	Rev. 3 3/29/83	Not Anticipated
9.	S0123-III-8.8	Alternate Methods of Post-Accident Sampling	Rev. 0 3/29/83	Rev. 1 5/1/83
10.	S0123-III-8.9	Radioactive Iodine Sampling Under Accident Conditions	Not Issued	Rev. 0 5/1/83
11.	S0123-G-19	PASS Program	Not Issued	Rev. 0 5/1/83

B. MAINTENANCE PROCEDURES

Item	Procedure Number	Description	Revision In Place	Revision Forecast
1.	S023-I-8.130	PASS Semi-Annual Preventative Maintenance	Rev. 0 3/30/83	Not Anticipated
2.	S023-I-8.131	PASS 18-Month Preventative Maintenance	Rev. 0 3/30/83	Not Anticipated
3.	S023-I-8.132	Refueling Interval PASS Air Cleanup System Charcoal Absorber Testing	Rev. 0 3/30/83	Not Anticipated
4.	S023-I-8.133	Refueling Interval PASS Air Cleanup System HEPA Filter Testing	Rev. 0 3/30/83	Not Anticipated
5.	Not Assigned	Maintenance of PASS Heat Tracing	Not Issued	Rev. 0 5/1/83

C. HEALTH PHYSICS PROCEDURES

Item	Procedure Number	Description	Revision In Place	Revision Forecast
1.	Not Assigned	Handling and Shipping of Offsite Chloride Sample	Not Issued	Rev. 0 5/1/83
2.	Not Assigned	Onsite Handling, Transfer and Storage of PASS Samples	Not Issued	Rev. 0 5/1/83
3.	Not Assigned	Handling and Shipping of WRGM Filter Offsite	Not Issued	Rev. 0 5/1/83

D. INSTRUMENT & CONTROL PROCEDURES

Item	Procedure Number	Description	Revision In Place	Revision Forecast
1.	S023-II-4.45	PASS Area Radiation Monitors Channel Calibration	Rev. 0 7/22/82	Not Anticipated
2.	S023-II-8.10	Loop Verification	Rev.6 2/11/83	Not Anticipated
3.	S023-II-9.361	Containment High Range Area Radiation Monitor Calibration	Rev. 4 2/10/83	Not Anticipated
4.	S023-II-9.362	Area Radiation Monitor Readout Calibration	Rev. 3 10/9/82	Not Anticipated
5.	S023-II-9.363	Area Radiation Monitoring System Calibration	Rev. 2 10/16/82	Not Anticipated
6.	S023-II-8.772	PASS Instrumentation Loops Calibration	Rev. 0 3/21/83	Not Anticipated
7.	S023-II-9.191	Sigma Indicator Model 9263 Calibration	Rev. 2 1/14/83	Not Anticipated
8.	S023-II-9.384	Sigma Boron Meter Converter Calibration	Rev. 0 6/11/82	Not Anticipated
9.	S023-II-9.382	PASS Liquid and Gaseous Flowmeter Calibration	Rev. 0 6/5/82	Not Anticipated
10.	S023-II-9.10	Rosemount Differential Pressure Transmitter Calibration	Rev. 3 7/16/82	Not Anticipated
11.	S023-II-9.37	Pneumatic Valve Calibration	Rev. 3 12/16/82	Not Anticipated
12.	S023-II-9.351	Fischer Porter Manual Station Calibration	Rev. 0 5/21/82	Not Anticipated
13.	S023-II-12.446	PASS Instrumentation Calibration	Rev. 0 3/18/83	Not Anticipated
14.	S023-II-9.383	Beckman pH Analyzer Calibration	Rev. 0 6/7/82	Not Anticipated

D. INSTRUMENT & CONTROL PROCEDURES (Continued)

Item	Procedure Number	Description	Revision In Place	Revision Forecast
15.	S023-II-9.381	Delphi Thermal Conductivity Analyzer Calibration	Rev. 0 6/5/82	Not Anticipated
16.	S023-II-9.82	Pressure Switch Calibration	Rev. 0 6/5/82	Not Anticipated
17.	S023-II-9.380	Delpha Paramagnetic Oxygen Analyzer Calibration	Rev. 0 6/5/82	Not Anticipated
18.	S023-II-9.183	Thermon Temperature Indicating Controller Calibration	Rev. 1 6/3/82	Not Anticipated

E. OTHER DOCUMENTS

Item	Description	Forecast
1.	Core Damage Assessment (SCE Nuclear Engineering)	Interim: 5/30/83 Final: 2/15/84
2.	Offsite Cloride Sample Analysis (General Atomic)	7/1/83
3.	Offsite Analysis of WRGM Filter (General Atomic)	Not Yet Scheduled

DEFINITION OF OPERABILITY OF THE POST-ACCIDENT SAMPLING SYSTEM

1. Routine surveillances described in Surveillance Procedure (SO123-III-8.1) are conducted at the prescribed intervals when plant conditions permit.

2. In the event of a PASS component malfunction, the specific alternate method of sampling listed in the "Alternate Methods of Post-Accident Parameter Sampling" procedure (SO123-III-8.8) is available and measures are being taken to effect repairs to the component that has malfunctioned.

3. Calibration of PASS Instruments is current.

BASIS FOR DEFINITION

1. Surveillances prove that the system will provide reliable, repeatable and accurate PASS parameter information. In addition, routine surveillances provide training opportunities.

2. The alternate methods procedure refers to techniques available to demonstrate the ability to obtain PASS parameter information when PASS components malfunction. The procedure also indicates the need to effect repairs on the PASS component that has malfunctioned.

3. A routine check of the PASS instrument calibration program ensures instruments are calibrated and available when PASS is needed under accident conditions.

DLC:7770

ALTERNATIVE POST ACCIDENT SAMPLING CAPABILITY

PASS FUNCTION	ALTERNATIVE CAPABILITY
I. Containment Atmosphere	
A. Hydrogen	An alternate method, independent of PASS, for determining containment atmosphere hydrogen levels is by means of hydrogen monitors inside containment. The containment hydrogen monitors are Seismic Class 1, Quality Class II and NUREG-0588 qualified Class 1E instruments. These two instruments provide redundant, channelized continuous readout in the control room of hydrogen levels inside containment from 0 to 10 volume percent. The hydrogen monitors are located inside containment at elevation 76' on the outside of the secondary shield. This location provides the hydrogen sensor with access to the upper regions of the containment building and allows optimum monitoring of the containment atmosphere while maintaining accessibility in the event of sensor malfunction. The limiting conditions for operation of these monitors is delineated in Technical Specification Section 3/4.6.4. Buildup of hydrogen inside the containment following an accident is caused primarily by the reaction between the Zirconium fuel cladding and the reactor coolant. FSAR Figure 6.2-44 (Attachment (1)) shows the anticipated hydrogen buildup inside containment. The hydrogen monitor system tracks and records this hydrogen buildup. The operator uses the system to energize the hydrogen recombiners at 2 volume percent hydrogen in containment approximately two days into the accident. However, the hydrogen recombiners will not begin to remove hydrogen until concentrations reach 3-1/2 volume percent which occurs approximately two weeks into the accident. The system alarms in the control room when 3 volume percent hydrogen is reached inside the containment after

PASS FUNCTION	ALTERNATIVE CAPABILITY
	approximately 8 days. The rate of hydrogen buildup for a PWR causes the hydrogen lower combustible limit (4 volume percent) to be reached in approximately 21 days. This is significantly less rapid than the hydrogen buildup from a BWR whose drywell can experience up to a 6 volume percent level within one and one half minutes following the accident. The hydrogen monitors at San Onofre have a scale of 0 to 10 volume percent with an accuracy of ± 5 percent full scale reading. The specified response time for the monitors is 90 percent of reading in two hours based upon a 4 percent step change in hydrogen concentration. The vendor ran a test evaluating the response time capability of a monitor identical to those installed at San Onofre to track changes in hydrogen concentration. The recorder trace showed the monitors response to a 2.06 percent step increase, a gradual reduction to 2.04 percent and a step reduction to 1.7 percent hydrogen. The curve showed a 90 percent reading in approximately 30 minutes for the 2.06 percent step increase and virtually immediate response to the reduction to 1.7 percent. This response is more than adequate to track and record changes in post-LOCA hydrogen generation.
B. Radionuclide	The normal method of obtaining a containment grab sample could be used for most of the FSAR Chapter 15 postulated accidents because radiation levels would not preclude these samples from being taken or processed in the normal sample laboratory. The PASS provides two methods of determining containment atmosphere radionuclide concentrations. An intrinsic germanium multi-channel analyzer is provided to measure post accident containment airborne samples. In addition, the PASS is provided with the capability to take a diluted containment airborne sample. This diluted sample could be measured using onsite laboratory equipment.

