SNUPPS
Standardized Nuclear Unit
Power Plant System

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Nicholas A. Petrick
Executive Director

March 20, 1981

SLNRC 81-017 FILE: 0541
SUBJ: SNUPPS FSAR - NRC Request for Additional Information

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

Docket Nos.: STN 50-482, STN 50-483, STN 50-486

Reference: NRC (Tedesco) letter to J. K. Bryan and G. L. Koester, dated February 9, 1981, Same subject

Dear Mr. Denton:

The referenced letter requested additional information regarding the SNUPPS FSAR. The enclosure to this letter provides the requested information and will be incorporated into the next revision to the FSAR.

Very truly yours,

Nicholas A. Petrick

RLS/srz

Enclosure

cc: J. K. Bryan UE
G. L. Koester KGE
D. T. McPhee KCPL
T. Vandel NRC/Wolf Creek Site
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331.0 Radiological Assessment Branch

331.1 (12.1.2.5b) Section 12.1.2.5.b addresses a neutron shield design at the RPV in containment. Please specify the neutron and gamma dose equivalent rates that will exist at specific locations within the various levels of containment prior to shield installation and after the shield is installed. A figure or table showing respective dose rates would be a suitable format. Describe your plan for neutron personnel dosimetry whenever an entry is made while the reactor is at power, the frequencies at which entries are made, and the number of people making these entries.

RESPONSE

The neutron shield design for the reactor vessel cavity consists of a water bag shield located in the reactor vessel flange area above the hot and cold leg nozzles, as shown in FSAR Figure 3.8-61a. Based on this design, an average neutron dose rate at the top of the refueling pool has been calculated to be 1.8 rem/hour, using the Morse Monte Carlo code (Ref. 1). Dose rates in other areas of the containment were estimated using Cain's Hypothesis (Ref. 2) along with actual dose rate measurements (Ref. 3) taken at the Farley Nuclear Plant by Lawrence Livermore Laboratories (LLL). The dose rate values obtained using this technique are given below.

Location	Neutron Dose Rate (mrem/hr)
Equipment hatch	8-31
Personnel hatch	56

D. E. Hankins and R. V. Griffith (Ref. 3) of LLL found that the neutron-gamma dose rate ratio in the Farley containment was 7:1. Based on this ratio, the SNUPPS gamma dose rates are expected to be as follows.

Location	Gamma Dose Rate (mrem/hr)
Top of refueling pool	260
Equipment hatch	1-4
Personnel hatch	8

The neutron dosemetry method will comply with Revision 1 of Regulatory Guide 8.14. Exposures will be determined by film and/or time-dose calculations, using rem meters. There are no specific requirements for personnel entry into the containment during normal operating conditions. The frequency of entries will be based on operational needs and indications of abnormal conditions within the containment.

Entries into the containment when the reactor is at power will be made by at least two persons, one of whom will provide health physics surveillance.

REFERENCES

1. Straker E. A., Stevens P. N., Irving D. C., and Cain V. R., "The MORSE Code -- A Multigroup Neutron and Gamma-Ray Monte Carlo Transport Code," ORNL-4585, September, 1975.

2. Hopkins W. C., "Calculations of the Neutron Environment Inside PWR Containments," ORNL/RSIC-43, Page 127, February, 1979.

3. Hankins D. E. and Griffith R. V., "A Survey of Neutrons Inside the Containment of a Pressurized Water Reactor," ORNL/RSIC-43, Page 114, February, 1979.

331.2 (12.2.1.3) Radiation levels in excess of 100 R/hr can occur in the vicinity of spent fuel transfer tubes; therefore, all accessible portions of the transfer tubes must be shielded during fuel transfer. Please address the manner in which shielding, access control and radiation monitoring will be incorporated into the radiation protection program to prevent either occupants or transient workers from receiving very high exposures during transfer of spent fuel from the reactor to the spent fuel pool through the fuel transfer tubes. Use of removeable shielding for this purpose is acceptable. Provide appropriate figures (e.g. plan and elevation) that show the shielding arrays for all direct gamma radiation and streaming pathways from the spent fuel during the transfer. On the same figure show the location of any administrative controls by barriers, signs, audible and visual alarms, locked doors, etc. All accessible portions of the transfer tubes that cannot be adequately shielded shall be clearly marked with a sign stating that potentially lethal fields are possible during fuel transfer.

