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December 19, 1985

Ref: LCR 85-18

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Washington, D. C. 20555

Attention: Mr. Steven A. Varga, Chief
Operating Reactors Branch 1
Division of Licensing

Gentlemen:

REQUEST FOR ADDITIONAL INFORMATION
SERVICE WATER HEADER OUTAGE
UNIT NOS. 1 AND 2
SALEM GENERATING STATION
DOCKET NOS. 50-272 AND 50-311

In our letter dated August 30, 1985, we transmitted copies of our request for amendment with the accompanying analyses associated with modifications to the requirements for residual heat removal system line-up, while in MODES 5 and 6. This amendment was requested to permit us to take one of two service water headers out of service for an extended period to permit detailed inspection and upgrading.

In your letter dated December 10, 1985, you requested additional information concerning the service water header outage. PSE&G's responses to these questions are provided as Attachment 1.

Additionally, we are providing as Attachment 2 revisions to our proposed change to the Technical Specifications. These revisions clarify which safety grade equipment is to be operable with one service water header out of service.

A001
1/1

ADD:
AD - J. KNIGHT (1tr only)
EB (BALLARD)
EICSB (ROSA)
PSB (GAMMILL)
RSB (BERLINGER)
FOB (BENAROYA)

Pursuant to the requirements of 10CFR50.91, a copy of these revised pages has been sent to the State of New Jersey as indicated below.

Sincerely,

C A McNeill Jr/JTB

Attachments

C Mr. Donald C. Fischer
Licensing Project Manager

Mr. Thomas J. Kenny
Senior Resident Inspector

Mr. Samuel J. Collins, Chief
Projects Branch No. 2, DPRP
Region 1

Mr. Frank Cosolito, Acting Chief
Bureau of Radiation Protection
Department of Environmental Protection
380 Scotch Road
Trenton, NJ 08628

Honorable Charles M. Oberly, III
Attorney General of the State of Delaware
Department of Justice
820 North French Street
Wilmington, DE 19801

ATTACHMENT 1

RESPONSE TO REQUEST FOR
ADDITIONAL INFORMATION
SERVICE WATER HEADER OUTAGE

ATTACHMENT 1

ADDITIONAL INFORMATION REQUEST
SALEM SERVICE WATER OUTAGE

QUESTION 1:

The Safety Evaluation provides descriptions of alternate cooling modes which would lead to containment contamination and radioactive releases to the environment and further relies on non-safety related equipment. These measures would be used as a backup in the event that the single remaining service water header failed. The staff believes that reliance on these measures should be minimized and that restoration of service water should be the first priority of plant personnel.

a) Describe in detail the inspection and upgrading operations that would be performed on the service water headers when taken out of service. At each stage describe the actions and time required to restore the affected service water header in the event that failure occurred in the operating header.

b) Describe procedures and training that will be implemented for emergency restoration of a service water header which is out for maintenance.

RESPONSE 1a:

The backup alternate cooling configurations identified and described in the Safety Evaluation provide defense in depth for residual heat removal. While these systems provide additional assurance that decay heat removal can be accomplished, they will not be relied on and the first priority of operating personnel will be to restore a decay heat removal path including the associated service water loop. The specific inspections to be performed will vary from cycle to cycle, depending on the areas vulnerable to erosion and corrosion and on what has been found during previous system inspections (prior refueling outages) and the operating experience during the last cycle. The specific upgrading and repair operations will depend on what is found during the inspections.

For the areas identified to be inspected, detailed inspections will be performed as discussed below. The piping will be disassembled and visually inspected from the inside for any pitting or liner damage. For some of the small bore piping, a boroscope, TV camera or similar device will be used to inspect the piping. In cases where the pipe lining is worn away and pitting is found on the pipe wall, the pitting depth will be measured. For cases where pitting depth is significant, ultrasonic tests will be performed to determine pipe wall thickness. If significant pipe wall thinning exists, weld repairs will be performed on the piping, followed by

replacement of the pipe lining. For cases without significant thinning, the lining will be repaired as necessary. Some of the lines will also be backflushed to remove sediment buildup. In addition to piping, several valves will also be removed, inspected and repaired or replaced as necessary.
It should be noted that a key objective of the Outage Planning process is to perform the inspections in a sequence which minimizes the time required to have a service water header out of service. In addition, components that are due to be replaced or are considered to have a high potential for significant erosion or corrosion are ordered well ahead of the outage to ensure their availability and prevent unnecessary delays in returning the service water header to service.

The areas to be inspected include both small and large bore piping. Some of the areas, especially the large bore piping, require significant time for disassembly, inspection and reassembly. In particular, some of the steps that must taken prior to initiating the inspection include tagging out the system, draining the system down for entry, removing valves and large segments of piping, placing blank flanges on the piping from the intake structure, setting up support equipment, and the placement of temporary sump pumps. As mentioned above, the inspections to be performed will vary from cycle to cycle. For the next refueling outage on Salem Unit 1, the areas to be inspected include:

1. Large bore piping in the service water bays including two 30" distribution headers and 20" branch connections.

2. Underground piping from the service water intake structure (divers will be used for these inspections).

3. 24" diesel lube oil and jacket water supply lines.

