Public Service Company of Colorado

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50-267

September 27, 1982
Fort St. Vrain
Unit #1
P-82423

Mr. Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: NUREG-0737 II.B.3
Post-Accident Sampling System

REFERENCE: G-82194

Dear Mr. Clark:

Enclosed is our submittal documenting how we have satisfied each criterion of NUREG-0737 item II.B.3, Post-Accident Sampling System. In order to reduce confusion, an attempt has been made to incorporate all past submittals into this document, thereby eliminating the need to reference other correspondence. As stated in our letter dated July 28, 1982 (P-82283) we feel that the only remaining open item under II.B.3 is ORNL's investigation of source terms.

Please note that correspondence (G-80049, received March 31, 1980) from Mr. Themis P. Speis, Chief, Advanced Reactors Branch, Division of Project Management to Mr. J. K. Fuller, Vice President, Public Service Co. of Colorado, states that ". . .the licensee has met the "Category A" requirements. . ." for Lessons Learned item 2.1.8.a, Post Accident Sampling.

A046

If questions arise in regard to this matter, please contact Mr. Ted Borst, Radiation Protection Manager, Fort St. Vrain Nuclear Generating Station, (303) 571-7436.

Very truly yours,

Don W. Warembourg

Don W. Warembourg
Manager, Nuclear Production
Fort St. Vrain Nuclear
Generating Station

DWW/skr

Enclosure

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.

Clarification: Provide information on sampling(s) and analytical laboratories locations including a discussion of relative elevations, distances and methods for sample transport. Responses to this item should also include a discussion of sample recirculation, sample handling and analytical times to demonstrate that the three-hour time limit will be met (see (6) below relative to radiation exposure). Also describe provisions for sampling during loss of off-site power (i.e. designate an alternate backup power source, not necessarily the vital (Class IE) bus, that can be energized in sufficient time to meet the three-hour sampling and analysis time limit).

PSC RESPONSE

Fort St. Vrain has the capability to promptly obtain and analyze reactor coolant samples. Primary coolant (helium) samples are obtained from the Analytical Instrumentation Room located on Level 7 (4829 elevation) of the Reactor Building. Health Physics Procedure HPP-14, "Analytical Instrumentation Room", contains procedures for obtaining primary coolant grab samples under normal and accident conditions. (Please note that the sample collection system is identical for normal and accident conditions.) Per HPP-14, sampling under accident conditions is preceded by a one minute purge and then a 2 ml sample is collected. The sample is transported to the Radiochemistry Laboratory, which is located in the on-site Technical Support Building on Level 5 (grade-4791' elevation). Walking time between the Analytical Instrumentation Room and the Radiochemistry Laboratory is approximately two (2) minutes. A "rabbit" system is available which is capable of delivering the sample to the Radiochemistry Laboratory within one (1) minute after sample isolation. In the event that projected exposures exceed 10CFR 20 limits, a portable lead transport container is available.

Primary coolant grab samples are analyzed on a GeLi detector system in accordance with Radiochemistry Procedure RCP-38, "Operation and Calibration of Computer-Based Gamma Analysis System". The maximum count time for primary coolant grab sample as specified in RCP-38 is 1000 seconds, (approximately 17 minutes).

From the above discussion it can be seen that Fort St. Vrain can easily satisfy the three-hour time limit specified in this criterion. It should also be noted that we have a continuous primary coolant activity monitor, RT-9301, which constantly monitors primary coolant noble gas activity, which is located in the Control Room and alarms in the Control Room, allowing for prompt operator action.

In the event of a loss of off-site power, primary coolant grab samples would be taken to Colorado State University in Fort Collins, located approximately 40 miles from the plant for analyses. Normal driving time for this trip is approximately one hour.

With respect to containment atmosphere samples, Fort St. Vrain as a High Temperature Gas Cooled Reactor does not utilize a containment as do Light Water Cooled Reactors. The Fort St. Vrain primary coolant system is completely contained within a 9 to 14 foot thick Prestressed Concrete Reactor Vessel (PCRV). This includes the helium circulators and steam generators. Thus the requirement to obtain containment atmosphere samples is not per se applicable. Fort St. Vrain does utilize a Reactor Building within which the PCRV is contained. The Reactor Building is accessible during power operations and is kept slightly subatmospheric. Health Physics Procedure HPP-12, "Portable Air Sample Collection and Analysis", contains guidance on obtaining air samples anywhere on site. This procedure is utilized to obtain air samples from the Reactor Building. Analysis of the samples is accomplished in the on-site Radiochemistry Laboratory. Relative elevations, distances, and sample transport and analysis methodology are similar to the case described for obtaining primary coolant grab samples. In the event of a loss of offsite power, we have two (2) gas driven generators which can be used for sample collection.

We feel that we have satisfied this criterion.

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 nonvolatile 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.

Clarification: 2 (a) A discussion of the counting equipment capabilities is needed, including provisions to handle samples and reduce background radiation (ALARA). Also a procedure is required for relating radionuclide concentrations to core damage. The procedure should include:

1. Monitoring for short and long lived volatile and non volatile radionuclides such as ^{133}Xe, ^{131}I, ^{137}Cs, ^{134}Cs, ^{85}Kr, ^{140}Ba, and ^{188}Kr (See Vol. II, Part 2, pp. 524-527 of Rogovin Report for further information).

2. Provisions to estimate the extent of core damage based on radionuclide concentrations and taking into consideration other physical parameters such as core temperature data and sample location.

2 (b) Show a capability to obtain a grab sample, transport and analyze for hydrogen.

2 (c) Discuss the capabilities to sample and analyze for the accident sample species listed here and in Regulatory Guide 1.97 Rev. 2.

2 (d) Provide a discussion of the reliability and maintenance information to demonstrate that the selected on-line instrument is appropriate for this application. (See (8) and (10) below relative to back-up grab sample capability and instrument range and accuracy).

PSC RESPONSE

(a) As discussed under Criterion 1, Fort St. Vrain has the onsite capability to quantify all radionuclides present in our primary coolant (Helium), and our Reactor Building atmosphere, within three hours. This is accomplished via our Radiochemistry Computerized Analysis System, comprised of NaI (TL) and GeLi detectors and associated multichannel analyzers. Our isotopic identification and quantification is handled via pre-programmed functions on the computer. In addition we have inline monitoring capabilities (RT-9301) for measuring noble gas activity in our primary coolant. This monitor reads out in the Control Room.

(b) The chemical impurities CO, CO_2, H_2, and CH_4 are indicators of the condition of the reactor core. Modest increases in the gaseous species H_2, CO, CO_2, CH_4 and N_2 in the primary coolant gas would indicate that core heatup has occurred causing outgassing of graphite components. Heatup alone would not damage the graphite but would cause fuel particle failure. Large increases in H_2, H_2O and CO could indicate oxidation of core components due to water ingress. Large increases in CO, N_2 and O_2 (but not H_2) could be indicative of air ingress and graphite oxidation. It is to be noted that small amounts of graphite oxidation would produce relatively large levels of gaseous product CO. For example, a CO pressure of 1 atm in the primary circuit would be indicative of only about 300 lbs. of graphite oxidized or about 0.1% of the fuel element graphite.

Continuous on-line monitoring equipment exists at the FSV facility for measuring the behavior of CO and moisture. All other impurities are measured with a gas chromatograph located in the Analytical Instrumentation Room. Chemical impurity sampling requires approximately one hour to complete.

(c) Boron and chloride analyses during the accident are not pertinent to the HTGR. Negative reactivity is assured by the control rods and by insertion of the reserve shut down balls which are solid boronated graphite materials. Chloride analysis is not required since the steam generators are essentially removed from service.

We feel that we have satisfied this criterion.

Criterion: (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.

Clarification: System schematics and discussions should clearly demonstrate that post accident sampling, including recirculation, from each sample source is possible without use of an isolated auxiliary system. It should be verified that valves which are not accessible after an accident are environmentally qualified for the conditions in which they must operate.

PSC RESPONSE

Primary Coolant and Reactor Building atmosphere sampling during post accident conditions do not require an isolated auxiliary system to be placed in operation in order to use the sampling system. Sample recirculation and collection during normal and post-accident conditions utilize the same systems. All valves necessary for post accident sampling are accessible after an accident.

We feel that we have satisfied this criterion.

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 O_2 concentration is recommended, but is not mandatory.

Clarification: Discuss the method whereby total dissolved gas or hydrogen and oxygen can be measured and related to reactor coolant system concentrations. Additionally, if chlorides exceed 0.15 ppm, verification that dissolved oxygen is less than 0.1 ppm is necessary, Verification that dissolved oxygen is <0.1 ppm by measurement of a dissolved hydrogen residual of ≥ 10 cc/kg is acceptable for up to 30 days after the accident. Within 30 days, consistent with ALARA, direct monitoring for dissolved oxygen is recommended.

PSC RESPONSE

The chemical impurities CO, CO_2, H_2, and CH_4 are indicators of the condition of the reactor core. Modest increases in the gaseous species H_2, CO, CO_2, CH_4 and N_2 in the primary coolant gas would indicate that core heatup has occurred causing outgassing of graphite components. Heatup alone would not damage the graphite but would cause fuel particle failure. Large increases in H_2, H_2O and CO could indicate oxidation of core components due to water ingress. Large increases in CO, N_2 and O_2 (but not H_2) could be indicative of air ingress and graphite oxidation. It is noted that small amounts of graphite oxidation would produce relatively large levels of gaseous product CO. For example, a CO pressure of 1 atm in the primary circuit would be indicative of only about 300 lbs. of graphite oxidized or about 0.1% of the fuel element graphite.