RESPONSE

The fuel transfer tube is completely shielded with permanent shielding to within radiation zone limits. No special access control, radiation monitoring, or posting is required.

The expansion bellows for the fuel transfer tube are under water in the fuel transfer canal in the fuel building (see revised Figure 3.8-48). There is no bellows inspection room or opening. The fuel transfer tube is completely surrounded by concrete or water, with the exception of the seismic gaps, so that no personnel access is possible.

The fuel transfer tube in the seismic gap between the containment wall and the internal containment structure and in the seismic gap between the containment wall and the fuel building is shielded, using permanently installed lead loaded silicone foam rubber, to meet the radiation zone limits (Figure 12.3-2). Therefore, there is no unshielded portion of the fuel transfer tube.

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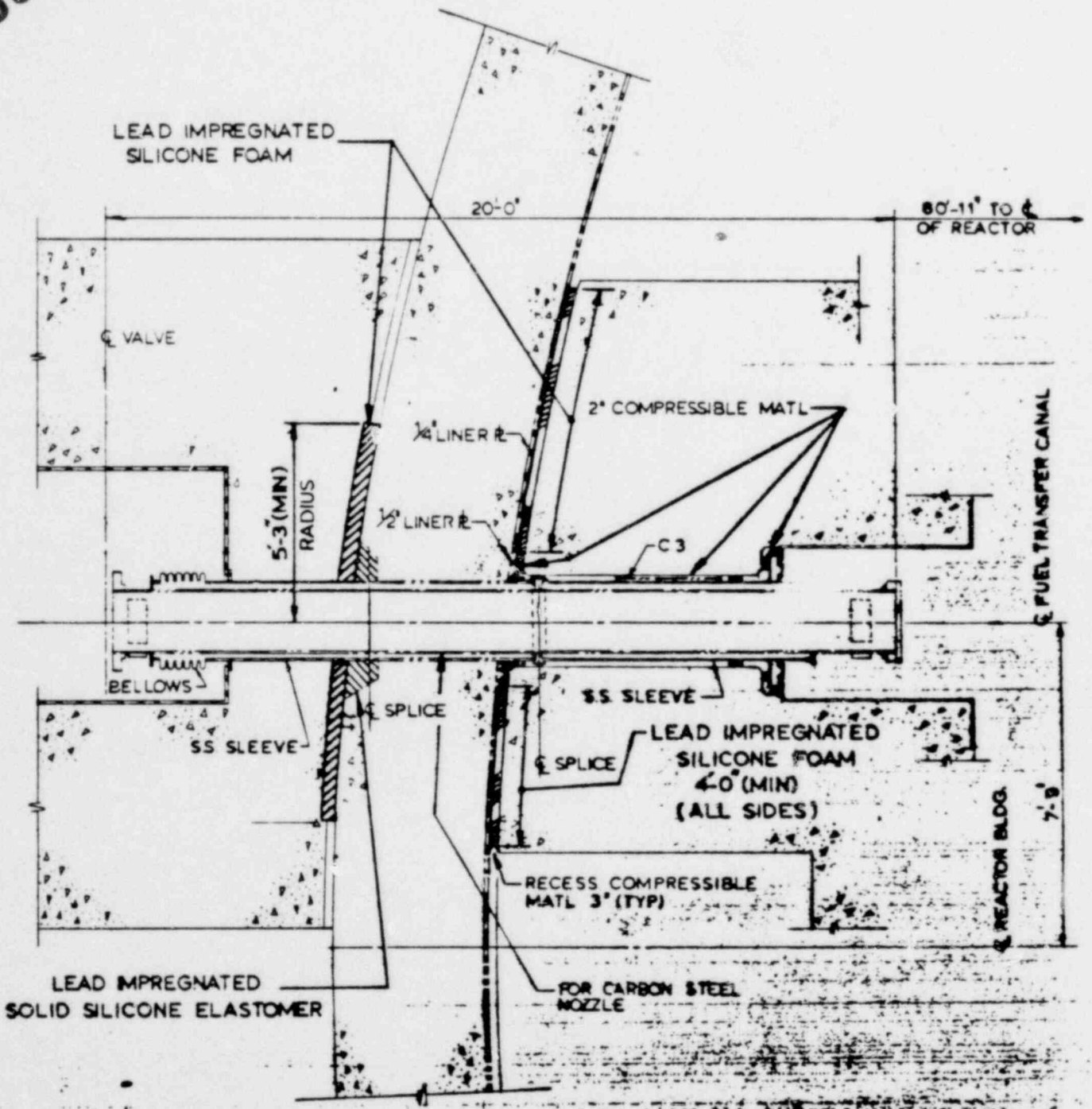
FIGURE 3.8-48