4. Several spool pieces in the auxiliary building including the 16" fan coil supply and return headers, 10" return piping from the fan coil units, 8" supply and return piping for the diesel lube oil and jacket water coolers, elbows on the inlet and outlet of the component cooling heat exchanger, the baffle plates in the CCW heat exchanger, 14" and 20" piping in the vicinity of the CCW heat exchanger and the emergency supply to the auxiliary feedwater system.

5. Several of the large strainers on the discharge of the service water pumps will be completely disassembled.

6. Selected small bore piping to and from motor coolers of the fan coils.

The actions required to restore the service water header at any point in time would simply be to terminate ongoing inspections and replace the piping and components as quickly as possible. However, as discussed below in response (1b), priority will be given to restoring the intact service water loop. If this cannot be done in a short period of time, the service water loop out for maintenance will be restored to service.

The time required to return the service water header back to service will vary depending on what inspections are being conducted and the stage of the inspections. The longest time required would be when the large bore piping and associated components are disassembled and inspections are underway. The best estimate to restore a service water header at the worst point in time is three days, given the scope of inspection planned for the next refueling outage.

RESPONSE 1b:

Due to the elimination of potential single failure points, the only single failure that would lead to a full loss of service water is a large pipe rupture, which is considered very unlikely. The more likely passive failure would be a leak in the piping. A piping leak would result in loss of some service water but would not completely defeat the system so that decay heat removal could be sustained. If a leak should occur, it can be repaired with lead or rubber patches without taking the service water loop out of service. Experience at both nuclear and fossil plants has demonstrated that this type of temporary repair can be easily performed and will last for several weeks. Therefore, it is extremely unlikely that there will ever be a need for emergency restoration of the service water header out for maintenance.

If service water should be lost or degraded, the failure mode of the service water system will influence the course of action associated with recovery of service water. This plan of action will be developed following an evaluation of the failure and failure mode including an assessment of the time and resources required to recover the service water system. It is expected that the least time required will be associated with recovering the loop of service water which was operating. Therefore, the initial emphasis will be to restore the intact loop to service. The decisions to deploy alternate cooling methods and to restore the service water loop that is out for maintenance will be based on comparisons of the estimated time to recover the service water loop versus the time to significantly reduce the coolant inventory above the core.

As discussed in the Safety Evaluation, a new emergency procedure is being developed that addresses loss of service water in the proposed configuration. This procedure will provide guidance to address the restoration of the service water loop out for maintenance, as discussed above. Operators will be trained on this procedure as discussed in the response to Question 4 below.

QUESTION 2:

The Safety Evaluation for the proposed change states that single service water header availability will only occur for a 32 day period during anticipated refueling outages. Since part of the justification for the proposed change is that the time interval will be relatively short, the staff requires that the outage time for a service water header be provided and included in the proposed technical specifications.

RESPONSE:

Based on discussions between the reactor systems branch reviewer and the licensee, the intent of this proposed staff requirement is to limit the amount of time that the plant could be in a degraded state, relying on the make-up and boil-off processes for decay heat removal. The 32 day period mentioned above represents the total time that one service water header is expected to be out of service during the upcoming refueling outages for each Salem Unit (each header will be out for 16 days). However, for a portion of this time period the current Salem Technical Specifications already allow one service water header to be out of service due to the refueling cavity being filled or the RCS loops being full of water. Therefore, it is not appropriate to include the 32 day time period in the Technical Specifications.

As discussed in the Safety Evaluation, it is very unlikely that the plant will ever be in a state where the make-up and boil-off process will be required to remove decay heat. Compensatory action will be taken to eliminate all credible active failure points (valves disabled or locked in position). Thus, even with one service water header out of service for maintenance, there is no single active failure that would be expected to defeat residual heat removal. Then, the only thing that can result in a loss of residual heat removal is a passive failure (e.g., large service water pipe rupture) which is considered very unlikely.

Even if normal residual heat removal capability were lost, alternate short term means of heat removal, via use of the water in the refueling water storage tank and the spent fuel pit as heat sinks, have been identified. This alternate means of heat removal will provide the necessary decay heat removal capability for a period ranging from five to nine hours when the water level is at the centerline of the nozzle. In addition, while this short term heat removal process is being employed, the reactor coolant system can be filled up to the level of the reactor vessel flange (if a steam generator manway cover is removed, it can be installed in a relatively short time period before the short term heat removal capability is exhausted). This would provide an additional heat sink for decay heat removal and will extend the time to core uncovery, in the absence of normal residual heat removal capability, by several hours. Thus, if the normal decay heat removal path is lost for any reason, a significant amount of time exists to restore the normal residual heat removal process before the make-up and boil-off process would be needed.

It is unlikely that the make-up and boil-off process will ever be required to remove decay heat. In the unlikely event that this process is required, the length of time it will be utilized will also be limited since the recovery of a service water loop even from pessimistic conditions can be accomplished in under ten days. Adequate make-up water exists between the refueling water storage tank, primary make-up tank and demineralized water tank to support the make-up and boil-off process for this entire ten day period. In most cases, the time to restore a service water header will be significantly less than ten days.