Continuous on-line monitoring equipment exists at the FSV facility for measuring the behavior of CO and moisture. All other impurities are measured with a gas chromatograph located in the Analytical Instrumentation Room. Access to the facility will be necessary during an accident to privide additional information concerning these impurities. Chemical impurity sampling currently requires approximately 1 hour to complete.

We feel that we have satisfied this criterion.

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.

Clarification: BWR's on sea or brackish water sites, and plants which use sea or brackish water in essential heat exchangers (e.g. shutdown cooling) that have only single barrier protection between the reactor coolant are required to analyze chloride within 24 hours. All other plants have 96 hours to perform chloride analysis. Samples diluted by up to a factor of one thousand are acceptable as initial scoping analysis for chloride, provided (1) the results are reported as ____ppm Cl (the licensee should establish this value; the number in the blank should be no greater than 10.0 ppm Cl) in the reactor coolant system and (2) that dissolved oxygen can be verified at <0.1 ppm, consistent with the guidelines above in clarification no. 4. Additionally, if chloride analysis is performed on a diluted sample, an undiluted sample need also be taken and retained for analysis within 30 days, consistent with ALARA.

PSC RESPONSE

Chloride analysis during the accident is not pertinent to the HTGR. Chloride analysis is not required since the steam generators are essentially removed from service.

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).

Clarification: Consistent with Regulatory Guide 1.3 or 1.4 source terms, provide information on the predicted man rem exposures based on person-motion for sampling, transport and analysis of all required parameters.

PSC RESPONSE

To obtain a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3, 1.4 and 1.7 requires a permanent loss of all forced circulation for the FSV HTGR. This specific accident was identified as DBA #1 in FSAR Section 14.10 and Appendix D. These analyses performed by General Atomic Company at the time of licensing did not consider Regulatory Guides 1.3 and 1.4 source terms (i.e., the equivalent of the 50% of the core radioiodine and 100% of the core noble gas inventory for release to the primary coolant) appropriate for the HTGR. However, because of past precedence by the then Atomic Energy Commission (AEC) of using the above source terms, offsite doses resulting from the postulated accident were calculated and presented in the previously mentioned FSAR sections using both the General Atomic Company release assumptions and AEC TID-14844 release assumptions. In both cases the offsite doses are within 10CFR100 limits.

DBA #1 Description:

A non-mechanistic loss of forced circulation is postulated from full power operation, where the reactor is scrammed by the plant protective system and all attempts to restore forced circulation using the multiple heat sinks, circulators and motive power for the circulators fail. Because of the large heat sink provided by the graphite core, considerable time is available to initiate primary coolant depressurization and to restore forced circulation. The FSV FSAR specifies the time available to initiate depressurization to be 5 hours, which was later amended by PSC letter P-77250 dated December 22, 1977 to be 2 hours. The reduction in time was due to the capability of the helium purification system to process primary coolant during the planned blowdown of the clean primary coolant to the reactor building ventilation stack. Thus, the depressurization of the PCRV is initiated after 2 hours and completed 7 hours later (or 9 hours from the onset of the accident), at which time the PCRV has been depressurized to 5 psig.

The fuel is slow to heat up due to the large heat sink provided by the core graphite. A peak average active core temperature of 5400 degrees F is reached about 80 hours after the onset of the accident. At this temperature, the core structural integrity and geometry are not compromised since the vaporization temperature of graphite is 6900 degrees F. Peak activity released to the primary coolant, considering decay, is reached about 24 hours into the accident.

Heat removal is provided by the liner cooling system in the redistribute mode which maximizes cooling in the top head of the PCRV.

Leakage of primary coolant from the PCRV is assumed to occur at a conservatively high leakage rate of 0.2% of the primary coolant inventory per day.

Offsite doses were calculated for a 6 month duration of the accident, but most of the offsite dose occurs in the first 200 hours of the accident, due to fission product decay.

The reactor building ventilation system maintains continous venting of the reactor building environment at 1.5 volumes/hr during the entire period of the accident.

Primary Coolant Leakage Rate During DBA #1:

The FSV FSAR DBA #1 (Appendix D, page D.1-56) assumed an arbitrarily conservative non-mechanistic estimate of PCRV leakage after the intentional depressurization by assuming that the liner has failed completely (or does not exist) and only concrete permeability controls the leakage. An internal 5 psi pressure differential was assumed which purportedly gave a PCRV leak rate of 8.33×10^{-5} fraction per per hr (0.2%/day). Reference was made to Question IX.7 of Amendment No. 2 and Question D.2 of Amendment No. 9 of the FSV PSAR for the calculation of the permeation rate for the FSV PCRV concrete under these conditions.

Examination of Question D.2 revealed simply the conclusion that a 5 psi positive differential pressure led to 0.2%/day and 2 psi positive differential pressure led to 0.08%/day. Question IX.7 also did not provide details of the calculation of the 0.2%/day rate. However, considerable detail and a derivation was provided for the analysis of leakage rate tests at high pressure. The following equation was provided (eqn.14 on page IX.7-8):

$$W\ (\text{lb/day}) - 1.13 \times 10^{-5}\ \frac{\Delta P}{\Delta P_o}\,\frac{A}{X}\,\ln\left(\frac{P_1}{P_2}\right)$$
$$+\ 2.2 \times 10^{-6}\ \frac{\Delta P}{\Delta P_o}\,\frac{A}{X}\left(P_1 2 - P_2 2\right)$$

(eqn. 14)

Where ΔP = PCRV inside pressure in psig

ΔP_o = PCRV inside pressure in psig for which the net compressive stress in concrete = 0

A = Face area of concrete, ft²

X = Concrete thickness, ft

P_1 = Permeation or high side pressure, psia

P_2 = Ambient or low side pressure, psia

Numerical values were inserted for P_1 = 845 psig with the assumption that ΔP_o was approximately equal to P_1 in the following equation (eqn.15 on same page):

$$W = 1.13 \times 10^{-5} \times \frac{9000}{10}\ \ln \frac{857.5}{12.5} + 9.1 \times 10^{-7}\ \frac{9000}{10}\left(857.5^2 - 12.5^2\right)$$

$$= 0.043 + 602 = 600\ \text{lb/day}$$

(eqn. 15)

The first item to note is that the coefficient for the second (laminar flow) term is in error which is most likely a single error in transcribing from equation 14 to 15 since equation 13 has the 9.1×10^{-7} coefficient. Equation 15 should read:

$$W = 1.13 \times 10^{-5} \times \frac{9000}{10}\ \ln \frac{857.5}{12.5} + 2.2 \times 10^{-6}\ \frac{9000}{10}\left(857.5^2 - 12.5^2\right)$$

$$= 0.043 + 1445 = 1450\ \text{lb/day}$$

(eqn. 15 revised)

The second item is that the $\Delta P/\Delta P_o$ term has been dropped in going from eqn. 14 to eqn. 15, which is significant if it is assumed that these equations are appropriate for evaluating the leak rate at P_1 = 5 psig.

Pressure P1 (psig)	lb/day			%/day			
	Eqn 14	15	15 Revised	14	15	15 revised	Given
							App D; Amend 9 Question D.2
5	.0019	.13	.30	.001	.07	.17	.20
							Amend 9 Question D.2
2	.003	.046	.107	.0001	.025	.59	.08

Since equation 14 is the appropriate equation, the 0.2%/day leak rate is conservative by a factor of 200. Furthermore, the only equation that comes close to the values given in the SAR is 15 Revised, that is, $\Delta P/\Delta P_o$ has been neglected which accounts for the factor of 200.

For purposes of plant shielding and equipment environmental evaluations, the historic 0.2%/day is assumed to exist as an upper limit of all potential contaminated primary coolant leakage including permeability through the PCRV concrete. This is judged to be conservative since the primary coolant with any significant activity is contained within the PCRV or helium purification components contained in wells within the PCRV.

Radionuclide Source Terms for DBA-1:

As previously stated, the fuel within the graphite core is slow to heatup during DBA#1. Once it has reached the FSAR fuel particle coating failure temperature of 1725°C (3137°F), the fission products are assumed, for purposes of this shielding evaluation, to be released per the TID-14844 assumptions. For release to the primary coolant within the PCRV, this is 100% of noble gases, 50% of the iodines and 1% others. The total activity in curies contained in the primary coolant, assuming no leakage from the PCRV, as a funciton of lapsed time, is given in Table 1.

Consistent with TID-14844 release assumptions, 50% of the iodines plateout within the primary coolant system resulting in a depletion of the iodine to 25% of core inventory in the reactor building air. Thus, the total activity in curies in the reactor building, assuming the upper limit of 0.2%/day leakage (which is being purged by the reactor building ventilation system at the rate 1.5 volumes/hr), is given in Table 2.