PASS FUNCTION	ALTERNATIVE CAPABILITY
	An alternate method, independent of PASS, for determining containment atmosphere radiation levels is by means of high range in-containment detectors. These detectors are Seismic Class 1, Quality Class II and NUREG-0588 qualified Class 1E instruments. These two instruments provide redundant, channelized continuous readout, immediately adjacent to the main control room panels, of radiation levels inside containment from 10^0 to 10^8 R/hr.
	As shown on Attachment (2), the high range in-containment monitors are located at elevation 94' adjacent to the polar crane access ladder and on the secondary shield wall at elevation 99'. These locations permit the detector to "see" the containment atmosphere without being obstructed by concrete shield walls. The detector is designed to minimize attenuation of low energy gamma. The limiting condition for operation of these monitors is delineated by Technical Specification Tables 3.3-6 and 3.3-10.
	Correlations between the radiation readings for the high range incontainment monitor and accident releases have been developed. The basis of the correlation is provided in Attachment (3). This attachment provides the accident definitions and associated releases into the containment free volume. Attachment (4) provides a graph showing the projected relationship between monitor dose rate, time after shutdown and accident release. These curves in conjunction with the radiation levels indicated on the high range in-containment detector provide an indication of degraded core conditions.
C. Diluted samples	Normal sampling is available as discussed for Radionuclides in I.B above for most of the spectrum for Chapter 15 accidents. However, for accidents with TMI type source terms there is no backup capability. In that this capability is a backup, no further backup capabilities are required.

PASS FUNCTION	ALTERNATIVE CAPABILITY
II. RCS Analysis	
A. Gas	
1. Hydrogen	The capability to measure the hydrogen concentration of the containment atmosphere is described in discussion I.A above. The hydrogen concentration in the containment atmosphere is considered representative of the reactor coolant hydrogen content. Reactor coolant hydrogen is produced as a result of degraded core zirconium-water oxidation reactions. This hydrogen would not be produced without a significant displacement of water in the reactor vessel to a level below the core. Such a displacement of water volume could occur only if an open path existed to the containment thereby releasing the hydrogen to the atmosphere. This process is enhanced by the presence of the Reactor Coolant Gas Vent System to release to the atmosphere any gas which remains in the vessel. The presence of non-condensible gas in the reactor vessel is additionally monitored through the 1E Qualified Reactor Vessel Water Level Monitor. Station Chemistry Procedure S0123-III-8.8 describes a second alternate procedure using analysis of grab samples obtained from the PASS or Radiochemistry Laboratory. This procedure identifies the specific laboratory instrument to be used and references Station Procedure S023-III-1.8 for use of this instrument.
2. Oxygen	Oxygen gas measurements are not a requirement of NUREG-0737.
3. Radionuclide	Use the high-range incontainment monitors. See the discussion of I.B above.
4. Total Dissolved Gas	The Seismic I, Quality Class II, IE, NUREG-0588 qualified subcooled margin monitor and the interim core exit thermocouple system for detection of ICC will provide temperature and pressure parameters necessary to determine the amount of dissolved gas which reactor coolant can retain.

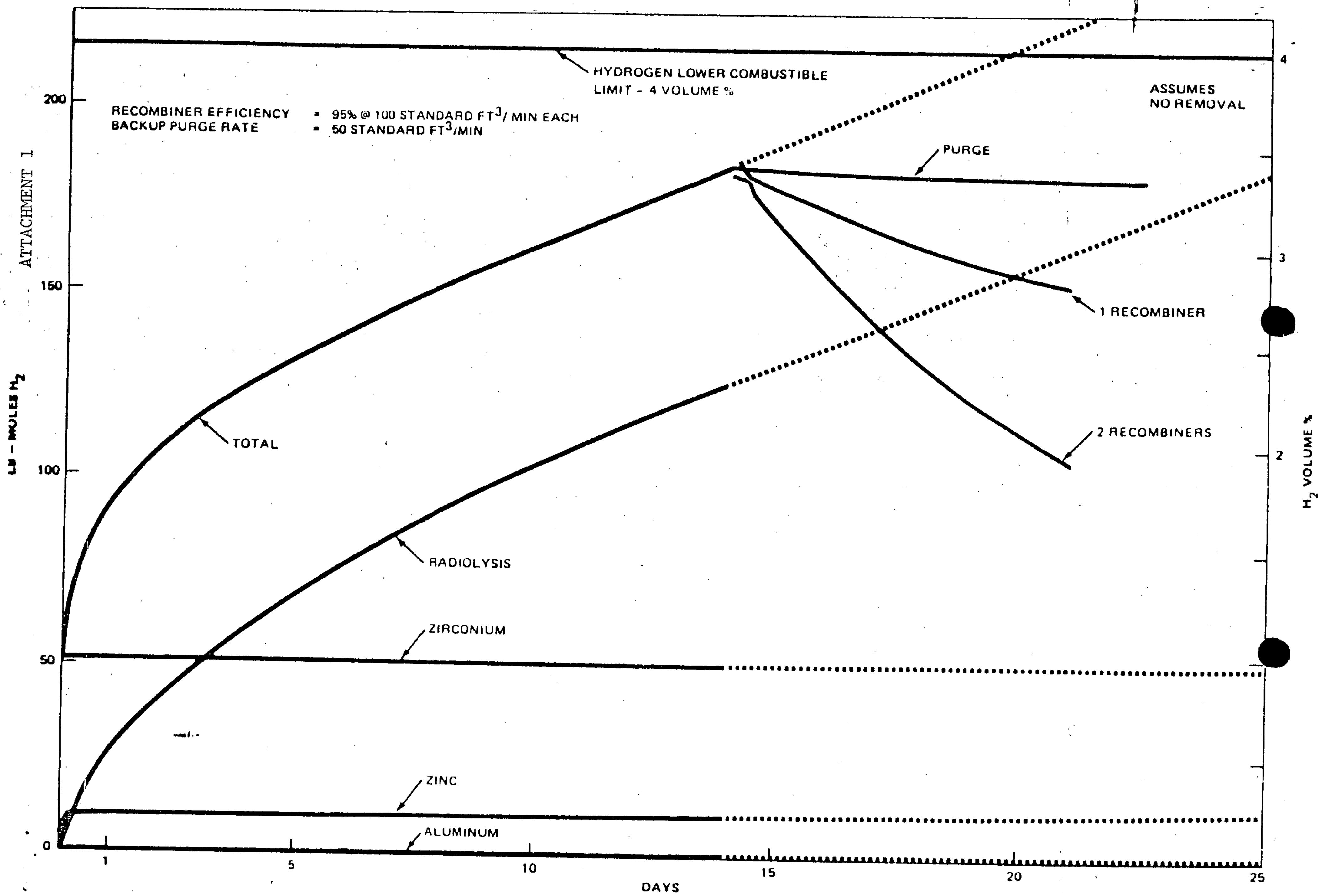
PASS FUNCTION	ALTERNATIVE CAPABILITY
	Consistent with the discussion provided for hydrogen gas content in Item II.A.1, the presence of excess total gas in the reactor coolant system is monitored by the reactor vessel water level monitor. This monitor will detect the presence of noncondensible gas voids in the reactor vessel. The Reactor Coolant Gas Vent System is available to remove such gas from the vessel. Station Chemistry Procedure S0123-III-8.8 identifies specific instrumentation which is available under post-accident conditions to monitor reactor coolant temperature and pressure. With known values of temperature and pressure the maximum dissolved gas content in the coolant may be determined using the principal of Henry's Law. Station Chemistry Procedure S0123-III-8.8 describes a second alternate method for total gas determination which employs grab samples from the Radiochemistry Laboratory. This alternate method is available for most of the spectrum of FSAR Chapter 15 accidents. The procedure identifies the instrument to be used in the analysis and references Station Chemistry Procedure S023-III-1.8 for operating procedures for that instrument.
5. Diluted Grab Sample (Backup to inline instrumentation)	In that this capability is a backup, no further backup capabilities are required.
B. Liquid	
1. Boron, pH, Radionuclide	Boron concentration is measured as a means to verify reactor shutdown. The immediate means for operator verification of this safety function is to verify the open positon on the control rod scram breakers. The backup to that action is the measure of neutron flux using either the incore or excore neutron detectors. Therefore, measurement of boron concentration is in itself a third level alternate method to verify reactor shutdown.