QUESTION 3:

The proposed technical specifications provide for additional equipment to be operable during the period when one service water header is out of service. Options are provided that either (a) two steam generators will be operable or (b) redundant RHR, CCW and service water pumps will be available. Service water is required to cool the auxiliary feedwater pumps, which would be required for the first option. Provide justification for not requiring redundant service water pumps for option (a). Discuss operability limitations for the AFW pumps without service water.

RESPONSE:

The proposed Technical Specification allows for the following available equipment to be substituted for one residual heat removal loop (or one service water header out of service):

a) Four filled reactor coolant loops, with at least two steam generators with their secondary side water levels greater than or equal to 5% narrow range; or,

b) Two RHR pumps and heat exchangers, two CCW pumps, two service water pumps and two ECCS pumps capable of supporting the make-up and boil-off method of heat removal.

Option (a) above is identical to what is contained in the current Salem Plant Technical Specifications and does not represent any change from what is already allowed. It should be noted that Option (a) does not require the steam generators to be fully operable. It only requires that a minimum amount of water inventory be available in the reactor coolant system to provide a heat sink redundant to the intact residual heat removal loop.

Water can be supplied to the steam generators by the condensate pumps, if available, or by the motor driven auxiliary feedwater pumps. Service water is not used to cool the auxiliary feedwater pumps directly but is used to supply the pump room coolers. These pumps can operate within design limits without room coolers for many hours.

QUESTION 4:

The Safety Evaluation states that emergency procedures will be written to include steps to be taken in the event that all service water were lost. Provide the timetable for implementing the revised procedures and associated operator training relative to the time when the inspection and upgrading operations on the service water system would be performed.

RESPONSE:

The emergency procedures to address loss of all service water are currently under development. Prior to the next Salem Unit 1 refueling outage (mid-March, 1986), the procedures will be completed and all operators will be trained on the new procedures.

QUESTION 5:

The Safety Evaluation states that control rod withdrawal accidents cannot occur during the proposed operations, since the control rods will not be energized. We believe that this provision should be included in the technical specifications or that a safety analysis be provided demonstrating that the plant can withstand a postulated control rod withdrawal accident during mode 5 and 6 operation.

RESPONSE:

As discussed in the Safety Evaluation, a new procedure is being developed to identify the steps that must be taken prior to entering into the desired configuration (water level at the nozzle centerline with one service water header out for maintenance, modes 5 and 6). Normally, the control rods are de-energized in modes 5 and 6 during a refueling outage. However, a provision will be added to this procedure that requires the control rods to be de-energized.

QUESTION 6

The Safety Evaluation discusses a short term means of cooling the core if all service water were lost, by which water from the RWST would be pumped into the reactor system and back into the RWST until the 120°F limit on RWST temperature were reached. S.ice the RWST is vented, radioactive releases to the public might result from recycling reactor coolant. Assuming maximum coolant radioactivity, demonstrate that the offsite radiation release would be acceptable.

RESPONSE TO (6)

The dose calculation assumptions, methodology and results are summarized below.

ASSUMPTIONS & GIVENS:

1. Total Reactor Water Volume:
   a. Nozzle Centerline 2929 ft^3*
   b. Total Vessel 4945 ft^3

   4945 ft^3 used for conservatism (Highest Reactor Water Concentration and volume).
2. Total RWST usable water volume (Minimum Tech Spec) 342,500 gals.
3. Tech Spec Reactor Coolant Activity Used. (As Obtained From Calculated Source Term, Table 1). Extremely conservative, assumes no stripping.

   5.3E1 uCi/cc Noble Gas, 7.8E-1 uCi/cc (Total Iodine)

4. For calculations, the dose rate conversion factor for I-131 was used.
5. For conservatism, core mix assumed in reactor coolant.
6. 2600 gpm flow rate of circulated water.
7. Maximum duration given at 9 hours (maximum time that plant would be using the RWST method of core cooling).

MAXIMUM ACTIVITIES IN REACTOR COOLANT

System Volumes

$$4945\ ft^3 \left[\frac{28320\ cc}{ft^3}\right] = 1.4E8\ cc \qquad 342{,}500\ gal \left[\frac{3785\ cc}{gal}\right] = 1.3E9cc$$

Activities

$$1.4E8\ cc \left[\frac{5.3E1\ uCi}{cc}\right] = 7.4E9\ uCi \text{ Total Noble Gas Activity in vessel water}$$

$$1.4E8\ cc \left[\frac{7.8E-1\ uCi}{cc}\right] = 1.1E8\ uCi \text{ Total Iodine Activity in vessel water}$$

* Reference Salem Generating Station UFSAR, Section 5, Rev. 0, July 1982.