TABLE 1

SV-NUREG-0578 STUDY TOTAL ACTIVITY (Ci) PRESENT IN THE PCRV PRIMARY COOLANT AT GIVEN ELAPSED TIME (hours). PCRV PRESS BOUNDARY REMAINS INTACT. TID-14844 NORMALIZATION FRACTIONS USED, 100% NOBLE GASES, 50% IODINE 1% OTHERS

ELAPSED TIME (Hours)

NUCLIDE	2	8	24	34	40	48	52	58	72	120	240	475	720	4320
Kr-88	1.05+04	2.89+05	2.80+05	2.39+04	5.89+03	1.37+03	5.50+02	1.76+02	7.04+01	0	0	0	0	0
Rb-88	8.57+03	2.79+05	2.80+05	2.66+04	6.51+03	1.46+03	6.07+02	1.89+02	7.08+01	0	0	0	0	0
Zr-95	3.15+01	6.66+03	1.84+05	2.57+05	3.01+05	3.59+05	3.69+05	3.84+05	4.18+05	4.12+05	3.88+05	3.43+05	3.02+05	4.60+04
Nb-95	3.18+01	6.74+03	1.87+05	2.63+05	3.09+05	3.69+05	3.80+05	3.97+05	4.35+05	4.37+05	4.31+05	4.12+05	3.88+05	8.90+04
I-131	1.33+03	3.50+05	6.18+06	6.91+06	7.33+06	7.88+06	7.90+06	7.93+06	7.98+06	7.57+06	4.89+06	2.07+06	8.45+05	0
I-132	1.44+03	2.34+05	1.79+06	6.09+05	5.64+05	5.61+05	3.68+05	2.96+05	2.72+05	1.76+05	4.02+04	4.99+03	5.46+02	0
I-133	2.48+03	5.30+05	6.44+06	5.25+06	4.70+06	4.12+06	3.65+06	3.05+06	2.04+06	4.84+05	8.81+03	0	0	0
Xe-133	5.25+03	1.40+06	2.50+07	2.78+07	2.94+07	3.14+07	3.14+07	3.12+07	3.09+07	2.73+07	1.41+07	3.86+06	9.90+05	0
I-135	1.98+03	2.46+05	1.40+06	5.49+05	3.31+05	1.88+05	1.25+05	6.83+04	1.78+04	2.94+02	0	0	0	0
Xe-135M	7.28+02	8.34+04	4.59+05	1.72+05	1.04+05	5.97+04	3.91+04	2.14+04	5.58+03	0	0	0	0	0
Xe-135	1.75+03	5.43+05	6.24+06	3.86+06	2.93+06	2.31+06	1.62+06	1.08+06	4.39+05	1.81+04	0	0	0	0
Ba-140	5.44+01	1.44+04	2.58+05	2.92+05	3.13+05	3.39+05	3.42+05	3.45+05	3.54+05	3.57+05	2.70+05	1.56+05	8.80+04	0
La-140	3.34+01	7.37+03	2.01+05	2.60+05	2.93+05	3.36+05	3.43+05	3.54+05	3.75+05	3.96+05	3.10+05	1.80+05	1.01+05	0

TABLE 2

SV-NUREG-0578 STUDY TOTAL ACTIVITY (Ci) PRESENT IN THE REACTOR BUILDING ATMOSPHERE AT GIVEN ELAPSED TIME (hours). PCRV LEAK RATE TO BUILDING 0.2%/DAY. REACTOR BUILDING VENTED AT 1.5 VOLUMES/HR. TID-14844 NORMALIZED FRACTIONS USED; 100% NOBLE GASES, 25% IODINE, 1% OTHERS

NUCLIDE	ELAPSED TIME (Hours) 2	8	24	34	40	48	52	58	72	120	240	475	720	4320
Kr-88	3.77-01	1.31+01	1.33+01	1.32+00	3.22-01	7.10-02	3.00-02	9.23-03	3.38-03	0	0	0	0	0
Rb-88	3.55-01	1.37+01	1.42+01	1.48+00	3.58-01	7.77-02	3.34-02	1.02-0[illegible]	3.61-03	0	0	0	0	
Zr-95	1.20-03	3.29-01	9.81+00	1.40+01	1.64+01	1.97+01	2.04+01	2.12+01	2.31+01	2.29+01	2.16+01	1.91+01	1.68+01	2.56+00
Nb-95	1.21-03	3.33-01	9.98+00	1.43+01	1.69+01	2.02+01	2.10+01	2.19+01	2.41+01	2.43+01	2.39+01	2.29+01	2.16+01	4.94+00
I-131	2.52-02	8.64+00	1.65+02	1.90+02	2.02+02	2.17+02	2.19+02	2.20+02	2.21+02	2.10+02	1.36+02	5.76+01	2.35+01	0
I-132	2.57-02	5.24+00	4.24+01	1.58+01	1.46+01	1.46+01	1.05+01	8.46+00	7.75+00	5.14+00	1.30+00	1.61-01	1.77-02	0
I-133	4.68-02	1.30+01	1.70+02	1.44+02	1.29+02	1.13+02	1.01+02	8.45+01	5.65+01	1.34+01	2.45-01	0	0	0
Xe-133	1.99-01	6.94+01	1.34+03	1.54+03	1.63+03	1.75+03	1.75+03	1.75+03	1.75+03	1.52+03	7.85+02	2.14+02	5.50+01	0
I-135	3.68-02	5.89+00	3.60+01	1.51+01	9.05+00	5.09+00	3.47+00	1.89+00	4.89-01	0	0	0	0	0
Xe-135M	3.14-02	7.71+00	6.59+01	7.38+01	5.53+01	3.53+01	2.75+01	1.81+01	6.20+00	1.07-01	0	0	0	0
Xe-135	6.75-02	2.73+01	3.40+02	2.26+02	1.72+02	1.22+02	9.56+01	6.40+01	2.56+01	1.01+00	0	0	0	0
Ba-140	2.06-03	7.12-01	1.38+01	1.61+01	1.72+01	1.87+01	1.89+01	1.91+01	1.96+01	1.98+01	1.50+01	8.67+00	4.89+00	0
La-140	1.27-03	3.66-01	1.08+01	1.43+01	1.61+01	1.85+01	1.90+01	1.96+01	2.08+01	2.20+01	1.72+01	9.98+00	5.62+00	0

Radiation Levels During DBA-1:

Based upon TID-14844 source term release assumptions, the radiation levels were calculated in the reactor building and the control room to determine the operator accessibility. Details are described herein.

Assumptions

In addition to the assumptions used in deriving the source terms, the following assumptions were made for evaluating shielding adequacy:

1. Credit was taken for a 50% depletion of the iodines due to plateout in the primary coolant system prior to release to the reactor building atmosphere.

2. All fission products were assumed to remain gasborne. In other words, no plateout of fission products was contemplated.

3. All the activities were uniformly distributed throughout the free space of the reactor building or the PCRV.

4. The iodines and particulates removed by the reactor building ventilation filters were deposited in any two of the three filters available.

5. Only major shielding such as concrete walls was considered.

Reactor Building

To determine the accessibility of the reactor building during the course of DBA-1, the gamma dose rate in the reactor building was calculated as a function of elapsed time. The contributing sources consist of the gasborne activity in the reactor building as a result of the PCRV leakage, the primary coolant activity contained in the PCRV and the buildup of iodines and particulates in the reactor plant exhaust filters. However the contribution from the reactor plant exhaust filters was not included in this part of the analysis because the accumulation of iodines and particulates in the time frame of interest, the first 9 hours, is not significant.

Two dose points were selected for the dose rate calculation. The first point is located at the center of the space above the refueling floor (=40 ft from the floor), and the second point is on the refueling floor directly above the refueling penetration. The PATH code described in FSAR Section 11.2.2.4 was utilized to perform the calculation.

Figure 1 shows the dose rate at the first dose point. Essentially all the contributions come from the gasborne activity in the reactor building. The activity in the PCRV is relatively insignificant to the first dose point, because of a large separation distance between the source and dose point. Short-term access to the reactor building is possible.

The activities given in Table 1 for 2 hours after initiation of the accident correspond to a 2.5 mrem/hour gamma radiation field in the reactor building as shown in Figure 1. Based upon these findings, it will be possible to sample the primary coolant throughout the postulated accident without incurring excessive personnel radiation exposures. The highest radiation levels expected in the reactor building would occur at 24 hours into the accident and correspond to approximately 1.4 rem/hour radiation field.

The dose rate of the second dose point (i.e., the refueling floor) is given in Figure 2. The contributions from the reactor building and from the PCRV are individually represented, along with the total dose rate. The contribution from the PCRV is due to the primary coolant activity present in the interspace below the primary closure for the control rod drive. The maximum dose rate on the floor is 1.0 rem/hr, which is less than the peak dose rate of 1.4 rem/hr at the first dose point. Therefore, the refueling floor is accessible on a short-term basis.

Control Room

The dose rates in the control room include the contributions from the airborne activity in the reactor building atmosphere, and from the iodine and particulate activity accumulated in the plant exhaust filters. The PATH code was used to determine the contribution from each source as a function of time into accident. For each source, the dose rates were calculated at various points in order to locate the point of the maximum dose rate.

The results of the PATH calculations were presented in Figure 3 as a function of elapsed time. The contributions from the reactor building and from the exhaust filters are shown separately at different dose points. Both curves in Figure 3 represent the maximum dose rates with respect to locations. The location of the maximum dose rate from each source is indicated in Figure 3.