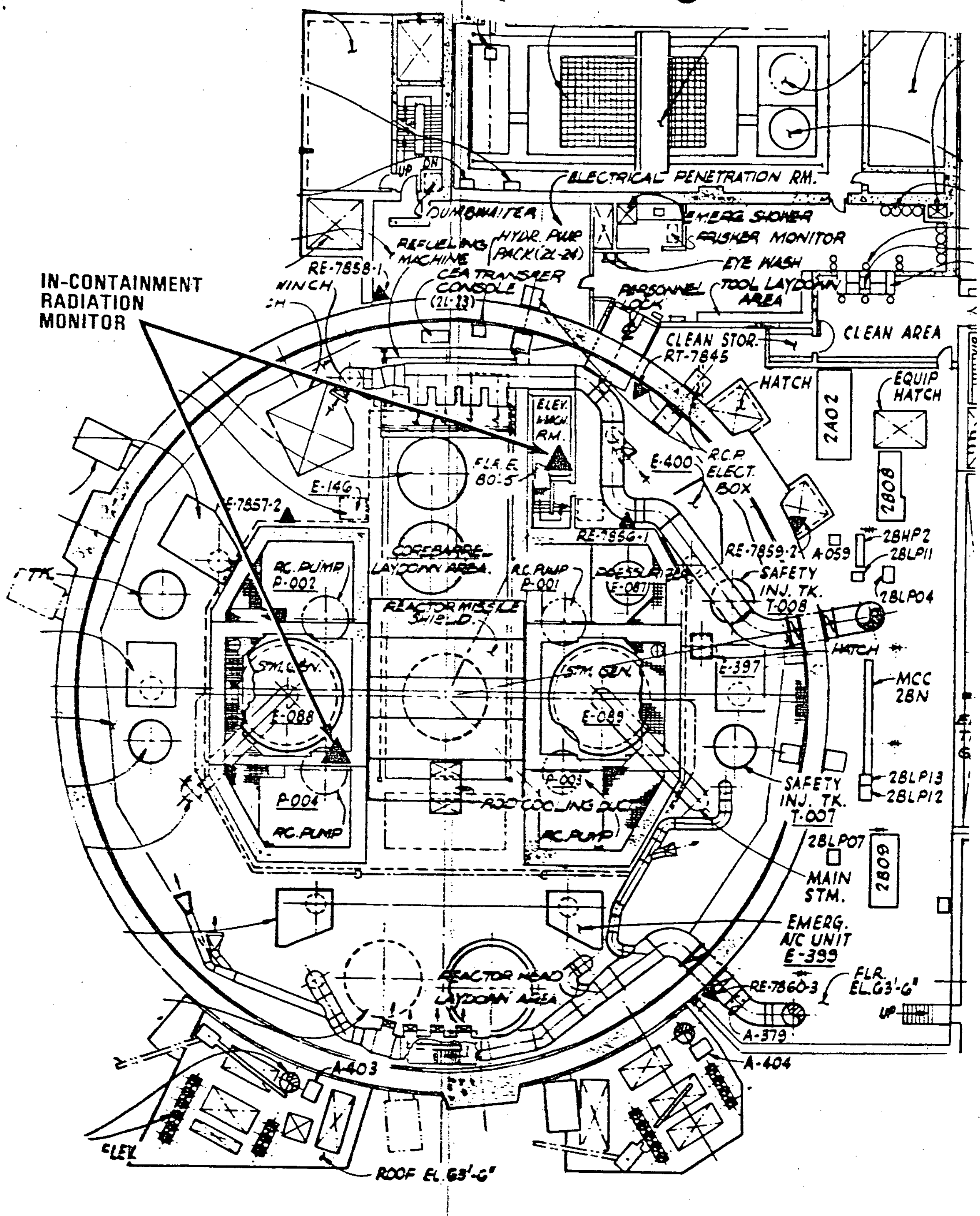
PASS FUNCTION	ALTERNATIVE CAPABILITY
	Boron concentration may be calculated by correcting the initial reactor coolant concentration prior to the accident by the amount of spray and safety injection water added during the accident. The amount of water injected and the RCS inventory can be determined from safety grade refueling water storage tank and Safety Injection Tank level indications. Station Chemistry Procedure S0123-III-8.8 describes the method to calculate the final boron concentration and dilution from safety injection. pH is an indication of the potential for long term corrosion. The potential for corrosion would be assessed through pH analysis of that sample which is obtained and stored in the shielded grab sample facility. The measurement of pH is only appropriate from an undiluted sample. The shielded grab sample facility is used to store the sample until the dose rates decay to a level consistent with ALARA. Station Chemistry Procedure S0123-III-4.8 describes the calibration and operation of the instrument to be used for this purpose. Diluted grab samples taken from the PASS can be analyzed for radionuclide concentration.
2. Chloride	Chlorides are monitored in response to the concern for long term stress corrosion. An undiluted RCS sample can be collected in an existing shielded grab sample cask and retained for chloride analysis for 30 days consistent with ALARA. The cask has been tested, delivered, and requires NRR approval for offsite shipment. In addition, a diluted grab sample can be obtained from the PASS and analyzed for chlorides and the dissolved hydrogen residual can be measured to assess the presence of oxygen as a contribution to stress corrosion.

PASS FUNCTION	ALTERNATIVE CAPABILITY
	Station Chemistry Procedures S0123-III-1.3 and S0123-III-4.7 describe the procedures for the chloride measurement of the diluted sample using water chemistry techniques and as an alternate using ion chromatography.
3. Diluted grab sample (backup to inline instrumentation)	In that this capability is a backup, no further backup capabilities are required.
III. Containment Sump	The analysis of containment sump liquids may be correlated with the RCS sample, corrected by the amount of spray and safety injection water. The amount of water injected and the RCS inventory are available from safety grade Refueling Water Storage Tank and Safety Injection Tank level indications. The activity of the RCS coolant is known from the PASS RCS liquid sample.

DLC:7770

ATTACHMENT 1





SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

HIGH RANGE RADIATION MONITORS
(UNIT 2 SHOWN)

Figure II.F.1-7

~~Enclosure (1)~~

Source Term Assumptions

1. 100% of the Average Reactor Coolant System Activity

   a) The source terms shown in FSAR Table 11.1-3 released to the containment.

2. 100% of the Maximum Reactor Coolant System Activity

   a) The source terms shown in FSAR Table 11.1-2 released to the containment.

3. 1% Failed Fuel Activity

   a) Of the 1% failed fuel, 100% of the noble gases, 50% of the halogens and 1% of the other isotopes are released to the containment.

4. 10% Failed Fuel Activity

   a) Of the 10% failed fuel, 100% of the noble gases, 50% of the halogens and 1% of the other isotopes are released to the containment.

5. 100% of the Gap Activity

   a) Based on the guidance provided in Regulatory Guide 1.25 the following core gas gap activity is released.

      1) 10% of Kr and Xe

      2) 30% of Kr-85

      3) 10% of Iodines

6. LOCA Source Term

   a) Based on guidance provided in Regulatory Guide 1.4 the following fractions of core inventory are released.

      1) 100% of noble gases

      2) 50% of halogens

      3) 1% other isotopes

   b) Of the 50% halogens half of it is considered to be airborne. The other half is considered to be plated on containment surfaces.