TABLE 1

ACTIVITY AND NATURE OF RADIONUCLIDES ASSUMED IN SOURCE TERM
NUCLIDE MIX AFTER DECAY (96 HRS) CORE MIX INVENTORY ASSUMED

Nuclide	Core Inventory (Ci)	Relative Fraction	Operational Chemistry Conc. (uCi/g) at Shutdown	Tech Spec Conc. (uCi/g) at Shutdown	(Hrs^{-1})	Tech Spec Conc. (uCi/g) After Decay
KR-85*	3.9E5	8.0E-4	4.8E-5	1.7E-1	7.4E-6	1.7E-1
KR-85m	1.8E7	3.7E-2	2.2E-3	7.9E0	1.5E-1	------
KR-87	3.2E7	6.5E-2	3.9E-3	1.4E1	5.5E-1	------
KR-88	4.6E7	9.3E-2	5.6E-3	2.0E1	2.4E-1	------
Xe-133*	2.0E8	4.0E-1	2.4E-2	8.5E1	5.5E-3	5.0E1
Xe-133m*	2.8E7	5.8E-2	3.5E-3	1.2E1	1.3E-2	3.4E0
Xe-135*	1.3E8	2.6E-1	1.5E-2	5.4E1	7.6E-2	3.7E-2
Xe-135m	4.3E8	8.7E-2	5.2E-3	1.8E1	2.7E0	------
Totals		1E0	6.0E-2	2.1E2		5.3E1
I-131*	1.0E8	1.2E-1	5.0E-3**	1.0E0**	3.6E-3	7.1E1
I-132	1.5E8	1.8E-1	7.5E-3	1.5E0	3.0E-1	------
I-133*	2.0E8	2.6E-1	1.1E-2	2.0E0	3.5E-2	6.9E-2
I-134	2.0E8	2.6E-1	1.1E-2	2.0E0	7.9E-1	------
I-135	1.8E8	2.2E-1	9.2E-3	1.8E0	1.1E-1	------
Totals		1E0	4.4E-2	8.2E0		7.8E-1

* Predominant nuclides after 96 hour decay

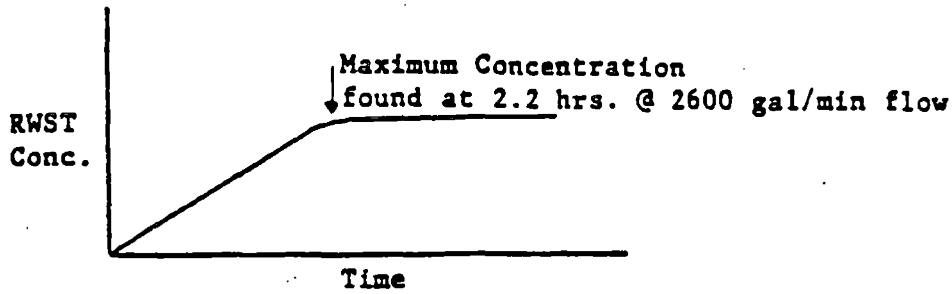
** For Dose Calculation, the dose rate conversion factor for I-131 was used (Conservative DEI). Other nuclides given for illustration only.

Reference Salem Generating Station, UFSAR, Section 15.4, Rev. 1, July 1983.

MAXIMUM CONCENTRATION IN RWST

(Initial RWST concentration neglected, no decay during dilution credited).

Maximum concentration will be found when both systems (vessel water and storage tank water) are in equilibrium.



MAXIMUM CONCENTRATION IN RWST GIVEN AS:

$$\frac{7.4E9 \text{ uCi}}{(1.4E8 + 1.3E9) \text{ cc}} = 5.1E0 \text{ uCi/cc} \quad \text{Noble Gas}$$

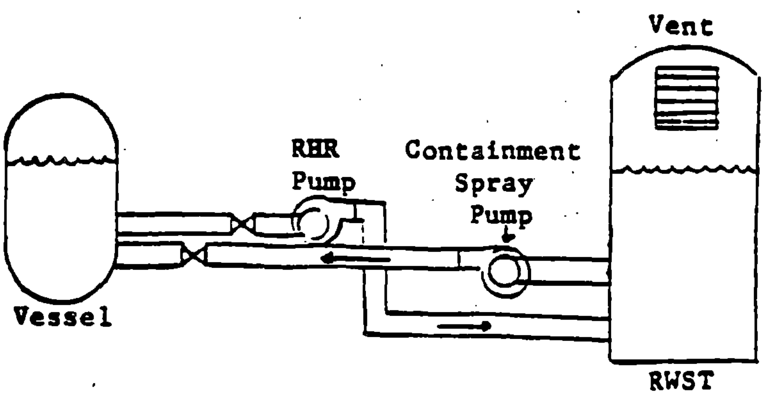
$$\frac{1.1E8 \text{ uCi}}{(1.4E8 + 1.3E9) \text{ cc}} = 7.6E{-2} \text{ uCi/cc} \quad \text{Total Iodine}$$

RELEASE RATE DETERMINATION

Note:

Since the vessel and tank structures are virtually a closed system the only postulated driving force for release through the RWST vent is vaporization. However, the water temperature will only be elevated to 120°F. It has been shown that water expands a small amount at this temperature so little vaporization would be expected.

Nevertheless, noble gases are expected to be released from the liquid. Even though there is no pressure differential, a highly conservative assumption will be made.



ASSUMPTION

Liquid flow (circulation) into the tank is given at 2600 gal/min. Assume that the Noble Gases and radioiodines from 10% of this flow are released to the tank free air space and are subsequently released to the environment. (Iodines undergo additional reduction prior to release, see below).