It appears that the dose rate due to the reactor building airborne activity is acceptable for occupational access. The filter activity gives a peak dose rate of 24 mrem/hr near the ceiling of the Reactor Engineer's Office. The dose rate drops by more than one order of magnitude at points within the personnel level (i.e., within 6-foot height off the control room floor). In other words, the peak dose rate from the filter activity should be less than 2.5 mrem/hr at the personnel level.

Summary Results

The peak dose rates in the reactor building and control room are summarized below. Also indicated are the time at which the peak dose rate occurs following an accident and the total dose accumulated over a period of 180 days from initiation of the accident. The 180-day dose is given for the control room only based on the occupancy factor provided in NRC Standard Review Plan 6.4.

Location and Condition	Peak Gamma Dose Rate	Time of Peak	180-Day Dose (rem)
Reactor Building (above refueling floor)	1.4 R/hr	1D	--
Control Room (at ceiling)			
From Exhaust Filters	24 mR/hr	~30D	25
From Reactor Building	2 mR/hr	1D	~0.3
Control Room (at personnel level)			
From Exhaust Filters	<2.5 mR/hr	~30D	<2.5
From Reactor Building	3 mR/hr	1D	0.4

Conclusions

The following conclusions are reached from the review of shielding design adequacy for DBA-1 conditions and TID-14844 source term release assumptions:

1. The radiation levels in the control room are acceptable for occupational occupancy during DBA-1. Both dose rate and integrated dose within the spaces occupied by the personnel meet the NRC regulatory criteria.

2. No additional shielding is required for the plant exhaust filters, as far as personnel access to the reactor building during the first 9 hours of the accident and continuous access to the control room is concerned.

3. Areas immediately outside the reactor building should be accessible only on a restricted basis because of direct radiation from the activity in the reactor building.

4. Primary coolant grab samples and Reactor Building atmosphere samples can be obtained and analyzed without exceeding GDC 19 radiation exposures. The time required to enter the Reactor Building, obtain either a primary coolant or Reactor Building atmosphere sample and exit is approximately 15 minutes.

It should be noted that Oak Ridge National Laboratory is continuing research on a Fort St. Vrain source term consistent with Regulatory Guides 1.3 and 1.4.

We feel that we have satisfied this criterion, pending additional source term information.

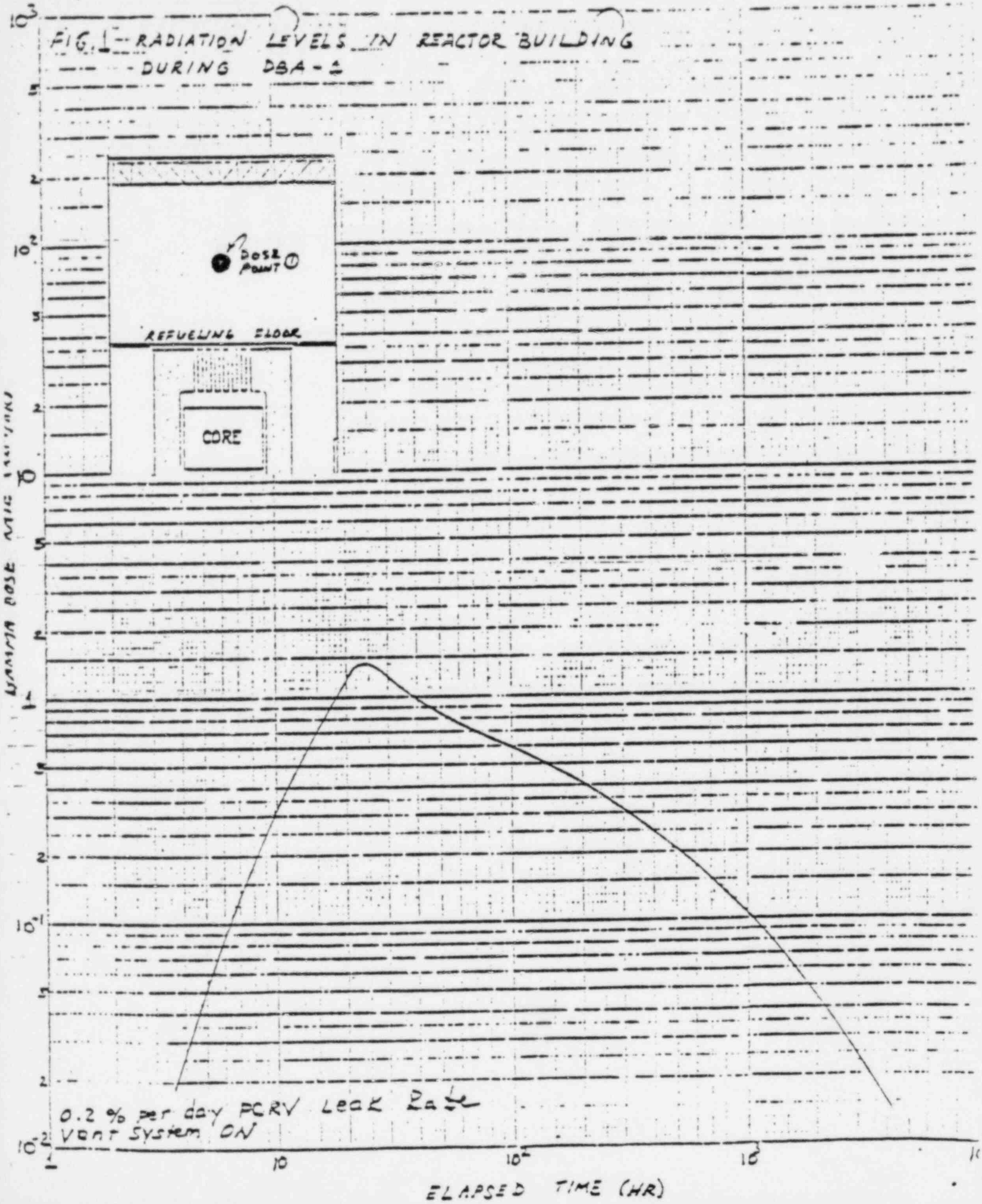
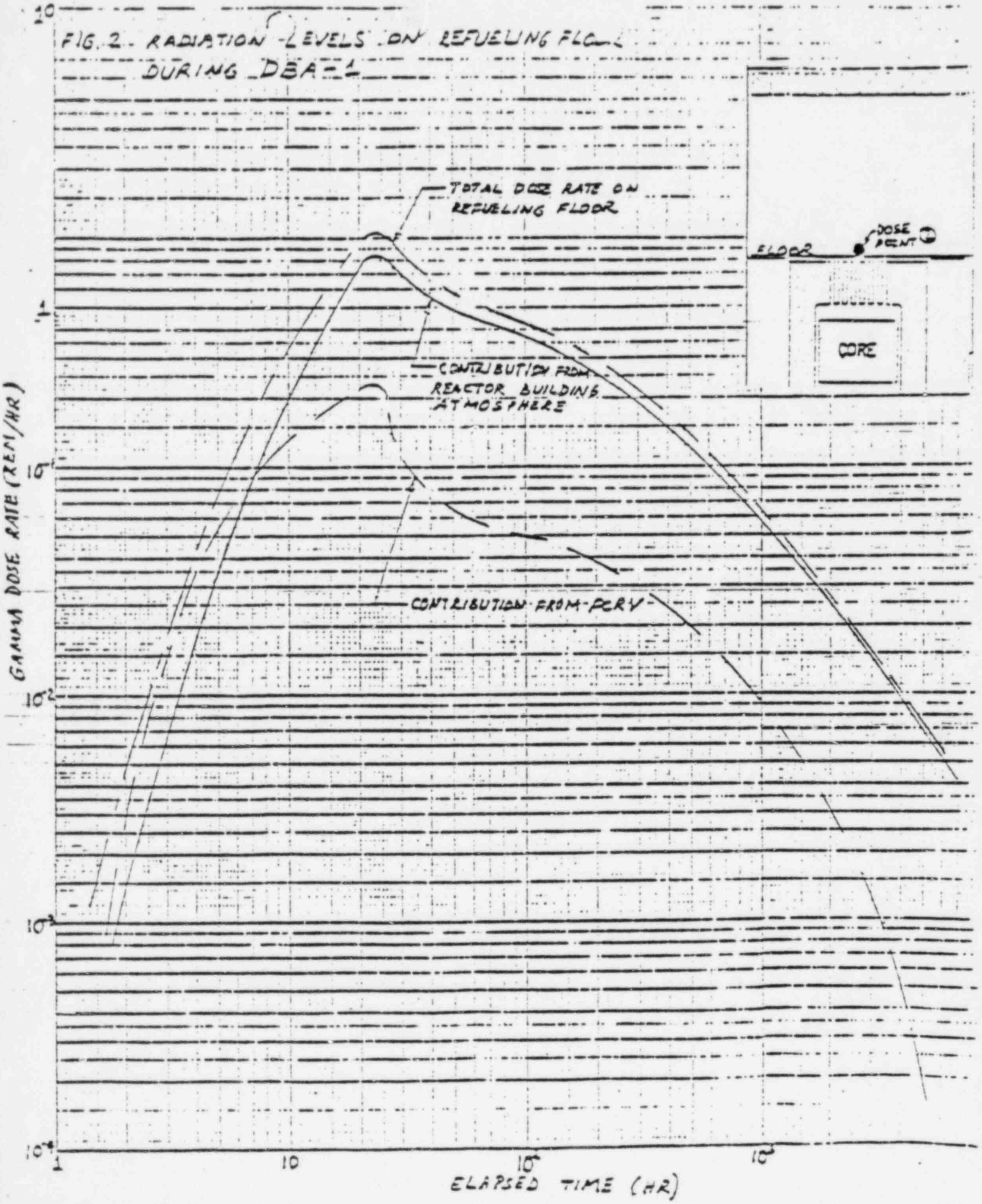