Table 11.1-3
AVERAGE REACTOR COOLANT RADIOISOTOPE CONCENTRATION[a]
(No Gas Stripping)

Nuclide	Activity ($\mu Ci/cm^3$)[b]	Nuclide	Activity ($\mu Ci/cm^3$)[b]
H-3	9.7(-1)[c]		
Br-84	2.65(-3)	I-134	4.88(-2)[c]
Kr-85m	2.20(-1)	Cs-134	4.03(-2)
Kr-85	1.50(-1)	I-135	2.26(-1)
Kr-87	6.00(-2)	Cs-136	2.03(-2)
Kr-88	2.00(-1)	Cs-137	2.90(-2)
Rb-88	2.02(-1)	Xe-131m	1.10(-1)
Sr-89	5.95(-4)	Xe-133	1.80(+1)
Sr-90	1.71(-5)	Xe-135	3.50(-1)
Y-90	2.64(-5)	Xe-135m	1.30(-2)
Y-91	3.42(-2)	Xe-138	4.40(-2)
Y-91m	3.91(-4)	Ba-140	3.67(-4)
Sr-91	8.06(-4)	La-140	2.22(-4)
Mo-99	6.98(-1)	Pr-143	8.35(-5)
Ru-103	7.70(-5)	Ce-144	5.64(-5)
Ru-106	1.71(-5)	Cr-51	3.36(-3)
Te-129	1.66(-3)	Mn-54	5.48(-4)
I-131	4.56(-1)	Zr-95	1.03(-4)
I-132	1.08(-1)	Co-60	3.54(-3)
Te-132	4.24(-2)	Fe-59	1.76(-3)
I-133	5.28(-1)	Co-58	2.82(-2)

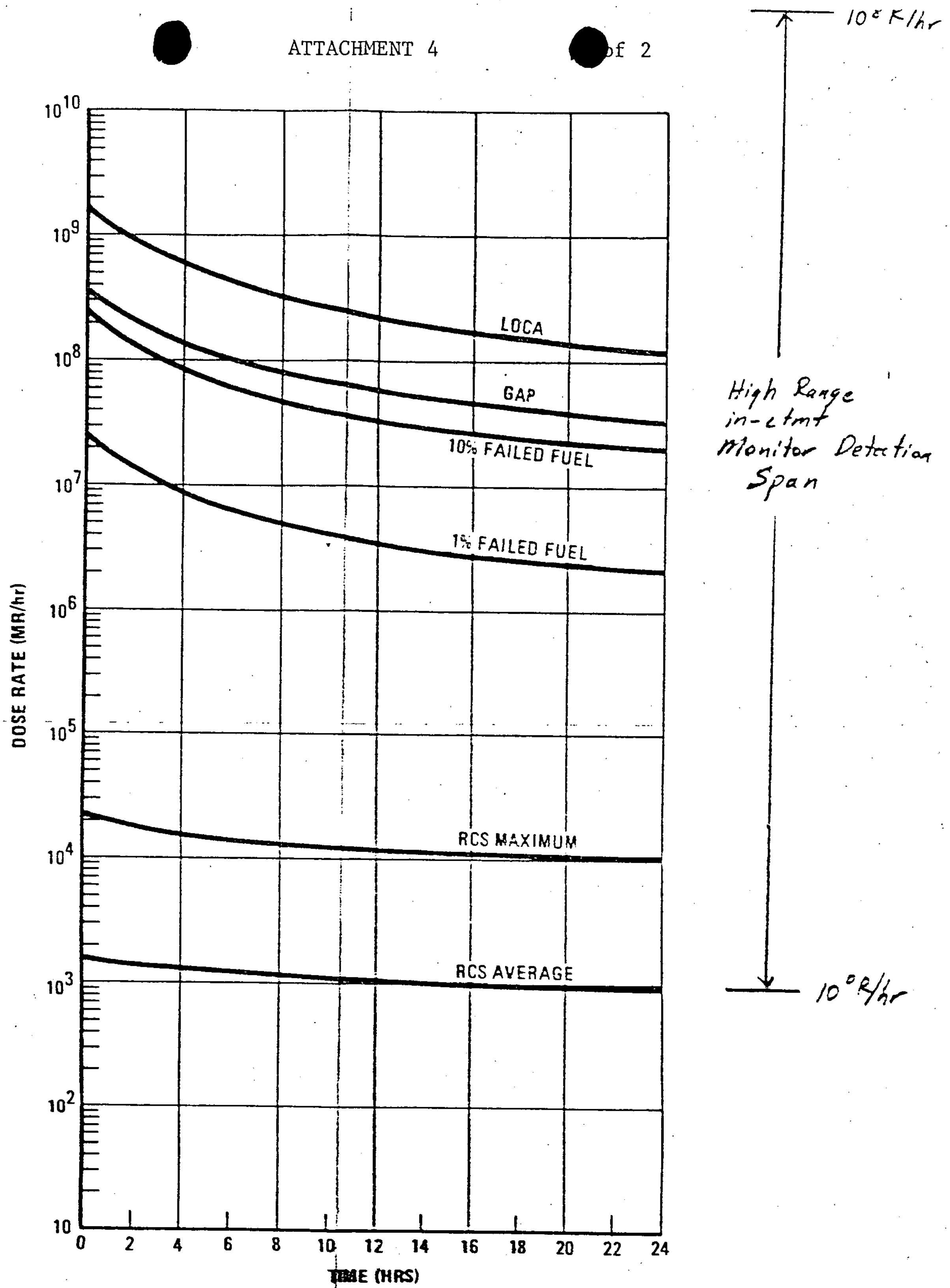
a. From draft N237 Standard
b. At 70F
c. ( ) denotes power of ten

Table 11.1-2
MAXIMUM REACTOR COOLANT RADIOISOTOPE CONCENTRATION
ONE PERCENT FAILED FUEL, NO GAS STRIPPING

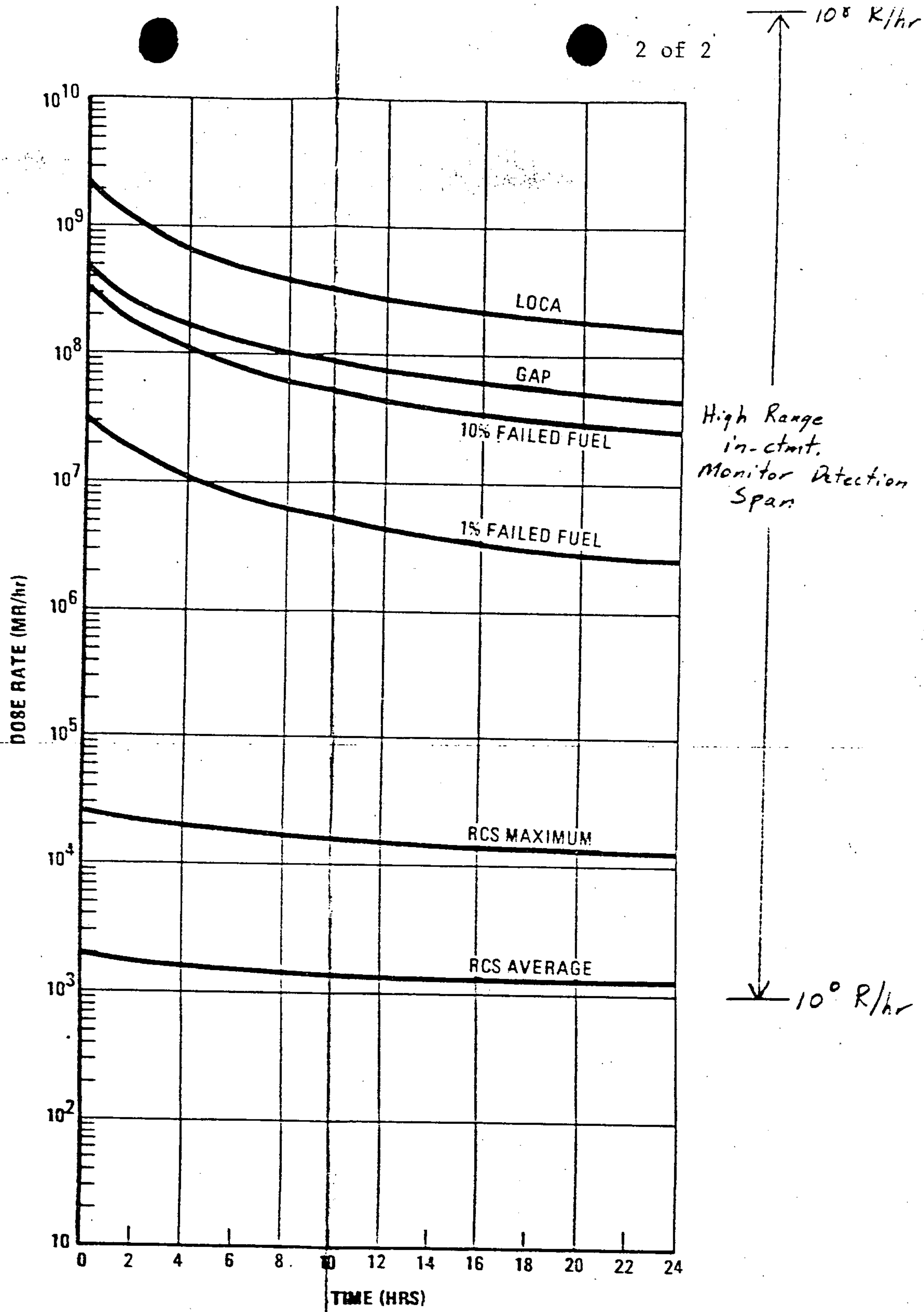
Nuclide	Activity ($\mu Ci/cm^3$)(a)	Nuclide	Activity ($\mu Ci/cm^3$)(a)
H-3	1.8(0)(b)		
Br-84	3.97(-2)	I-134	5.73(-1)(b)
Kr-85m	2.35(0)	Cs-134	1.53(-1)
Kr-85	5.13(0)	I-135	2.58(0)
Kr-87	1.26(0)	Cs-136	2.64(-2)
Kr-88	4.08(0)	Cs-137	6.20(-1)
Rb-88	4.06(0)	Xe-131m	2.39(0)
Sr-89	1.07(-2)	Xe-133	3.34(2)
Sr-90	5.79(-4)	Xe-135m	1.10(0)
Y-90	1.42(-3)	Xe-135	9.24(0)
Y-91	4.84(-2)	Xe-138	5.58(-1)
Y-91m	3.85(-3)	Ba-140	1.27(-2)
Sr-91	6.15(-3)	La-140	1.23(-2)
Mo-99	2.24(0)	Pr-143	1.13(-2)
Ru-103	8.53(-3)	Ce-144	7.82(-3)
Ru-106	5.12(-4)	Cr-51	3.36(-3)
Te-129	4.98(-2)	Mn-54	5.48(-4)
I-131	4.67(0)	Zr-95	1.15(-2)
I-132	1.32(0)	Co-60	3.54(-3)
Te-132	6.48(-1)	Fe-59	1.76(-3)
I-133	5.88(0)	Co-58	2.82(-2)

