APPENDIX 9

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QUESTION 9A.1 Discuss the consequences of a small leak and a large line break within the containment in the closed loop nuclear service system after an accident. Would the source of a large line break be detectable in time to prevent major loss of inventory from the system and resulting dilution of the borated recirculation water in the containment if the only indication is surge tank level?

ANSWER The safeguards cooling water system is shown in Figure 9.3-4 of the PSAR. This system is divided into two full capacity systems and each has a finite inventory, approximately 15,000 gallons of water. This system provides cooling water to the Reactor Building Emergency Cooling Units. 2

Monitoring alarms and level instrumentation on the surge tank of the safeguards cooling water system will provide for (1) leak detection in the entire system and not just within containment and (2) detection of small leaks.

In addition flow orifices are provided in each of the reactor building emergency cooling units. A difference in the differential pressure measurements between the orifices will indicate a leak and initiate an alarm in the control room. This signal will alert the control room operator and permit a check of the surge tank level and appropriate correction action. Should the difference in flow measurements indicate a major rupture, the individual coil would be isolated.

The entire inventory of water in each safeguards cooling water 2
closed loop cooling system would not sufficiently dilute the borated water being circulated following an accident to permit an approach to criticality. An analysis has shown that the inventory of approximately 15,000 gallons would dilute the 2
circulating borated water to a boron level of 1880 ppm. The preliminary design of the safeguards cooling water system shows that its contents are not sufficient to dilute the borated water concentration below the levels discussed in section 3.2.2.1.3 of the PSAR for the BOL case with all control rod assemblies stuck-out.

With the design provisions incorporated in the nuclear closed loop cooling systems, leaks in the system will be readily detected and the integrity of the system insured.

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QUESTION 9A.2 (DRL 9.1)

Submit the design revisions for the cooling water systems that were described at our meeting on March 1, 1968.

ANSWER

Design revisions for the cooling water systems that were described at our meeting on March 1, 1968, have been incorporated into the PSAR. See Section 9.3 of the PSAR.

QUESTION 9A.3 (DRL 9.2)

Discuss the maximum extent (frequency and duration) to which reservoir make-up water will be used in the event of canal water supply system outage.

ANSWER

It is not considered likely that there would be an outage of the Folsom-South Canal within the forty year life of the Rancho Seco Nuclear Generating Station based on the operating history of the Bureau of Reclamation's canals in the Central Valley Project. There are over 350 miles of main canal in operation in this system. More than 150 miles of this system has been in operation over twenty years. To this date there has not been a failure in any canal in the Project that would have adversely affected the delivery of canal water to a critical load.

The reliability of the District pumping station at the Rancho Seco turnout and the District pipeline between the canal and the on-site reservoir has been considered in evaluating this system. The pumping station will consist of four electrically driven pumps with the maximum capacity of any one pump equal to fifty percent of the total requirements. The station will receive power from two independent electrical sources. Based on the duplication of components in the system and the availability of replacement equipment, a failure of any component in the pumping station will be corrected within one week. The pipeline between the canal and the on-site reservoir will be a steel line with a proper protective coating to inhibit corrosion. Any failure on this line would be repaired within one week.

Since the on-site reservoir will contain in excess of thirty days make-up water supply for normal operation, and failure of the canal water supply system does not affect plant safety, SMUD believes the design of the canal water supply system is adequate.

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QUESTION 9A.4 (DRL 9.3) Discuss the plant's capability for detecting fuel failure. This discussion should include the detection time as a function of fuel fialure severity.

ANSWER It is planned to monitor the reactor coolant letdown flow using a scintillation counter (see Figure 11A.1-2). The sensitivity of the counter is 10^{-5} $\mu c/cc$ Co 60. It is anticipated that this will detect a major increase in activity but will not be capable of distinguishing between a corrosion product hurst or gross fuel failure. However, a reactor alarm will alert the operator to sample the reactor coolant to assess the source of the activity and determine the proper action to be taken. Further analysis is required as described in the Answer to Question 14A.29 to determine detection time as a function of fuel failure severity. 3

QUESTION 9A.5 (DRL 9.4) Submit a brief statement of your provisions in the emergency cooling water supply to cope with the lowest anticipated ambient temperatures ($\sim 19^oF$).