FIG. 1 -- RADIATION LEVELS IN REACTOR BUILDING
- DURING DBA - 1
DOSE POINT ①
REFUELING FLOOR
CORE
GAMMA DOSE RATE (R/HR)
0.2 % per day PCRV Leak Rate
Vent System ON
ELAPSED TIME (HR)

FIG. 2 RADIATION LEVELS ON REFUELING FLOOR DURING DBA-1



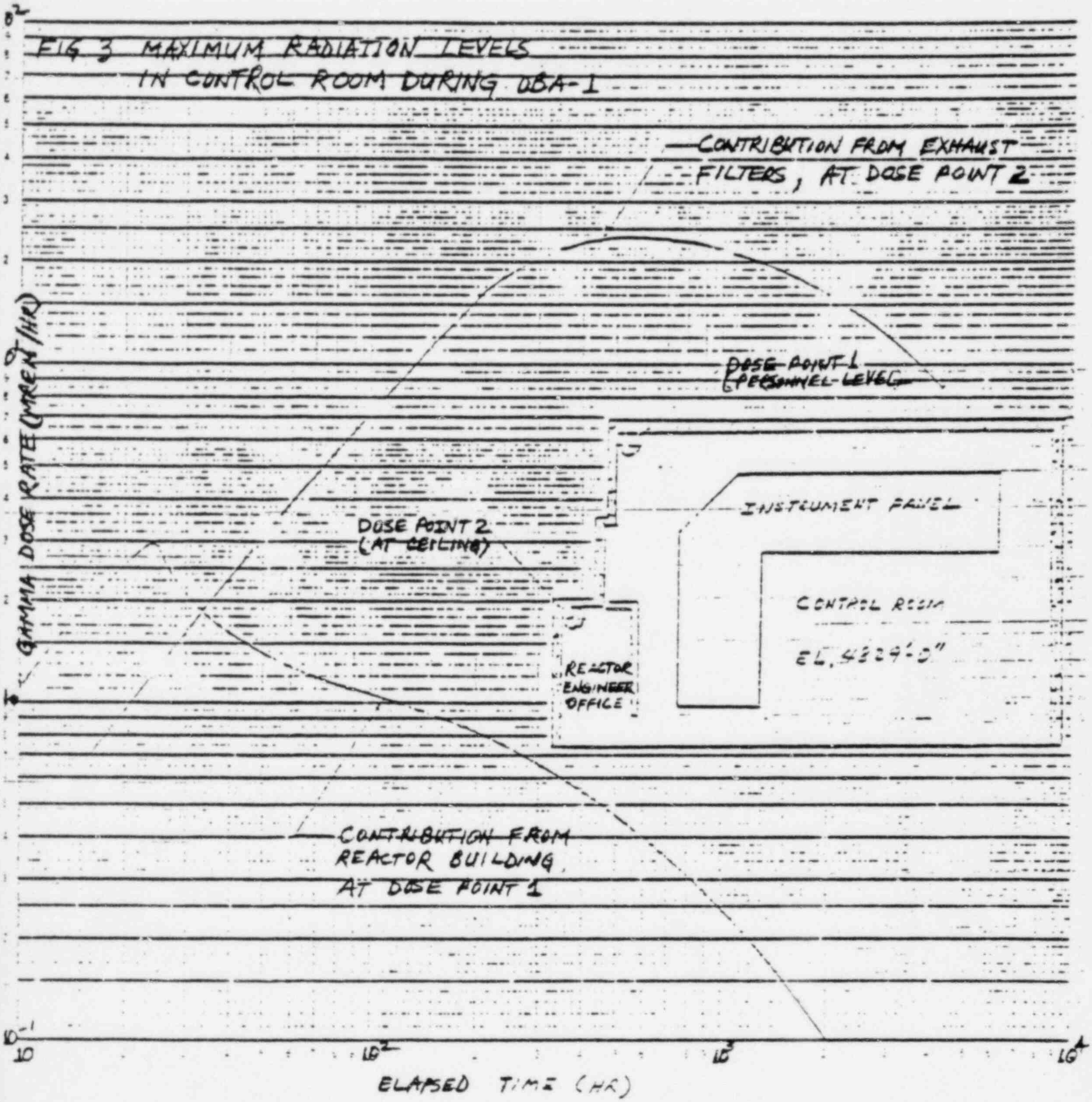


FIG 3 MAXIMUM RADIATION LEVELS IN CONTROL ROOM DURING DBA-1

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).

Clarification: PWR's need to perform boron analysis. The guidelines for BWR's are to have the capability to perform boron analysis but they do not have to do so unless boron was injected.

PSC RESPONSE

This criterion is not applicable to the Fort St. Vrain High Temperature Gas Cooled Reactor. Negative reactivity is assured by the control rods and by insertion of the reserve shutdown balls which are solid boronated graphite materials.

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 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.

Clarification: A capability to obtain both diluted and undiluted backup samples is required. Provisions to flush inline monitors to facilitate access for repair is desirable. If an off-site laboratory is to be relied on for the backup analysis, an explanation of the capability to ship and obtain analysis for one sample per week thereafter until accident condition no longer exists should be provided.

PSC RESPONSE

As previously noted for criteria 1, 2, and 4, backup sampling capability exists for all online monitoring used. Sample analysis is provided our onsite Radiochemistry Laboratory, which is located in the Technical Support Building. In the highly unlikely event that this laboratory would not be available, sample analysis would be provided via the facilities of Colorado State University in Fort Collins, which is approximately one hour from the plant. Planning with Colorado State University has been established via a letter of agreement in the Fort St. Vrain Radiological Emergency Response Plan, (attached). Additional sample analysis could be provided by the NRC mobile laboratory, which is located in Arlington, Texas.

We feel that we have satisfied this criterion.

IIB-5(h) COLORADO STATE UNIVERSITY

Department of Radiology and Radiation Biology
303/491-5222

Colorado State University
Fort Collins, Colorado
80523

September 27, 1979

TO: Public Service Company of Colorado
P. O. Box 840
Denver, CO 80200
Attn: R. Petzke

FROM: James E. Johnson, Professor James Johnson

SUBJECT: Environmental and personal radiation monitoring capabilities in the event of a non-normal radioactivity release from the FSV reactor.

Colorado State University has conducted the radiation environmental surveillance program at the FSV reactor since 1969. This period includes the entire preoperational phase as well as the current operational phase. This program has been on a contractual basis with the University.

All personnel and facilities that currently are committed to the environmental radiation surveillance program at the FSV reactor would be available for immediate response in the event of a release. The procedure would be to collect current samples from all stations and begin a new sampling period. In addition all backup equipment could be used for additional sampling stations if necessary.

The principle teaching program of the department is radiation protection technology and major research programs are in the areas of environmental radioactivity and personal radiation dosimetry. The personnel in the radiation protection areas include four certified health physicists as well as two radioecologists. These would be available for assistance to reactor, state or governmental groups. In addition the department has a full range of monitoring equipment which is continuously available. A whole-body counter is in operation at the department main building which is less than 30 miles from FSV. The department also has expertise in agricultural considerations of food chain transport of any environmental radioactivity.

The personnel and resources of the department would be available to assist and/or direct any radiation protection effort.

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 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 source, and by the use of a ventilation system design which will control the presence of airborne radioactivity.

Clarification: (9)(a) Provide a discussion of the predicted activity in the samples to be taken and the methods of handling/dilution that will be employed to reduce the activity sufficiently to perform the required analysis. Discuss the range of radionuclide concentration which can be analyzed for, including an assessment of, the amount of overlap between post accident and normal sampling capabilities.

(9)(b) State the predicted background radiation levels in the counting room, including the contribution from samples which are present. Also provide data demonstrating what the background radiation levels and radiation effect will be on a sample being counted to assure an accuracy within a factor of 2.

PSC RESPONSE

a) Based on the activity assumed to be present in the PCRV primary coolant following Fort St. Vrain Design Basis Accident 1 (Table 1 of response to Criterion 6), a study was initiated to determine the predicted activity in primary coolant grab samples. Health Physics Procedure HPP-14, "Analytical Instrumentation Room", contains guidance for primary coolant sampling during accident conditions. Based on Table 1 data and HPP-14, the following primary coolant sample activities for various post-accident times were calculated:

Post-Accident Time (hr)	Sample Activity μci	Sample Exposure Rate at the detector* (mR/hr)
2	1.5E+2	1.8E-1
8	1.6E+4	6.0E+0
24	2.0E+5	2.1E+1 (Maximum Post-Accident Value)
34	2.0E+5	1.3E+1
40	2.0E+5	1.1E+1
48	2.2E+5 (Maximum Post-Accident Value)	1.0E+1

* The distance from the sample to the GeLi detector in the Fort St. Vrain Radiochemistry Laboratory is 86.5 cm. From previous studies it has been determined that it is within the capabilities of the laboratory electronics to analyze samples located at 86.5 cm from the detector reading up to 1 mR/hr at the detector.