21

a. At 70F
b. ( ) denotes power of ten



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
DOSE RATE VS TIME FOR HIGH RANGE IN-CONTAINMENT MONITOR AT EL. 94' ABOVE THE ELEVATOR SHAFT
FIGURE 432.42-2



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

DOSE RATE VS TIME FOR
HIGH RANGE IN-CONTAINMENT
MONITOR AT EL. 99'6" ON
SECONDARY SHIELD WALL

FIGURE 432.42-1

SAN ONOFRE 2

INTERIM PROCEDURE

FOR CORE DAMAGE ASSESSMENT

DRAFT

TABLE OF CONTENTS

LIST OF ENCLOSURES

1.0 PURPOSE

A basic procedure is provided for estimating post-accident reactor core damage by using fission product isotopes measured in samples obtained from the Post-Accident Sampling System (PASS). There are three factors considered in this procedure which are related to the specific activity of the samples. These are the identity of those isotopes which are released from the core, the respective ratios of the specific activity of those isotopes, and the percent of the source inventory at the time of the accident which is observed to be present in the samples. The resulting estimate of core damage can be related to one or more of the ten categories described in Enclosure 1. Reference 2.1 contains background information for the basic procedure.

An alternate procedure is provided for situations in which use of the basic procedure is precluded for some reason. The alternate procedure uses containment radiation measurements as a gross indication of reactor core damage. (Note that the alternate procedure may be used in addition to, as well as instead of, the basic procedure.) Reference 2.2 presents the analysis on which the alternate procedure is based.

2.0 REFERENCES

2.1 Development of the Interim Procedure Guidelines for Core Damage Assessment, CE Owners Group Task 467, January 1982.

2.2 Study for Fuel Failure/Fuel Melting and Radiation Monitor Responses for San Onofre Nuclear Generating Station (SONGS) Unit 2," Quadrex Corporation, July 1983.

2.3 S0123-III-8.0 "Post-Accident Sampling Program and Analytical Requirements."

3.0 DEFINITIONS

3.1 Fuel Damage: For the purpose of this procedure fuel damage is defined as a progressive failure of the material boundary to prevent the release of radioactive fission products into the reactor coolant starting with a penetration in the zircaloy cladding. The type of fuel damage as determined by this procedure is reported in terms of four major categories which are: no damage, cladding failure, fuel overheat, and fuel melt. Each of these categories are characterized by the identity of the fission products released, the mechanism by which they are released, and the source inventory within the fuel rod from which they are released. The degree of fuel damage is measured by the percent of the fission produce source inventory which has been released into fluid media and therefore available for immediate release to the environment. The degree of fuel damage as determined by this procedure is reported in terms of three levels which are: initial, intermediate, and major. This results in a total of ten possible categories as characterized in Enclosure 1.

3.2 Source Inventory: The source inventory is the total quantity of fission products expressed in curies of each isotope present in either source; the fuel pellets or the fuel rod gas gap.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 The assessment of core damage obtained by using this procedure is only an estimate. The techniques employed in this procedure are adequate only to locate the core condition within one or more of the 10 categories of core damage described in Enclosure 1. The procedure is based on radiological data. Other plant indications may be available which can improve upon estimation of core damage. These include incore temperature indicators, the total quantity of hydrogen released from zirconium degradation and containment radiation monitors. Whenever possible these additional indicators should be factored into the assessment.

4.2 This procedure relies upon samples taken from multiple locations inside the containment building to determine the total quantity of fission products available for release to the environment. The amount of fission products present at each sample location may be changing rapidly due to transient plant conditions. Therefore, it is required that the samples should be obtained within a minimum time period and if possible under stabilized plant conditions. Samples obtained during rapidly changing plant conditions should not be weighed heavily into the assessment of core damage.

4.3 A number of factors influence the reliability of the chemistry samples upon which this procedure is based. Reliability is influenced by the ability to obtain representative samples due to incomplete mixing of the fluids, equipment limitations, and lack of operator familiarity with rarely used procedures. The accuracy achieved in the radiological analyses are also influenced by a number of factors. The equipment employed in the analysis may be subjected to high levels of radiation exposure over extended periods of time. Chemists are requied to exercise considerable caution to minimize the spread of radioactive materials. Samples have the potential of being contaminated by numerous sources and they may not represent the average distribution of the contaminants in the sampled fluid. Cooling or reactions may take place in the long sample lines. Therefore, the results obtained may not be representative of plant conditions. To minimize these effects multiple samples should be obtained over an extended time period from each location.

5.0 PLANT CONDITION/SYMPTOMS

This procedure is to be employed for analysis of radiochemistry sample data when it is determined that a plant accident with the potential for core damage has occurred. The following is a list of plant symptoms to assist in this determination. This list is not a complete representation of all events or conditions which may indicate potential core damage. However, the existance of one or more of these events or conditions signals a possible need to activate this procedure.

5.1 High alarm on the containment radiation monitor.

5.2 High alarm on the CVCS letdown radiation monitor.

5.3 High alarm on the main condenser air ejector exhaust radiation monitor.

5.4 Pressurizer level low.

5.5 Safety Injection System may have automatically actuated.

5.6 Possible high quench tank level, temperature, or pressure.

5.7 Possible noise indicative of a high energy line break.

5.8 Decrease in volume control tank level.

5.9 Standby charging pumps energized.

5.10 Unbalanced charging and letdown flow.

5.11 Reactor Coolant System subcooling low or zero.

6.0 PREREQUISITES

An operational Post-Accident Sampling System with the capability to obtain and analyze the identity and concentration of fission product isotopes in accordance with the provisions of Reference 2.3.

7.0 BASIC PROCEDURE

7.1 Record of Plant Condition

Record the following plant indications using Enclosure 2 as a worksheet. The values should be recorded as close as possible to the time at which the radiological samples are obtained from the Post-Accident Sampling System. If additional samples are taken at a later time, record another set of values at or near that time.

7.1.1 Reactor Coolant System:

Pressure

Temperature

Reactor Vessel Level

Pressurizer Level

7.1.2 Containment Building:

Atmosphere Pressure

Atmosphere Temperature

Sump Level

7.1.3 Prior 30 Days Power History

7.1.4 Time of Reactor Shutdown

7.2 Selection of Sample Location

Obtain specific activity data from samples of the reactor coolant, the containment sump water, and the containment atmosphere.

7.3 Sample Recording

Record the required data for each sample. Enclosure 3 is provided as a worksheet.

Some of the isotopes listed in the enclosure may not be observed in the sample.

7.4 Temperature and Pressure Correction

Correct the measured sample specific activity to standard temperature and pressure.

7.4.1 Reactor coolant liquid samples are corrected for temperature using the factor for water density from Enclosure 4. The measured value of specific activity is divided by the correction factor corresponding to the sample temperature from Enclosure 3. This corrected value of specific activity is recorded in Enclosure 5.