RELEASE RATES

$$\left[\frac{260\ \text{gal}}{\text{min}}\right] \left[\frac{1\ \text{min}}{60\ \text{sec}}\right] \left[\frac{3785\ \text{cc}}{\text{gal}}\right] \quad 1.6\text{E4} \quad \text{cc/sec}$$

NOBLE GAS RELEASE RATE

$$\left[\frac{5.1\text{E0 uCi}}{\text{cc}}\right] \left[\frac{1.6\text{E4 cc}}{\text{sec}}\right] = 8.2\text{E4 uCi/sec}$$

IODINE (DEI) RELEASE RATE

$$\left[\frac{7.6\text{E-2 uCi}}{\text{cc}}\right] \left[\frac{1.6\text{E4 cc}}{\text{sec}}\right] \underset{\text{Plate-out}}{\left[.1\right]} \underset{\text{Partition}}{\left[.1\right]} = 1.2\text{E1 uCi/sec}$$

PROJECTED DOSE RATES AT THE MINIMUM EXCLUSION AREA (MEA)

$$\left[\frac{8.2\text{E4 uCi}}{\text{sec}}\right] \left[\frac{3.45\text{E4 mrem/hr*}}{\text{uCi/cc}}\right] \left[\frac{1.2\text{E-6 sec**}}{m^3}\right] \left[\frac{m^3}{1\text{E6cc}}\right] = 3.4\text{E-3 mrem/hr Whole Body}$$

$$\left[\frac{1.2\text{E1 uCi}}{\text{sec}}\right] \left[\frac{2.0\text{E9 mrem/hr*}}{\text{uCi/cc}}\right] \left[\frac{1.2\text{E-6 sec**}}{m^3}\right] \left[\frac{m^3}{1\text{E6cc}}\right] = 2.9\text{E-2 mrem/Inhalation hr Thyroid}$$

* Noble Gas and Iodine DRCF's Post Decay. Reference Salem Generating Station, Emergency Plan Procedures, EP IV-111, Rev. 6, April 1985.

** Site Average X/Q. Reference Salem Generating Station UFSAR, Section 11.3, Rev. 0, July 1982.

TOTAL DOSE AT SITE BOUNDARY (9 HOUR DURATION)

$$\left[\frac{3.4E-3 \text{ mrem}}{\text{hr}}\right]\left[9 \text{ hrs}\right] = 3.1E-2 \text{ mrem Whole Body}$$

$$\left[\frac{2.9E-2 \text{ mrem}}{\text{hr}}\right]\left[9 \text{ hrs}\right] = 2.6E-1 \text{ mrem Thyroid}$$

CONCLUSION

Projected dose rates at site boundary calculated to be 3.4E-3 mrem/hr whole body and 2.9E-2 mrem/Inhalation hr Thyroid. The dose after 9 hours is 3.1E-2 mrem whole body and 2.6E-1 mrem Thyroid. This is well below Unit 1, Technical Specificatons, Section 3.11.2.1, Gaseous Effluent LCO's of 500 mrem/yr whole body, 1500 mrem/yr to any organ (Thyroid).

QUESTION 7

In the event that service water were lost for an extended period of time, the Safety Evaluation states that the core would be cooled by injection of RWST water which would be allowed to boil in the containment. Since service water would be required for containment heat removal, containment overpressure would have to be prevented by venting.

a. The Safety Evaluation states that a detailed dose calculation was performed and that the results were acceptable. Provide further information concerning the dose calculation including the initial coolant activity assumed, the activity and nature of the radionuclides assumed in the source term, and the dose calculated at the site boundary.

b. Provide analyses of the effect of continued boiling in the core on boron concentration within the coolant channels. If boron precipitation is calculated to occur, provide the effect on core heat transfer and fuel heatup.

RESPONSE TO (7a)

The dose calculation assumptions, methodolody and results are summarized below.

INITIAL COOLANT ACTIVITY ASSUMED:

Two cases examined

1. Operational Chemistry Data. That is, the anticipated RCS Activity Post Stripping before Mode 5

   Given As: 0.06 uCi/ml* Noble Gas, 0.005 uCi/ml* Iodine-131
   These values were obtained from actual 1984 RCS Chemistry Data. For conservatism, the highest concentrations in that period were used.

2. Tech Spec Limit Chemistry Data. This case is used as the upper conservative bound. RCS concentrations are postulated at the Tech Spec Limit (Section 3.4.8) and no stripping or off-gassing is considered. This is not however believed to be a realistic case as every effort will be made to lower coolant concentrations prior to system breach. In addition, most of the noble gas will be released from the coolant during depressurization.

   Tech Spec Limit is: $\leq 100\ \bar{\bar{E}}$ uCi/g and thus calculates to be approx. 211 uCi/g Noble Gas**, 1 uCi/g DEI

* For the purposes of this study, at 120°F (RCS Temp Limit) the specific volume of water at 1 ATM = 1.01208. At 0°F, 1 ATM =1.00017. Therefore, it is assumed 1 gm water = 1 cc = 1 ml. From ASME Steam Tables Fifth Ed., 1983.