REACTOR BUILDING
FUEL TRANSFER PENETRATION



FUEL TRANSFER TUBE PLAN

331.3 (12.2.1.2.3) Describe the procedure for extracting a sample from the Nuclear Sampling System of RCS, RHR and CVCS with as low as is reasonably achievable exposures to personnel withdrawing the sample. In your response include use of shielding, area monitoring, portable survey meters, hand contact with sample containers, dose rate levels in sampling area, dose rate level of sample container, etc. Consider samples taken during normal operations, anticipated operational occurrences and accidents. The response to this question should satisfy the requirements of NUREG-0578 item 2.1.8.a, Post Accident Sampling, with regard to Radiation Protection.

RESPONSE

The sampling systems for (a) reactor coolant, (b) containment sump water, and (c) containment atmosphere are being modified to meet the guidelines of NUREG-0737, Item II.B.3. A subset of these modifications pertains to the piping systems shown in Figures 9.3-2 and 6.2.5-1. These changes will enhance the capability to collect samples in a post-accident situation.

In addition, a new post-accident sample analysis system is being procured. The sample panel for this system is to be located in the auxiliary building at El. 2000, in the room where the boronmeter is located. However, the post-accident sampling system is to be functionally independent of the boronmeter. The post-accident dose rate in this location has been calculated to be less than 10 r/hr.

The new system will be operable from a remote location, either in the control building or computer room, and will have the capability of in-line radionuclide spectral and chemical analysis of each of the three types of samples. The chemical analyses will include those specified in NUREG-0737, Item II.B.3. The capability of collecting shielded grab samples from the sample panel will also be provided. A purchase specification has been written for this system, and bids are currently being evaluated.

An FSAR change containing more detailed information about the revised sample systems will be prepared after selection of the supplier for the in-line sample analysis system.

The existing sampling systems, which provide the capability to make required analyses under normal conditions, will be retained. Tables 9.3-3, 9.3-4, 9.3-5, and 9.3-6 list the various systems which are sampled. Figures 9.3-2, 9.3-3, and 9.3-4 show the piping and instrumentation diagrams for

the sampling systems. Process radioactivity monitors for the sampling systems are indicated in the tables and figures mentioned above. The area and airborne radioactivity monitors for worker protection are given in Section 12.3.4 and are shown in Figure 12.3-2.

331.4 (TABLE 12.2-7) Table 12.2-7 indicates the radionuclide concentration in the spent fuel pool (SFP) water. Relevant reactor operating experience shows that the ^{60}Co activity, from crud transferred to the SFP from the interchange of the primary coolant water during refueling, is several orders of magnitude greater than that shown in the table even after purification by the SFP clean-up system. Please justify the values given in the table for ^{60}Co, ^{58}Co, ^{134}Cs, and ^{137}Cs and show that these values will be retained after several years of reactor operation. Provide an estimate of the dose rate above the SFP during a refueling operation and for the period thereafter. Include in the estimate the effect on the dose rate of any radioactive equipment that might be stored therein.