7.4.2 Containment building sump samples do not require correction for temperature and pressure within the accuracy of this procedure.

7.4.3 Containment building atmosphere samples are corrected for temperature and pressure using the following equation.

$$\text{Specific Activity (STP)} = \text{Specific Activity} \times \left(\frac{P_2}{P_1 + P_2}\right) \times \left(\frac{T_1 + 460}{T_2 + 460}\right)$$

where:

T_1, P_1 = Measured sample temperature and pressure recorded on Enclosure 3

T_2, P_2 = Standard Temperature, 32^{o}F and Standard Pressure, 14.7 psia.

Record the corrected values of specific activity on Enclosure 5.

7.5 Decay Correction

Correct the sample specific activity for decay back to the time of reactor shutdown using the following equation. Enclosure 6 is provided as a worksheet.

$$A_o = \frac{A}{e^{-\lambda t}}$$

where:

A_o = the specific activity of the sample corrected back to the time of reactor shutdown, $\mu ci/cc$.

A = the measured specific activity, $\mu ci/cc$.

λ = the radioactive decay constant, 1/sec.

t = the time period from reactor shutdown to sample analysis, sec.

7.6 Identification of the Fission Product Release Source

7.6.1 Calculate the following ratios for each noble gas isotope and each iodine isotope using the decay corrected specific activities recorded on Enclosure 6. Enclosure 7 is provided as a worksheet.

$$\text{Noble Gas Ratio} = \frac{\text{Noble Gas Isotope Specific Activity}}{\text{Xe-133 Specific Activity}}$$

$$\text{Iodine Ratio} = \frac{\text{Iodine Isotope Specific Activity}}{\text{I-131 Specific Activity}}$$

7.6.2 Determine the source of release by comparing the results obtained to the predicted ratios provided in Enclosure 7 Identify as the source that ratio which is closest to the value obtained in step 7.6.1.

7.7 Quantitative Release Assessment

Calculate the total quantity of fission products available for release to the environment. Enclosure 8 is provided as a worksheet.

7.7.1 The quantity of fission products in the reactor coolant is determined as follows:

7.7.1.1 If the water level in the reactor vessel recorded on Enclosure 2 indicates that the vessel is full, the quantity of fission products in the reactor coolant is calculated by the following equation.

$$\text{Total Activity (Ci)} = A_o(\mu ci/cc) \times \text{RCS Volume}$$

where:

A_o = the specific activity of the reactor coolant sample recorded on Enclosure 6, $\mu ci/cc$.

RCS Volume = the full reactor coolant system water volume corrected to standard temperature by multiplying by the factor for water density provided in Enclosure 4.

7.7.1.2 If the water levels in the reactor vessel and pressurizer recorded in step 7.1.1 indicates that a steam void is present in the reactor vessel, then the quantity of fission products found in the reactor coolant is again calculated by step 7.7.1.1. However, it must be recognized that the value obtained will overestimate the actual quantity released. Therefore, this sample should be repeated at such time when the plant operators have removed the void from the reactor vessel.

7.7.1.3 If the water level in the reactor vessel recorded in step 7.1.1 is below the low end capability of the indicator, it is not possible to determine the quantity of fission products from this sample because the volume of water in the reactor coolant system is unknown. Under this condition, assessment of core damage if obtained using the containment sump sample.

7.7.2 The quantity of fission products in the containment building sump is determined as follows:

7.7.2.1 The water volume in the containment building sump is determined from the sump level recorded on Enclosure 2 and the curve provided in Enclosure 9.

7.7.2.2 The quantity of fission products in the sump is calculated by the following equation.

Total activity, Ci = $A_o(\mu ci/cc)$ x Sump Water Volume

where:

A_o = the specific activity of the containment sump sample recorded on Enclosure 6, $\mu ci/cc$.

7.7.3 The quantity of fission products found in the containment building atmosphere is determined as follows:

7.7.3.1 The volume of gas in the containment building is corrected to standard temperature and pressure using the following equation.

$$\text{Gas Volume (STP)} = \text{Gas Volume} \times \frac{(P_2 + P_1)}{P_2} \quad \frac{(T_2 + 460)}{(T_1 + 460)}$$

where:

T_1, P_1 = Containment atmosphere temperature and pressure recorded on Enclosure 2

T_2, P_2 = Standard temperature, 32^oF and Standard Pressure, 14.7 psia.

7.7.3.2 The quantity of fission products in the containment is calculated by the following equation.

$$\text{Total Activity, Ci} = A_0(^{\mu ci}/cc) \times \text{Gas Volume (STP)}$$

where:

A_0 = The specific activity of the containment building sample recorded on Enclosure 6, $^{\mu ci}/cc$.

7.7.4 The total quantity of fission products available for release to the environment is equal to the sum of the values obtained from each sample location. Record the total quantity of fission products on Enclosure 8.

7.8 Plant Power Correction

The quantitative release of the fission products is expressed as the percent of the source inventory at the time of the accident. The equilibrium source inventories are to be corrected for plant power history.

7.8.1 To correct the source inventory for the case in which plant power level has remained relatively constant prior to reactor shutdown the following procedure is employed. Enclosure 10 is provided as a worksheet.

7.8.1.1 The fission products are divided into two groups based upon the radioactive half lives. Group 1 isotopes are to be employed in the case where core power had not changed greater than ± 10 percent within the last 30 days prior to the reactor shutdown. Group 2 isotopes are to be employed in the case where core power had not changed greater than ± 10 percent within the last 4 days prior to the reactor shutdown.

7.8.1.2 The following equation is applied to the appropriate fission product group.

$$\text{Group 1 Power Correction Factor} = \frac{\text{Average Power for Prior 30 Days, \%}}{100}$$

$$\text{Group 2 Power Correction Factor} = \frac{\text{Average Power for Prior 4 Days, \%}}{100}$$

7.8.2 To correct the source inventory for the case in which plant power level has not remained constant prior to reactor shutdown, the following equation is employed. The entire 30 days power history should be employed. Enclosure 11 is provided as a worksheet.

$$\text{Power Correction Factor} = \frac{\sum_j P_j (1-e^{-\lambda t_j}) e^{-\lambda t_j^{o}}}{100}$$

where:

P_j = steady reactor power in period j, %

t_j = duration of period j, sec

t_j^o = time from end of period j to reactor shutdown, sec

λ = decay constant for isotope, sec^{-1}

7.9 Comparison of Measured Data with Source Inventory

The total quantity of fission products available for release to the environment recorded in Enclosure 8 is compared to the source inventory corrected for plant power history recorded in Enclosure 10 or 11. This comparison is made by dividing the two values for each isotope and calculating the percent of the corrected source inventory that is now in the sampled fluid and therefore available for release to the environment. Enclosure 12 is provided as a worksheet.

7.10 Assessment of Core Damage

The conclusion on core damage is made using the three parameters developed above. These are:

1. Identification of the fission product isotopes which most characterize a given sample (Enclosure 3).

2. Identification of the source of the release (Enclosure 7).

3. Quantity of the fission produce available for release to the environment expressed as a percent of source inventory (Enclosure 12).

Knowledgeable judgment is used to relate the above three parameters to the definitions of the 10 categories of fuel damage found in Enclosure 1. Core damage is not anticipated to take place uniformly. Therefore, when evaluating the three parameters listed above the procedure is anticipated to yield a combination of one or more of the 10 categories defined in Enclosure 1. These categories will exist simultaneously.

8.0 ALTERNATE PROCEDURE

The alternate core damage assessment procedure is based on calculated values of containment radiation levels in the period following an unmitigated large break LOCA. These calculations considered the release of fission products from gap and fuel, their escape from the RCS, and their subsequent dispersal within the containment building. The phenomena treated in such an analysis are difficult to model and

a number of simplifying assumptions must be made. The results, therefore, are subject to large uncertainties. Thus, the alternate procedure produces only a rough estimate of the degree of core damage.

8.1 Limitations and Precautions

8.1.1 The alternate procedure is based on calculations of containment radiation levels after a double-ended cold leg break, with no post-accident cooling of the core. Use of the procedure following an accident of lesser severity will tend to underestimate the extent of core damage.

8.1.2 The calculations on which the alternate procedure is based assumed full-power, equilibrium values of core fission product inventories. Use of the procedure in a situation in which the reactor has been operating for a short time and/or at lower power will tend to underestimate the extent of core damage.

8.2 Application of Procedures

8.2.1 Record readings of dose rate (mr/hr) from both of the containment high-range radiation monitors. Take the average of these two values.