** Reference LRC-85-03 letter to Mr. Steven A. Varga, Chief Operations Reactors Branch 1, Division of Licensing. Request for Amendment Facility Operating Licenses Unit Nos. 1 and 2, Salem Generating Station, Docket Nos. 50-272 and 50-311. Attachment 2, February 8, 1985.

TABLE 1

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NUCLIDE MIX AFTER DECAY (96 HRS) CORE MIX INVENTORY ASSUMED

Nuclide	Core Inventory (Ci)	Relative Fraction	Operational Chemistry Conc. (uCi/g) at Shutdown	Tech Spec Conc. (uCi/g) at Shutdown	(Hrs^{-1})	Tech Spec Conc. (uCi/g) After Decay
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KR-87	3.2E7	6.5E-2	3.9E-3	1.4E1	5.5E-1	------
KR-88	4.6E7	9.3E-2	5.6E-3	2.0E1	2.4E-1	------
Xe-133*	2.0E8	4.0E-1	2.4E-2	8.5E1	5.5E-3	5.0E1
Xe-133m*	2.8E7	5.8E-2	3.5E-3	1.2E1	1.3E-2	3.4E0
Xe-135*	1.3E8	2.6E-1	1.5E-2	5.4E1	7.6E-2	3.7E-2
Xe-135m	4.3E8	8.7E-2	5.2E-3	1.8E1	2.7E0	------
Totals		1E0	6.0E-2	2.1E2		5.3E1
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I-132	1.5E8	1.8E-1	7.5E-3	1.5E0	3.0E-1	------
I-133*	2.0E8	2.6E-1	1.1E-2	2.0E0	3.5E-2	6.9E-2
I-134	2.0E8	2.6E-1	1.1E-2	2.0E0	7.9E-1	------
I-135	1.8E8	2.2E-1	9.2E-3	1.8E0	1.1E-1	------
Totals		1E0	4.4E-2	8.2E0		7.8E-1

* Predominant nuclides after 96 hour decay

** For Dose Calculation, the dose rate conversion factor for I-131 was used (Conservative DEI). Other nuclides given for illustration only.

Reference Salem Generating Station, UFSAR, Section 15.4, Rev. 1, July 1983.

DOSE CALCULATED AT SITE BOUNDARY

ASSUMPTIONS & GIVENS:

1. Coolant Activity as given in source term for two cases Table 1.
2. 100 gal/min make-up and boil-off of reactor coolant.*
3. 9/10 reduction of iodine due to liquid to steam partition.
4. 9/10 reduction of iodine due to plate-out on containment surfaces.
5. 100% of noble gases produced escape without hold up, by containment systems. 95% of Iodines produced are retained by charcoal banks in the ventilation system before escape.
6. For Unit 1, Technical Specifications, Section 3.11.2.1 calculations, MET conditions given as site average X/Q (1.2E-6 sec/m^3). For 10 CFR 100, MET given as Accident X/Q (5.0E-4 sec/m^3).**
7. For calculations, the dose rate conversion factor for I-131 was used.
8. For conservatism, core mix assumed in reactor coolant.
9. At given conditions (120°F, 1 ATM Pressure) 1cc = 1ml = 1gm
10. Assumed maximum event duration given at 10 days.

NOBLE GAS AND IODINE RELEASE RATES

OPERATIONAL CHEM DATA

$$\left[\frac{100\text{ gal}}{\text{min}}\right]\left[\frac{1\text{ min}}{60\text{ sec}}\right]\left[\frac{3785\text{ ml}}{\text{gal}}\right]\left[\frac{.06\text{ uCi}}{\text{ml}}\right] = 3.8\text{E}2\text{ uCi/sec Noble Gas}$$

$$\left[\frac{100\text{ gal}}{\text{min}}\right]\left[\frac{1\text{ min}}{60\text{ sec}}\right]\left[\frac{3785\text{ ml}}{\text{gal}}\right]\left[\frac{.005\text{ uCi}}{\text{ml}}\right]\underbrace{\left[.1\right]}_{\text{Partition}}\underbrace{\left[.1\right]}_{\text{Plate-out}}\underbrace{\left[.05\right]}_{\text{Filtration}} = 1.6\text{E-}2\text{ uCi/sec Total Iodine}$$

TECH SPEC CHEM DATA (POST DECAY)

$$\left[\frac{100\text{ gal}}{\text{min}}\right]\left[\frac{1\text{ min}}{60\text{ sec}}\right]\left[\frac{3785\text{ gm}}{\text{gal}}\right]\left[\frac{5.3\text{E}1\text{ uCi}}{\text{gm}}\right] = 3.3\text{E}5\text{ uCi/sec Noble Gas}$$

$$\left[\frac{100\text{ gal}}{\text{min}}\right]\left[\frac{1\text{ min}}{60\text{ sec}}\right]\left[\frac{3785\text{ gm}}{\text{gal}}\right]\left[\frac{7.8\text{E-}1\text{ uCi}}{\text{gm}}\right]\underbrace{\left[.1\right]}_{\text{Partition}}\underbrace{\left[.1\right]}_{\text{Plate-out}}\underbrace{\left[.05\right]}_{\text{Filtration}} = 2.5\text{E}0\text{ uCi/sec Total Iodine}$$

* A flow of less than 100 gal/min is required to maintain a constant reactor vessel level at the nozzle centerline when in the makeup and boil-off mode of core cooling.