A lead sample transport container is available for the primary coolant grab samples if required. The radiochemistry sample analysis equipment is identical for both normal and post-accident sampling situations. Capability for performing analyses on post-accident samples is obtained by increasing the source to detector distance.

From the above information it can be seen that after 8 hours post-accident, the sample exposure rate exceeds the capability of the Radiochemistry Laboratory. However, by this time (see response to Criterion 6) the PCRV has been depressurized to 5 PSIG, and the only leakage pathway hypothesized is through the PCRV concrete, at 0.2% per day into the Reactor Building. Thus sampling of the PCRV ("Primary Circuit") is not necessary or desirable. The critical monitoring areas are now the Reactor Building and the Reactor Plant Exhaust Stack. From Table 2 of the response to Criterion 6, it can be seen that the maximum concentration of a noble gas sample (1 standard liter) taken in the reactor building would be approximately 0.5 μCi/scc. This corresponds to a sample activity of 500 μCi per liter and would result in an exposure rate of approximately 0.6 mR/hr at the detector in the Radiocheimistry Laboratory. This is within the capabilities of the laboratory to analyze.

Even taking no credit for the removal efficiency of the Reactor Plant Exhaust stack components, a concentration of 0.5 μCi/scc is within the capabilities of the existing on line stack radiation monitors.

b) The radiochemistry laboratory is located in the Technical Support Building and has the same shielding characteristics as the Technical Support Center (TSC). The attached study indicates the expected radiation levels in the TSC following Fort St. Vrain Design Basis Accident 1. With respect to page 7 of the attached study, please note that the walls and ceiling of the Technical Support Building contain 12 inches of concrete shielding, and that the TSC input (makeup) air system includes a HEPA filter and activated charcoal adsorber. It can be seen that the radiation levels in the TSC are of no concern with respect to their effect on radiochemical sample analyses. The detectors located in the Radiochemistry Laboratory are housed in lead shields, and lead shielding is available for samples which are brought into the Radiochemistry Laboratory.

We feel that we have satisfied this criterion.

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 system.

Cla' 'ification: The recommended ranges for the required accident sample analyses are given in Regulatory Guide 1.97, Rev. 2. The necessary accuracy within the recommended ranges are as follows:

- Gross activity, gamma spectrum: measured to estimate core damage, these analyses should be accurate within a factor of two across the entire range.

- Boron: measure to verify shutdown margin.

In general this analysis should be accurate within ±5% of the measured value (i.e. at 6,000 ppm B the tolerance is ± 300 ppm while at 1,000 ppm B the tolerance is ± 50 ppm). For concentrations below 1,000 ppm the tolerance band should remain at ± 50 ppm.

- Chloride: measured to determine coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm chloride the analysis should be accurate within ± 10% of the measured value. At concentrations below 0.5 ppm the tolerance band remains at ± 0.05 ppm.

- Hydrogen or Total Gas: monitored to estimate core degradation and corrosion potential of the coolant.

An accuracy of ± 10% is desirable between 50 and 2000 cc/kg but ± 20% can be acceptable. For concentration below 50 cc/kg the tolerance remains at ± 5.0 cc/kg.

- Oxygen: monitored to assess coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm oxygen the analysis should be accurate within ± 10% of the measured value. At concentrations below 0.5 ppm the tolerance band remains at ± 0.05 ppm.

- pH: measured to assess coolant corrosion potential.

Between a pH of 5 to 9, the reading should be accurate within ± 0.3 pH units. For all other ranges ± 0.5 pH units is acceptable.

To demonstrate that the selected procedures and instrumentation will achieve the above listed accuracies, it is necessary to provide information demonstrating their applicability in the post accident water chemistry and radiation environment. This can be accomplished by performing tests utilizing the standard test matrix provided below or by providing evidence that the selected procedure or instrument has been used successfully in a similar environment.

STANDARD TEST MATRIX
FOR
UNDILUTED REACTOR COOLANT SAMPLES IN A POST-ACCIDENT ENVIRONMENT

Constituient	Nominal Concentration (ppm)	Added as (chemical salt)
I^-	40	Potassium Iodide
Cs^+	250	Cesium Nitrate
Ba^{+2}	10	Barium Nitrate
La^{+3}	5	Lanthanum Chloride
Ce^{+4}	5	Ammonium Cerium Nitrate
Cl^-	10	
B	2000	Boric Acid
Li^+	2	Lithium Hydroxide
NO_3^-	150	
NH_4^+	5	
K^+	20	
Gamma Radiation (Induced Field)	10^4 Rad/gm of Reactor Coolant	Adsorbed Dose

NOTES:

1) Instrumentation and procedures which are applicable to diluted samples only, should be tested with an equally diluted chemical test matrix. The induced radiation environment should be adjusted commensurate with the weight of actual reactor coolant in the sample being tested.

2) For PWRs, procedures which may be affected by spray additive chemicals must be tested in both the standard test matrix plus appropriate spray additives. Both procedures (with and without spray additives) are required to be available.

3) For BWRs, if procedures are verified with boron in the test matrix, they do not have to be tested without boron.

4) In lieu of conducting tests utilizing the standard test matrix for instruments and procedures, provide evidence that the selected instrument or procedure has been used successfully in a similar environment.

All equipment and procedures which are used for post accident sampling and analyses should be calibrated or tested at a frequency which will ensure, to a high degree of reliability, that it will be available if required. Operators should receive initial and refresher training in post accident sampling, analysis and transport. A minimum frequency for the above efforts is considered to be every six months if indicated by testing. Thes provisions should be submitted in revised Technical Specifications in accordance with Enclosure 1 of NUREG-0737. The staff will provide model Technical Specifications at a later date.

PSC RESPONSE

a) Gross activity, gamma spectrum - these analyses are accurate within a factor of two across the range applicable to Fort St. Vrain. Note that the ranges contained in Regulatory Guide 1.97, Rev. 2 are for Light Water Reactors and are not applicable to the Fort St. Vrain High Temperature Gas Cooled Reactor.

b) Boron - not applicable to Fort St. Vrain, as negative reactivity is assured by the control rods and if necessary by insertion of the reserve shutdown balls which are solid boronated graphite material.

c) Chloride - not applcable to Fort St. Vrain as the steam generators are essentially removed from service.

d) Hydrogen, Oxygen or Total Gas - The chemical impurities CO, CO_2, H_2, and CH_4 are indicators of the condition of the reactor core. Modest increases in the gaseous species H_2, CO, CO_2, CH_4 and N_2 in the primary coolant gas would indicate that core heatup has occurred causing outgassing of graphite components. Heatup alone would not dam age the graphite but would cause fuel particle failure Large increases in H_2, H_2O and CO could indicate oxidation of core components due to water ingress. Large increases in CO, N_2 and O_2 (but not H_2) could be indicative of air ingress and graphite oxidation. It is to be noted that small amounts of graphite oxidation would produce relatively large levels of gaseous product CO. For example, a CO pressure of 1 atm in the primary circuit would be indicative of only about 3000 lbs. of graphite oxidized or about 0.1% of the fuel element graphite.

Continuous on-line monitoring equipment exist at the FSV facility for measuring the behavior of CO and moisture. All other impurities are measured with a gas chromatograph located in the analytical instrumentation room.

Note that the normal and post-accident sampling systems are identical and as such calibration and training are accomplished on a routine (on going) basis. We will address potential Technical Specification revisions when models are provided by the NRC.

We feel that we have satisfied this criterion.

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 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-effieiency particulate air (HEPA) filters.

Clarification: (11)(a) A description of the provisions which address each of the items in clarification 11.a should be provided. Such items, as heat tracing and purge velocities, should be addressed. To demonstrate that samples are representative of core conditions a discussion of mixing, both short and long term, is needed. If a given sample location can be rendered inaccurate due to the accident (i.e. sampling from a hot or cold leg loop which may have a steam or gas pocket) describe the backup sampling capabilities or address the maximum time that this condition can exist.

BWR's should specifically address samples which are taken from the core shroud area and demonstrate how they are representative of core conditions.

Passive flow restrictors in the sample lines may be replaced by redundant, environmentally qualified, remotely operated isolation valves to limit potential leakage from sampling lines. The automatic containment isolation valves should close on containment isolation or safety injection signals.

(11)(b) A dedicated sample station filtration system is not required, provided a positive exhaust exists which is subsequently routed through charcoal absorbers and HEPA filters.

PSC RESPONSE

a) Consideration to the above items was given in the design of the post-accident sampling system, consistent with the guidance/recommendation extant at the time of construction. Please note again that at Fort St. Vrain, the normal sampling system and the post-accident sampiing system are identical. It is our understanding that Oak Ridge National Laboratory is currently evaluating this item for the Fort St. Vrain Station.

b) Positive exhaust from the sampling station exists and is routed through charcoal adsorbers and HEPA filters prior to discharge to the environment.

Pending the results of the Oak Ridge review, we feel that we have satisfied this criterion.

·Section 2.2.2.b

Technical Support Center Radiation Levels During DBA-1

The Technical Support Center (TSC) will be manned by personnel during the conduct of accident recovery operations. This Section (2.2.2.b) describes the expected radiation environment present within TSC spaces subsequent to the DBA-1 accident.