ANSWER The extreme low temperatures to be expected at the Rancho Seco site are generally night time lows and of short duration. On the basis of the 30 year average (1931-1960) for the month of January, the average minimum daily temperature is $+37^oF$ and the average monthly temperature is $+45^oF$. The median of annual extreme low temperature is $+24^oF$. It should be noted that on the average, the air temperature drops to $+32^oF$ or below an average of 56 hours a year and only 22 hours of that is the temperature below $+30^oF$. In view of the very short duration of subfreezing weather and the large spray pond depth it is considered unlikely that the water will freeze.

In the remote possibility that minor freezing should occur, the nuclear service water pumps will be started and pond water recirculated to promote mixing and thereby prevent surface freezing. As the spray nozzles will be bypassed in this event, and are normally automatically drained, freezing of the nozzles cannot occur.

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QUESTION 9A.6 (DRL 9.5)

Discuss the provisions for draining the spent fuel pool.

ANSWER

There will be no permanent provisions made for draining the entire spent fuel storage pool. The spent fuel cooling water pump suction is provided by a dip tube 15 feet below the surface and could be used to lower the level that far. Further dewatering would be done by portable submersible pumps. An installed drain system will not be provided for reasons of safety. The area enclosing the fuel transfer tube and carriage mechanism will be designed so that it may be isolated from the rest of the pool and dewatered to provide access to the equipment for repairs and maintenance.

QUESTION 9A.7 (DRL 9.6)

Discuss the potential for inadvertent draining of the spent fuel pool.

ANSWER

There will be no possibility for complete accidental draining of the spent fuel storage pool. All piping connections, other than the fuel transfer tube, will enter the pool at or above the water level to prevent gravity draining. Siphoning could, however, occur in two lines. The first is the spent fuel cooling water pump suction line, located 15 feet below the surface. However, even with the loss of 15 feet of water, 10 feet would remain above the top of the spent fuel assemblies which is sufficient to provide adequate cooling and keep the radiation levels within required limits. The cooling water pump discharge line, located at the bottom of the pool, will be provided with a check valve above the water level to prevent siphoning. The spent fuel transfer line is doubly protected by its block valve and blind flange against accidental draining of the spent fuel pool.

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QUESTION 9A.8 (DRL 9.7) Discuss the potential for inadvertantly draining the water in the fuel transfer canal and tube and specify the required fission product decay period after which the fuel elements do not require water cooling.

ANSWER Normal drainage of the fuel transfer canal will be through a 10" drain line located in a screened sump in the reactor internals laydown area. The shut-off valve for this line will be located on the underside of the transfer canal. This valve will be normally closed and locked shut and will be opened only after completion of refueling. In addition, the sump will have a removable stainless steel cover which will be lowered over the sump prior to filling the canal. A small vent hole will allow equalizing of pressures on both sides of the cover to permit its removal following refueling. In the unlikely event of failure of the shutoff valve, leakage will be limited to that through the small vent hole. The two 10" drain lines to the reactor cavity have a separate sump over which a plate will be bolted on prior to filling the canal and removed after the canal is empty. Because of the design of these outlets, it is considered that the fuel transfer canal can not be inadvertantly drained.

The fuel transfer tube will be drained by closing its gate valve in the spent fuel storage pool and allowing the water in the line to drain back into the fuel transfer canal as it empties.

Since the fuel pit cannot drain accidentally, the spent fuel assemblies will never be out of the water by accident. However, to be responsive to the AEC question, it has been estimated that a fuel element inside the fuel transfer tube must continue to be water cooled for about sixty (60) days after shutdown. This number was calculated using the fission product decay heat of the hottest bundle in the core at the hot rod maximum average burnup (Ref: pp. 3.2-60 - PSAR). The environment of the fuel element after a loss of water coolant was assumed to be air, and calculations were made for a range of times after shutdown until the requirement that the clad remain below failure conditions was met.

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