8.2.2 Estimate the elapsed time (hours) between initial release of fission products from the core and the time when dose rate readings are taken. (If a sudden, pronounced increase in dose rate is observed, the new, higher dose rate should be recorded immediately. In this case, the elapsed time is considered to be zero.)

8.2.3 On Enclosure 13, record the point which corresponds to the averaged value of dose rate from 8.2.1 and the elapsed time from 8.2.2.

8.2.4 Interpolate between the curves of Enclosure 13 to estimate the percentage of clad ruptures or of melted fuel represented by the point recorded in 8.2.3.

8.2.5 Repeat steps 8.2.1 through 8.2.4 subsequently as frequently as practible.

ENCLOSURE 1

Radiological Characteristics of NRC Categories of fuel Damage

NRC Catetory of Fuel Damage	Mechanism of Release	Source of Release	Characteristics Isotope	Release of Characteristic Isotope Expressed as a Percent of Source Inventory
1. No Fuel Damage	Halogen Spiking Tramp Uranium	Gas Gap	I 131, Cs 137 Rb 88	Less than 1
2. Initial Cladding Failure		Gas Gap		Less than 10
3. Intermediate Cladding Failure	Clad Burst and Gas Gap Diffusion Release	Gas Gap	Xe 131m, Xe 133 I 131, I 133	10 to 50
4. Major Cladding Failure		Gas Gap		Greater than 50
5. Initial Fuel Pellet Overheating		Fuel Pellet	Cs 134, Rb 88, Te 129, Te 132	Less than 10
6. Immediate Fuel Pellet Overheating	Grain Boundary Diffusion	Fuel Pellet		10 to 50
7. Major Fuel Pellet Overheating	Diffusional Release From UO_2 Grains	Fuel Pellet		Greater than 50
8. Fuel Pellet Melt		Fuel Pellet		Less than 10
9. Intermediate Fuel Pellet Melt	Escape from Molten Fuel	Fuel Pellet	Ba 140, La 140 La 142, Pr 144	10 to 50
10. Major Fuel Pellet Melt		Fuel Pellet		Greater than 50

ENCLOSURE 2

Record of Plant Indications

Date ______________ Time ______________

Reactor Coolant System

Pressure ______________ psia

Temperature ______________ ^oF

Vessel Level ______________ %

Pressurizer Level ______________ %

RCS Volume 3.14×10^8 cc

Containment Building

Pressure ______________ psia

Temperature ______________ ^oF

Sump Level ______________ %

Containment Volume 6.6×10^{10} cc

Prior 30 Days Power History

Power __________ % for __________ days

Power __________ % for __________ days

Power __________ % for __________ days

Power __________ % for __________ days

Reactor Shutdown Time ______________ Date ______________

ENCLOSURE 3

Record of Sample Specific Activity

Sample Number:

Location:

Time of Analysis:

Temperature, °F:

Pressure, PSIG:

Sample Activity, μCi/cc:

Kr 87

Xe 131m

Xe 133

I 131

I 132

I 133

I 135

Cs 134

Rb 88

Te 129

Te 132

Sr 89

Ba 140

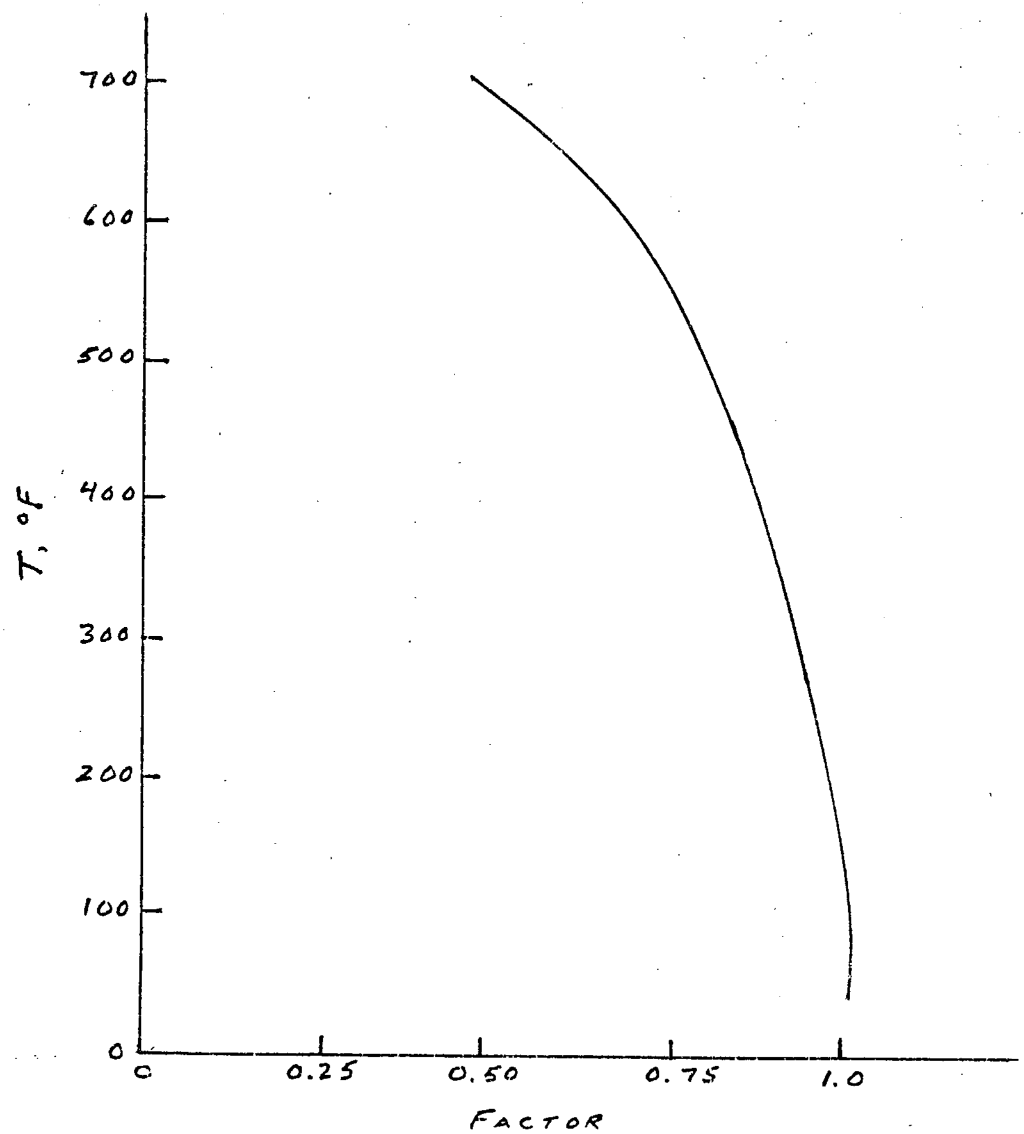
La 140

La 142

Pr 144

ENCLOSURE 4

Density Correction Factor for Reactor Coolant Temperature



ENCLOSURE 5

Record of Sample Temperature and Pressure Correcton

Sample Number:

Location:

Time of Analysis:

Temperature, ^{o}F:

Pressure, PSIG:

Isotope	Measured Specific Activity (Enclosure 3), μci/cc	Correction Factor	Specific Activity at STP, μci/cc
Kr 87			
Xe 131m			
Xe 133			
I 131			
I 132			
I 133			
I 135			
Cs 134			
Rb 88			
Te 129			
Te 132			
Sr 89			
Ba 140			
La 140			
La 142			
Pr 144			

ENCLOSURE 6

Record of Sample Decay Correction

Time of Reactor Shutdown:

Sample Number:

Location:

Time of Analysis:

Isotope	Decay Constant, 1/sec	Secific Activity at STP (Enclosure 5), μci/cc	Decay Corrected Specific Activity, μci/cc
Kr 87	1.5 (-4)		
Xe 131m	6.7 (-7)		
Xe 133	1.5 (-6)		
I 131	9.9 (-7)		
I 132	8.4 (-5)		
I 133	9.3 (-6)		
I 135	2.9 (-5)		
Cs 134	1.1 (-8)		
Rb 88	6.5 (-4)		
Te 129	1.7 (-4)		
Te 132	2.5 (-6)		
Sr 89	1.6 (-7)		
Ba 140	6.3 (-7)		
La 140	4.8 (-6)		
La 142	1.2 (-4)		
Pr 144	6.7 (-4)		