Assumptions

The assumptions delineating the accident scenario, primary coolant leakage rate and radionuclide source term development are those employed in the reply to Section 2.1.6.b. Additionally, the following assumptions were used to evaluate the radiation environment of the TSC:

1. The atmospheric dispersion of processed effluents from the plant vent stack are conservatively assumed to be subject to downwash meteorological conditions for the duration of the accident. The atmosphere dispersion factors are as follows:

	TSC, Persistant Atmospheric Dilution Factor (sec/m^3) Downwash Condition Wind Speed = 5m/sec*
most unfavorable 8 hour period	9.0 - 05
next most unfavorable 16 hours	5.4 - 05
for the 1 - 4 day period	2.7 - 05
for the 4-30 day period	1.0 - 05
for the 30 - 108 day period	4.3 - 06

* persistant down wash conditions for the accident duration are felt to be quite conservative, as FSAR reports down wash condition in only 18% of the long term meteorological data collected at FSV. (See FSV FSAR Section 2.8.3, Amendment 18, "Revised Supplemental Climatological Report.....", E. R. Reiter, 11.70).

2. For the estimation of inhalation dose, the breathing rate of personnel occupying the TSC was assigned at 10 m^3 per 8 hour work shift (i.e., 3.47 $\times 10^{-4}$ m^3/sec) consistent with ICRP-2 recommendations for the standard man. This conservative rate was applied to the total duration of the accident.

3. Fission products released from the reactor building vent (noble gases and iodine and particulates escaping filtration), are not assumed to be depleted by fallout enroute to the TSC's ventilation air supply intake. Vent filter efficiencies of 90% iodine and 95% particulates were used for the reactor building vent.

4. The TSC input ventilation air if filtered with HEPA and charcoal absorbers having the following filter efficiencies, noble gases 0%, Halogens and particulates 99%.

Technical Support Center

The dose rates in the Technical Support Center (TSC) include the contributions from the airborne activity in the reactor building atmosphere, accumulated activity present on plant ventilation filters, and the contribution of plume submersion of the TSC in reactor building vented gases. The PATH code was used to evaluate the contribution of the building atmosphere and plant ventilation source terms. The TDAC code was used to estimate the accumulated dose commitment to personnel manning the TSC from immersion in and inhalation of reactor building gases vented from the plant's vent stack.

Reactor Building Atmosphere and Vent Filter Source Terms

The dose rate in the TSC will arise in part from the airborne activity present in the reactor building and from reactor building ventilation filters loaded with radioactive particulates. However, the requirement of shielding for the filters to reduce the radiation levels in the control room will also limit the dose rate in the TSC. (See Section 2.1.6.b for a complete discussion of control room doses and plant vent filter shielding requirements).

As determined with the PATH code, the unshielded dose rate in the TSC is shown in Figure 1. It must be stressed that only the contribution from the reactor building atmosphere was considered. The peak dose rate in the TSC without any shielding is 120 mrem/hr, occurring at about one day into the accident.

Because of the excessive unshielded dose rate, shielding is required in order to allow continuous manned access. Figure 1 also presents the dose rate with one foot of concrete shielding for the TSC walls and ceiling. The peak dose rate of 120 mrem/hr in the unshielded case is reduced to 1.6 mrem/hr with shielding.

Plume Submersion of the TSC in Reactor Building Vented Gases

Under certain meteorological conditions, it is possible for radioactive gases released from the reactor building vent to be dispersed downward within the plant boundary by down wash meteorological conditions. This down wash condition, at FSV, can prevail in approximately 18% of the annually observed meteorological samples. With no down wash, the plume remains elevated and does not contact the ground within the exclusion boundary and hence does not significantly effect the TSC operation.

Estimates of the plume submersion dose commitment to TSC personnel have been made using the atmospheric pathway dose code TDAC. Figure 2 displays the total accumulated dose to the maximum exposed individual in the TSC over the course of the DBA-1 accident. For this assessment, the down wash condition is conservatively assumed to pervail over the total accident duration. Additionally, the TSC is assumed to be designed such that an activated charcoal and HEPA filter are installed in the TSC makeup air supply. This insures that potentially contaminated air from the environment is filtered to remove iodine and particulate radionuclides from TSC air inhaled by personnel. Without this filtration system, the inhalation doses in the TSC would be excessive.

The plume submersion pathway integrated dose reported in Figure 2 conservatively assumes continuous residency of the TSC by a single individual. In fact, these doses would be reduced for radiation workers due to shift limited operations.

Summary Results

The peak dose rates and total accumulated DBA-1 dose in the Technical Support Center are summarized below. Also indicated is the time at which the peak dose rate occurs following the accident.

SUMMARY TECHNICAL SUPPORT CENTER RADIATION ENVIRONMENT POST DBA-1

TSC Exposure Pathway (1)	Peak Dose Rate	Time of Peak	180 Day Accumulated Dose
Exposure to Reactor Building			
Airborne Activity			
Unshielded TSC	120 mr/hr (gamm)	1 day	37 rem
Shielded TSC	1.6 mr/hr (gamma)	1 day	470 mrem
Exhaust Gases (Down Wash Condition)			
Exposure to Reactor Building			
Noble gases	5 mr/hr (gamma)	1 day	340 mrem
Thyroid Inhalation (2)	--	--	2.1 rem
Bone Inhalation (2)	--	--	430 mrem

(1) Exposure contribution from the reactor building plant ventilation filters not included as the filters are expected to be properly shielded. (See Section 2.1.6.b.)

(2) TSC intake filters installed with efficiency of 0% noble gases, 99% all others.

Conclusion

The following conclusions are reached and are recommended for consideration in the design of the Technical Support Center.

1. The Technical Support Center should be shielded from reactor building atmosphere dose contribution.

2. HEPA and activated charcoal filters should be added to the TSC input (makeup) air system to reduce the inhalation dose commitment to TSC personnel from respirable radionuclides.

3. With shielding and filtration discussed above, accumulated dose to TSC personnel assuming continuous occupancy for 180 days are within NRC requirements of 5 rem whole body and 30 rem thyroid.

BY ED HICKS
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FORM NO. 372-30-2650