** Reference Salem Generating Station UFSAR, Section 11.3, Rev. 0, July 1982 and Section 15.4, Rev. 1, July 1983 respectively.

PROJECTED DOSE AT THE MINIMUM EXCLUSION AREA (MEA)

Dose Rate at Fence (0.79 miles)

OPERATIONAL CHEM DATA

$$\left[\frac{3.8E2\ \text{uCi}}{\text{sec}}\right]\left[\frac{3.45E4\ \text{mrem/hr}^{*}}{\text{uCi/cc}}\right]\left[\frac{1.2E{-}6\ \text{sec}^{**}}{\text{m3}}\right]\left[\frac{\text{m}^3}{1E6\text{cc}}\right] = 1.6E{-}5\ \text{mrem/hr}$$

Whole Body Dose Rate

$$\left[\frac{1.6E{-}2\ \text{uCi}}{\text{sec}}\right]\left[\frac{2.0E9\ \text{mrem/hr}^{*}}{\text{uCi/cc}}\right]\left[\frac{1.2E{-}6\ \text{sec}^{**}}{\text{m3}}\right]\left[\frac{\text{m}^3}{1E6\text{cc}}\right] = 3.8E{-}5\ \text{mrem/inhalation hr}$$

Thyroid Dose Rate

TECH SPEC CHEM DATA (POST DECAY)

$$\left[\frac{3.3E5\ \text{uCi}}{\text{sec}}\right]\left[\frac{3.54E4\ \text{mrem/hr}^{*}}{\text{uCi/cc}}\right]\left[\frac{1.2E{-}6\ \text{sec}^{**}}{\text{m3}}\right]\left[\frac{\text{m}^3}{1E6\text{cc}}\right] = 1.4E{-}2\ \text{mrem/hr}$$

Whole Body Dose Rate

$$\left[\frac{2.5E0\ \text{uCi}}{\text{sec}}\right]\left[\frac{2.0E9\ \text{mrem/hr}^{*}}{\text{uCi/cc}}\right]\left[\frac{1.2E{-}6\ \text{sec}^{**}}{\text{m3}}\right]\left[\frac{\text{m}^3}{1E6\text{cc}}\right] = 6.0E{-}3\ \text{mrem/inhalation hr}$$

Thyroid Dose Rate

* Noble Gas & Iodine DRCF's Post Decay. Reference Salem Generating Station, Emergency Plan Procedures, EP IV-111, Rev. 6, April 1985.

** Site Average X/Q. Reference Salem Generating Station UFSAR, Section 11.3, Rev. 0, July 1982.

TOTAL DOSE AT THE MINIMUM EXCLUSION AREA (10 DAYS MAXIMUM DURATION)

OPERATIONAL CHEM DATA

$$\left[\frac{1.6E\text{-}5\ \text{mrem}}{\text{hr}}\right]\left[\frac{24\ \text{hrs}}{\text{day}}\right]\left[10\ \text{days}\right] = 3.8E\text{-}3\ \text{mrem Whole Body}$$

$$\left[\frac{3.8E\text{-}5\ \text{mrem}}{\text{hr}}\right]\left[\frac{24\ \text{hrs}}{\text{day}}\right]\left[10\ \text{days}\right] = 9.1E\text{-}3\ \text{mrem Thyroid}$$

TECH SPEC CHEM DATA

$$\left[\frac{1.4E\text{-}2\ \text{mrem}}{\text{hr}}\right]\left[\frac{24\ \text{hrs}}{\text{day}}\right]\left[10\ \text{days}\right] = 3.4E0\ \text{mrem Whole Body}$$

$$\left[\frac{6.0E\text{-}3\ \text{mrem}}{\text{hr}}\right]\left[\frac{24\ \text{hrs}}{\text{day}}\right]\left[10\ \text{days}\right] = 1.4E0\ \text{mrem Thyroid}$$

ADDITIONAL CALCULATIONS

Total skin dose was examined for the worst case noble gas release rate. Results yield 3.5E-2 mrem/hr, total dose of 8.3 mrem for the maximum 10 day release.

In addition, an examination was made for the purposes of 10 CFR 100 limits study. The major differences from the above calculations involve the use of a specific accident X/Q (5.0E-4 sec/m^3). It should be noted that this accident X/Q is very conservative and is applicable to a 0-2 hour period. For a ten day evaluation, values as small as 5E-6 could be used. All other assumptions remain the same.