RESPONSE

The radionuclide concentrations given in Table 12.2-7 were calculated based on the primary coolant activities given in Table 11.4-7 without consideration of the crud which could be released to the refueling pool and the spent fuel pool during refueling operations. Typical spent fuel pool concentrations at operating plants with similar cleanup systems are listed in revised Table 12.2-7.

As stated in Section 9.1.2, the plant is designed so that dose rates above the spent fuel pool will not exceed 10 mrem/hr during refueling and 2.5 mrem/hr for the storage period.

The above dose rates consider the contribution from spent fuel and the spent fuel pool water. No other radioactive equipment that would significantly contribute to the dose rate is stored in the pool.

12.2.1.2 Auxiliary Building

12.2.1.2.1 Residual Heat Removal System

The pumps, heat exchangers, and associated piping of the residual heat removal (RHR) system contain radioactive materials. For plant shutdown, the RHR pumps and heat exchanger sources result from the radioactive isotopes carried in the reactor coolant, discussed in Section 12.2.1.1.2, considering 4 hours of decay following shutdown. The radiation source terms for the RHR system are listed in Table 12.2-5.

12.2.1.2.2 Chemical and Volume Control System

The CVCS source activity is the reactor coolant inventory which is provided in Table 11.1-4. More than 1 minute of N-16 coolant activity decay is provided before the letdown line exits the containment, and, therefore, is not significant in determining shielding requirements for the CVCS equipment outside the containment.

Major equipment items include the letdown and excess letdown heat exchangers, mixed bed and cation bed demineralizers, reactor coolant filter, volume control tank, and charging pumps. The seal water subsystem for the reactor coolant pumps includes the injection and return filters and the seal water heat exchanger. The design activities of the CVCS components are listed in Table 12.2-6. Heat exchanger and piping activities are derived from primary coolant activities. Radiation sources in the various pumps are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

12.2.1.2.3 Nuclear Sampling System

The major radiation sources in the nuclear sampling system originate from the RCS, RHR, and CVCS systems. The greatest radiation exposure would be to personnel taking the samples. To minimize this exposure, an integral shield has been incorporated into the sampling station design.

12.2.1.3 Fuel Building

12.2.1.3.1 Spent Fuel Storage and Transfer

The predominant radioactivity sources in the spent fuel storage and transfer areas in the fuel building are the spent fuel assemblies. Spent fuel assembly sources are discussed in Section 12.2.1.1.5. For shielding design, the spent fuel pool assumptions are given in Section 9.1.2. The major radionuclide concentrations in the water are provided in Table 12.2-7.

TABLE 12.2-7

SPENT FUEL POOL WATER ACTIVITIES

Isotope*	Activities (μCi/cc)
Co-58	1.3E-04
Co-60	4.0E-04
Cs-134	2.0E-05
Cs-137	2.0E-04

*Other isotopes will be present in much lower concentrations.

331.5 (12.3.4.2.2.2.2) Please clarify how iodine radioactivity levels can be "inferred from the particulate and noble gas radioactivity levels" when monitoring the exhaust from the radwaste and auxiliary buildings as addressed in section 12.3.4.2.2.2.2 and 12.3.4.2.2.2.4.

RESPONSE

The referenced FSAR sections have been revised to indicate that particulate iodine radioactivity levels in the HVAC duct flow paths (prior to filtration and discharge to the radwaste building and auxiliary building vents) will be determined by periodic laboratory isotopic analyses of the particulate cartridge filters associated with monitors 0-GH-RE-22 and 0-GL-RE-60. Also, if required, grab samples will be taken at various room locations to locate the source of the release. Analyses of these grab samples will be made for particulate, gaseous iodine, and noble gas activity.

The rate of iodine radioactivity discharged from the radwaste building vent and the unit vent is continuously monitored by 0-GH-RE-10A and 0-GT-RE-21A. These instruments monitor particulates and gaseous iodine. Parallel monitors 0-GH-RE-10B and 0-GT-RE-21B monitor noble gases.