ENCLOSURE 7

Record of Fission Product Release Source Identification

Sample Number:
Location:

Isotope	Decay Corrected Specific Activity (Enclosure 6), μci/cc	Calculated Isotope Ratio*	Activity Radio in Fuel Pellet	Activity Radio In Gas Gap	Identified Source
Kr 87			0.2	0.001	
Xe 131m			0.003	0.003	
Xe 133		1.0	1.0	1.0	
I 131		1.0	1.0	1.0	
I 132			1.4	0.01	
I 133			2.0	0.5	
I 135			1.8	0.17	

$$\text{*Noble Gas Radio} = \frac{\text{Decay Corrected Noble Gas Specific Activity}}{\text{Decay Corrected Xe 133 Specific Activity}}$$

or

$$\text{*Iodine Radio} = \frac{\text{Decay Corrected Iodine Isotope Specific Activity}}{\text{Decay Corrected I 131 Specific Activity}}$$

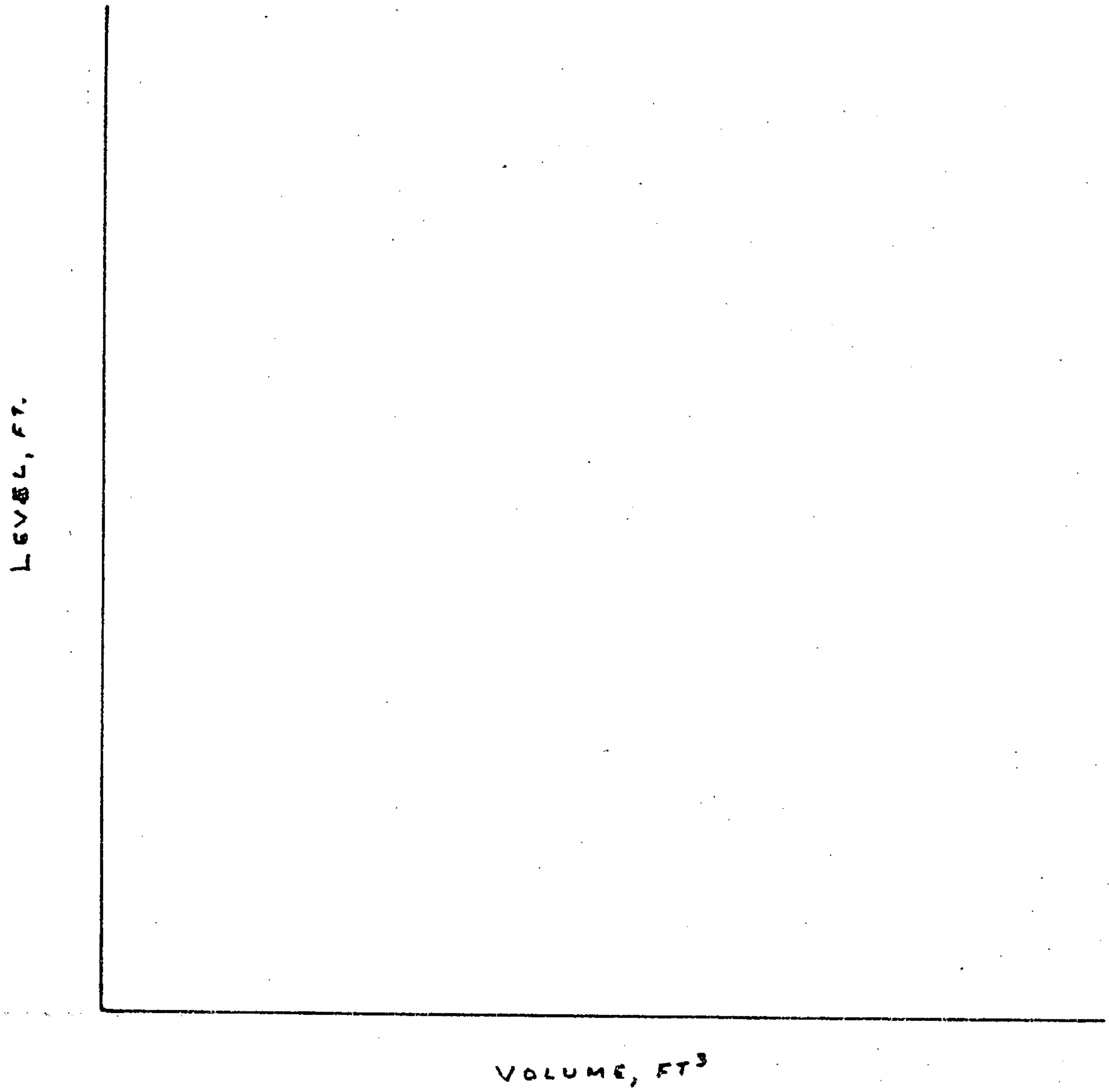
ENCLOSURE 8

Record of Release Quantity

Isotope	Reactor Coolant Sample Number, Ci	Containment Sump Sample Number, Ci	Containment Atmosphere Sample Number , Ci	Total Quantity Ci
Kr 87				
Xe 131m				
Xe 133				
I 131				
I 132				
I 133				
I 135				
Cs 134				
Rb 88				
Te 129				
Te 132				
Sr 89				
Ba 140				
La 140				
La 142				
Pr 144				

ENCLOSURE 9

Containment Building Sump Level



ENCLOSURE 10

Record of Correction for Constant Power Level

Sample Number:
Location:
Average Power for Prior 30 Days:
Average Power for Prior 4 Days:

Isotope	Fuel History Group	Power Correction Factor	Equilibrium Source Inventory, Ci	Corrected Source Inventory, Ci
Gas Gap Inventory				
Kr 87	2		9.5 (0)	
Xe 131m	1		6.6 (4)	
Xe 133	1		1.8 (7)	
I 131	1		9.0 (6)	
I 132	2		9.9 (3)	
I 133	2		8.9 (6)	
I 135	2		1.6 (6)	
Fuel Pellet Inventory				
Kr 87	2		4.7 (7)	
Xe 131m	1		7.0 (5)	
Xe 133	1		2.0 (8)	
I 131	1		9.9 (7)	
I 132	2		1.4 (8)	
I 133	2		2.0 (8)	
I 135	2		1.9 (8)	
Cs 134	1		1.8 (7)	
Rb 88	2		6.8 (7)	
Te 129	2		3.1 (7)	
Te 132	1		1.4 (8)	
Sr 89	1		9.4 (7)	
Ba 140	1		1.7 (8)	
La 140	1		1.8 (8)	
La 142	2		2.2 (8)	
Pr 144	2		1.2 (8)	

ENCLOSURE 11

Record of Correction for Non-Constant Power Level

(See Enclosure 2 for Power History)

Sample Number:
Location:

Isotope	Power Correction Factor	Equilibrium Source Inventory, Ci	Corrected Source Inventory, Ci
Gas Gap Inventory			
Kr 87		9.5 (0)	
Xe 131		6.6 (4)	
Xe 133		1.8 (7)	
I 131		9.0 (6)	
I 132		9.9 (3)	
I 133		8.9 (6)	
I 135		1.6 (6)	
Fuel Pellet Inventory			
Kr 87		4.7 (7)	
Xe 131m		7.0 (5)	
Xe 133		2.0 (8)	
I 131		9.9 (7)	
I 132		1.4 (8)	
I 133		2.0 (8)	
I 135		1.9 (8)	
Cs 134		1.8 (7)	
Rb 88		6.8 (7)	
Te 129		3.1 (7)	
Te 132		1.4 (8)	
Sr 89		9.4 (7)	
Ba 140		1.7 (8)	
La 140		1.8 (8)	
La 142		2.2 (8)	
Pr 144		1.2 (8)	

ENCLOSURE 12

Record of Percent Release

Isotope	Total Quantity Available for Release (Enclosure 8), Ci	Power Corrected Source Inventory, Ci (Enclosure 10 or 11)	Percent
Gas Gap Inventory			
Kr 87			
Xe 131			
Xe 133			
I 131			
I 132			
I 133			
I 135			
Fuel Pellet Inventory			
Kr 87			
Xe 131m			
Xe 133			
I 131			
I 132			
I 133			
I 135			
Cs 134			
Rb 88			
Te 129			
Te 132			
Sr 89			
Ba 140			
La 140			
La 142			
Pr 144			

ENCLOSURE 13

Containment Monitor Dose Rate vs. Time AfterInitial Release for Various Percentages of Clad Failure or Core Melt