DOSES BASED ON 10 CFR 100 CRITERIA

	Operational Chem Data	Tech Spec Chem Data
Dose Rates At MEA	6.5E-3 mrem/hr Whole Body 1.5E-2 mrem/Inhalation hr Thyroid	5.8E0 mrem/hr Whole Body 2.5E0 mrem/Inhalation hr Thyroid
Total Dose At MEA	1.6E0 mrem Whole Body 3.7E0 mrem Thyroid	1.4E3 mrem Whole Body 6.0E2 mrem Thyroid

CONCLUSIONS

Doses for ten days were calculated to be:

3.8E-3 mrem	Whole Body	Operational Data
9.1E-3 mrem	Thyroid	

3.4E0 mrem	Whole Body	Tech Spec Data
1.4E0 mrem	Thyroid	
8.3E0 mrem	Skin	

These are found to be within the bounds of Unit 1 Technical Specifications, Section 3.11.2.1, Gaseous Effluent LCO's of 500 mrem/yr whole body, 1500 mrem/yr to any organ (Thyroid) and 3000 mrem/yr to the skin.

10 CFR 100 accidental release doses for ten days were calculated to be:

1.6E0 mrem	Whole Body	Operational Data
3.7E0 mrem	Thyroid	

1.4E3 mrem	Whole Body	Tech Spec Data
6.0E2 mrem	Thyroid	

These are all found to be within the bounds of 10 CFR 100 allowable limits of 25 rem Whole Body and 300 rem to the Thyroid.

RESPONSE 7B:

As discussed in the Safety Evaluation, new emergency procedures are being developed that identify the steps to be taken if a loss of the intact service water loop should occur. This procedure will include a provision that if the make-up and boil-off process is required for core cooling, the water source must be switched over from the refueling water storage tank (RWST) to the unborated primary water make-up supply before the core boron concentration reaches the solubility limit. This action will ensure that no boron precipitation will occur.

The maximum allowable RWST water inventory that can be injected into the core depends on the initial core boron concentration and the initial RWST boron concentration. The calculation will be based on the following assumptions:

1. The core temperature is 212°F;

2. The maximum allowable core boron concentration is 23.5 weight percent determined as follows:

Boron solubility limit (@ 212°F)	= 27.5 w/o
Less 4% to account for uncertainties	= 4.0
Maximum allowable concentration	23.5 w/o

NOTE: This approach is consistent with the NRC guidelines issued for determining the time to switchover to hot leg recirculation for the emergency operating procedures.

3. All boron is assumed to concentrate in the area between the lower core plate and the reactor vessel nozzles. No credit is taken for mixing in the downcomer or lower plenum.

If the initial boron concentration of the core and RWST is at 2200 ppm (technical specification limit for RWST), 122,500 gallons of RWST water can be injected into the core before having to switchover to primary make-up. The time to reach this switchover point will vary depending on the decay heat level in the core while in the make-up and boil-off mode of core cooling. Based on the decay heat generated four days after shutdown, the anticipated time at which the plant will enter into the proposed configuration (water level at nozzle centerline and one service water header out of service), the time available to reach this switchover point is greater than 24 hours. This time will increase as the time after shutdown increases due to decreased decay heat level (at 14 days after shutdown, the time would be greater than 45 hours).

The emergency procedures will require switchover to unborated primary make-up when the allowable RWST water inventory is depleted (as determined by the change in RWST water level). The allowable RWST water depletion will either be based on the limiting value discussed above or a cycle specific value based on the actual RWST boron concentration.

QUESTION 8:

The Safety Evaluation discusses use of temporary hose connections and portable fans to replace some of the functions of the service water system if required. It is further stated that a new procedure will be written which, among other things, will verify that portable fans are easily available. Will the procedure also verify that the temporary hose connections are also easily available?

RESPONSE:

The procedure discussed above will require that any temporary hose connections needed for pump operability are either readily available or installed prior to entering into the proposed configuration.

QUESTION 9:

If service water is lost, several days were stated to be required before the spent fuel pit water would begin to boil. Justify that service water cooling could be restored within this period or provide analyses of the consequences of spent fuel pit boiling including offsite dose consequences and the long term heat transfer degradation from boron precipitation within the fuel element cooling channels.

RESPONSE:

The time required to initiate boiling in the spent fuel pit, if spent fuel cooling is lost, depends on the length of time that spent fuel has been in the pool. In the present case, there are two time periods of concern when one service water is not permitted to be out of service with the current technical specifications. During the first period, when the water level is at the nozzle centerline, the reactor has not yet been defueled. Therefore, the fuel in the spent fuel pit has been there for at least one full cycle. The heat up rate of the fuel pool for this case, in the absence of the normal spent fuel cooling, is approximately 2°F per day. Based on the maximum expected initial fuel pool temperature of 80°F, it would take more than two months to start boiling in the fuel pools.

The second period of concern in when the water level is at the reactor vessel flange, the reactor has been refueled and a 1/3 core discharge is in the spent fuel pit (> 21 days after shutdown). The heat up rate of the fuel pools for this case is approximately 3.5°F per hour. Based on an initial fuel pool temperature of 80°F, it would take more than 1-1/2 days to start boiling in the pool.

As discussed in the Safety Evaluation, an existing cross-connect between the spent fuel pool cooling systems of Salem Units 1 and 2 can be used to cool the spent fuel pool, if service water is lost to one of the units, for an indefinite period of time. In addition, as discussed in FSAR section 9.1, there are several sources of make-up water for the spent fuel pools if boiling ever did occur.