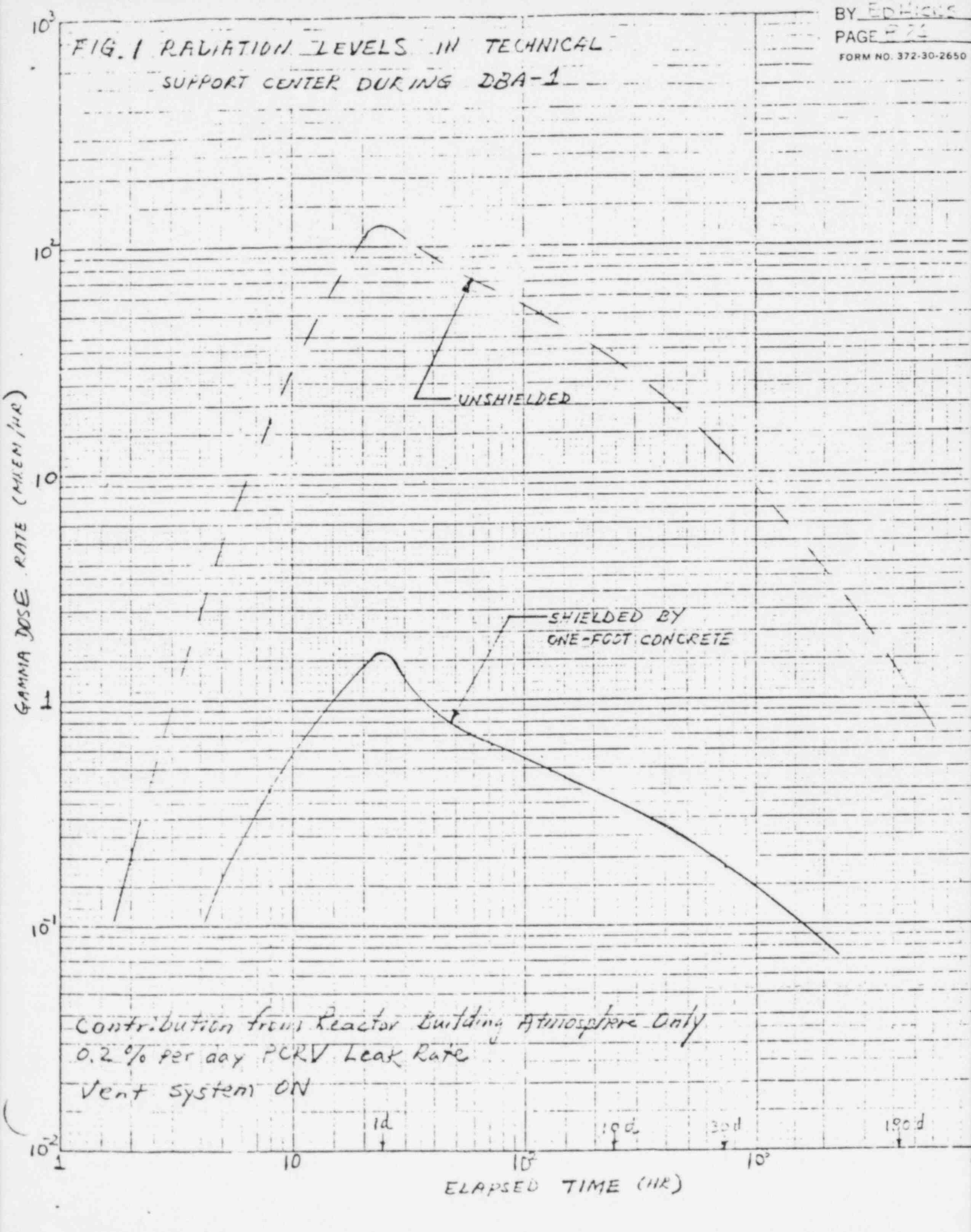
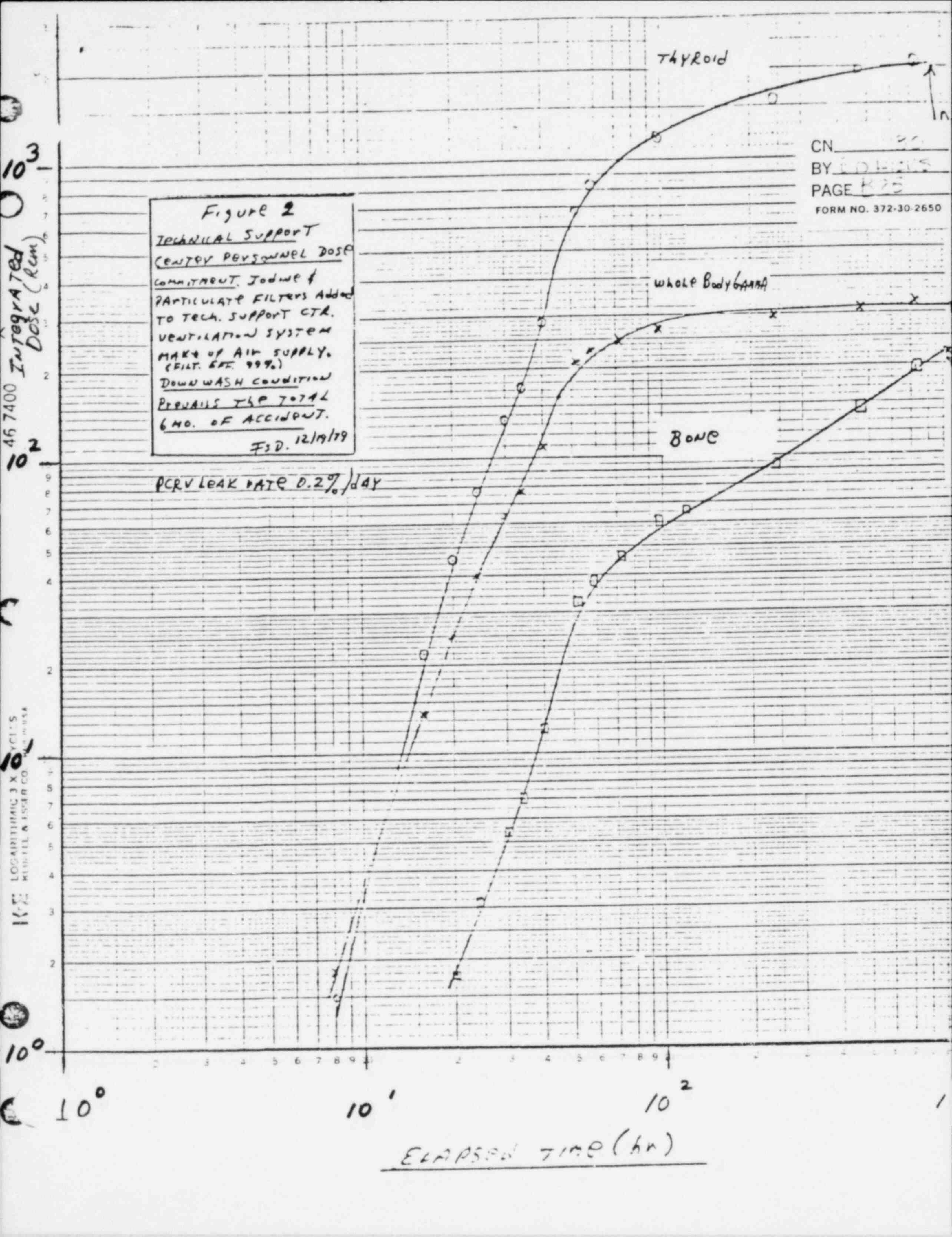


FIG. 1 RADIATION LEVELS IN TECHNICAL SUPPORT CENTER DURING DBA-1


THYROID
whole Body GAMMA
BONE
Figure 2
TECHNICAL SUPPORT CENTER PERSONNEL DOSE COMMITMENT. Iodine & PARTICULATE FILTERS Added TO TECH. SUPPORT CTR. VENTILATION SYSTEM MAKE UP AIR SUPPLY. (FILT. EFF. 99%) DOWNWASH CONDITION PREVAILS THE TOTAL 6 MO. OF ACCIDENT.
FSD. 12/19/79
PCRV LEAK RATE 0.2%/day
Integrated Dose (Rem)
ELAPSED TIME (hr)
10^0
10^1
10^2
10^3
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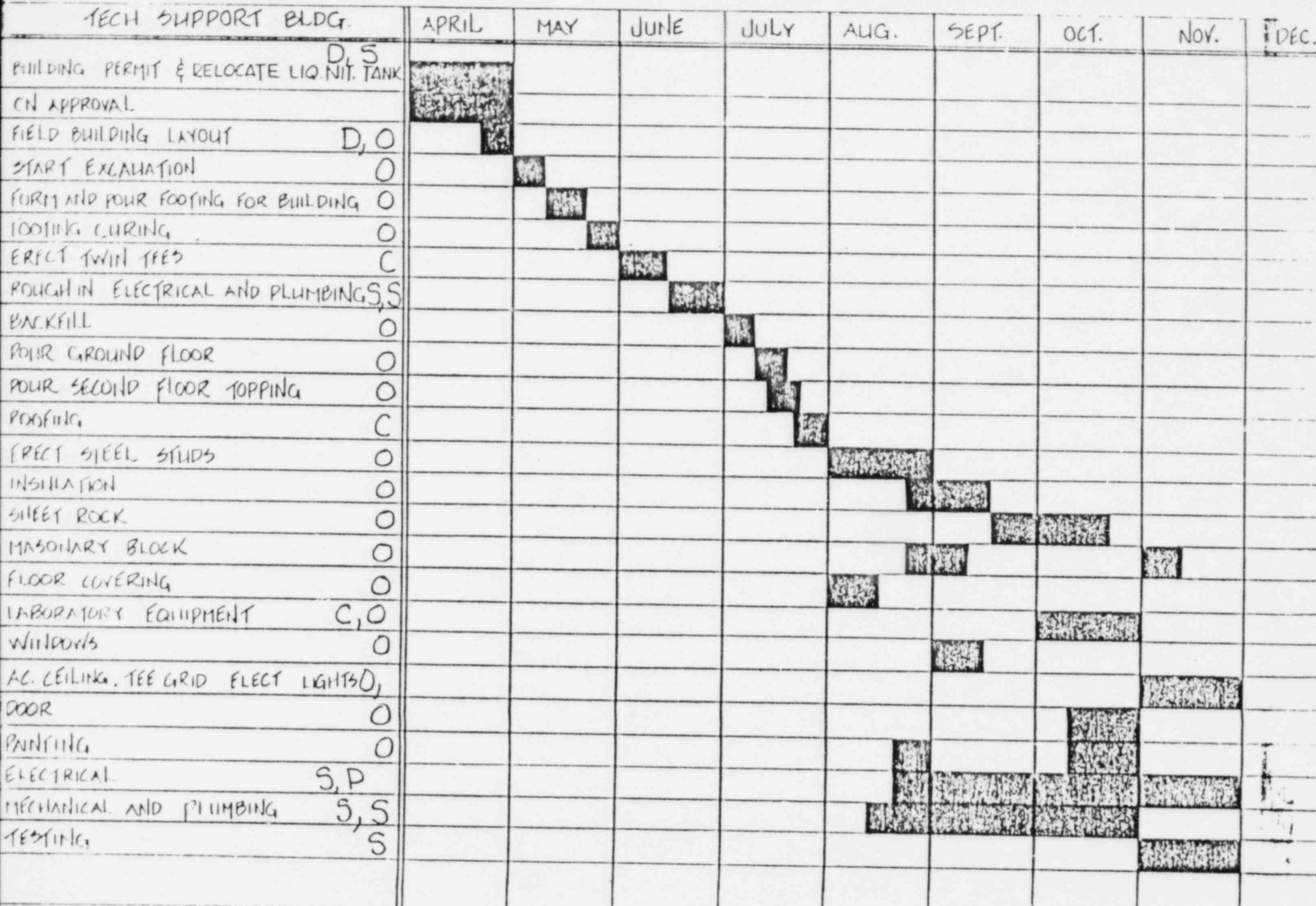

C = CONTRACTOR S = STEARNS ROGER D = DENVER ENGINEERING

C P.S.C. CONSTRUCTION P = P.S.C ANT ELECT