An alarm from either the particulate monitor upstream of the HVAC filters or the noble gas monitor downstream of the filters will indicate that an increase in airborne activity is occurring. Laboratory analyses of the cartridge filters from the continuous monitors and the grab samples would then be used to determine the level of gaseous iodine.

duct through an isokinetic nozzle, in accordance with ANSI Standard N13.1-1969, to ensure that a representative sample is obtained. After passing through the fixed filter detector assembly and the pumping system, the sample is discharged back to the duct. The cartridge filter will be removed periodically for laboratory isotopic analyses. Monitoring upstream of the filter adsorber provides the most rapid response to airborne radioactivity in the system.

The high and high-high alarms function to alert the operator to airborne particulate radioactivity in the radwaste building. Indication for this monitor is provided on the AiRMS CRT in the control room.

If required, the portable monitor described in Section 12.3.4.2.2.2.9 and/or grab samples will be utilized to determine airborne radioactivity levels and iodine concentrations in specific areas to aid in the determination of the source of the release.

12.3.4.2.2.2.3 Waste Gas Decay Tank Area Ventilation Exhaust Radioactivity Monitor

The waste gas decay tank area ventilation radioactivity monitor, 0-GH-RE-23, continuously monitors for gaseous radioactivity in the discharge duct from the waste gas decay tank area upstream of the radwaste building exhaust filter adsorber. The sample point provides rapid detection of a leak in the waste gas processing system and, in conjunction with the radwaste building exhaust radioactivity monitor and the radwaste building effluent monitor, helps localize the affected area in the event of an alarm on either monitor.

The sample is extracted from the exhaust duct and passed through the fixed volume noble gas detector assembly and the pumping system. Then the sample is discharged back to the duct. The high alarm provides indication of a leak in the decay tanks, compressors, piping, or valves. The high-high alarm indicates that concentrations in the decay tank room are at or near 10 MPC for the most restrictive isotope expected to be present (Kr-85 or Xe-133).

Back up for this monitor is provided by the radwaste building exhaust and effluent monitors.

Indication of this monitor is provided on the AiRMS CRT in the control room.

12.3.4.2.2.2.4 Auxiliary Building Ventilation Exhaust Radioactivity Monitor

The auxiliary building ventilation exhaust radioactivity monitor, 0-GL-RE-60, continuously monitors for particulate radioactivity in the auxiliary building ventilation system upstream of the filter-adsorber units. The sample point is located to monitor between the last point of possible radioactivity entry to the ventilation system from the areas served and the filter adsorber unit. The sample is extracted through an isokinetic nozzle, in accordance with ANSI Standard N13.1-1969, to ensure that a representative sample is provided to the fixed filter particulate detector assembly. Then the sample is discharged through the pumping system back to the duct.

The cartridge filter will be removed periodically for laboratory isotopic analyses.

The high alarm alerts the operator to high airborne particulate radioactivity levels in the auxiliary building. Indication of this monitor is provided on the AiRMS CRT in the control room.

If required, the portable monitor described in Section 12.3.4.2.2.2.9 and/or grab samples will be utilized to determine airborne radioactivity levels and iodine concentrations in specific areas to aid in the determination of the source of the release.

12.3.4.2.2.2.5 Containment Atmosphere Radioactivity Monitors

The containment atmosphere radioactivity monitors, 0-GT-RE-31 and 0-GT-RE-32, continuously monitor the containment atmosphere for particulate, iodine, and gaseous radioactivity which could result in personnel exposure during periods of containment access. Other functions of these monitors are covered in Sections 5.2.5, 7.3, 9.4, and 11.5.

Samples are extracted from the operating deck level (El. 2047'-6") of the containment through the monitoring system sample lines.

The monitors are located as close as possible to the containment penetrations to minimize the length of the sample tubing and the effects of sample plateout. The sample points are located in areas which ensure that representative samples are obtained. Each sample passes through the penetration and then through the fixed filter (particulate), charcoal filter (iodine), and fixed volume (gaseous) detector assemblies. After passing through the pumping system, the sample is discharged back to the containment through a separate penetration.