PSE&G

Public Service Electric and Gas Company 80 Park Place Newark, N.J. 07101 Phone 201/430-7000

July 1, 1980

Director of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. A. Schwencer, Acting Chief
Licensing Branch No. 3
Division of Licensing

Gentlemen:

AUXILIARY FEEDWATER SYSTEM
NO. 1 AND 2 UNITS
SALEM NUCLEAR GENERATING STATION
DOCKET NOS. 50-272 AND 50-311

Public Service Electric and Gas hereby submits additional information in response to your concerns regarding the Auxiliary Feedwater System. This information was requested at a meeting with members of your staff on June 18, 1980.

The Auxiliary Feedwater Systems for both Salem 1 and 2 are identical in design, including equipment and automatic initiation logic, as indicated in our letter dated June 27, 1980.

Should you have any questions, do not hesitate to contact us.

Very truly yours,

R. L. Mittl
General Manager -
Licensing and Environment
Engineering and Construction

Enclosure
CC: Mr. S. A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

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Additional Information
Auxiliary Feedwater System
No. 1 and 2 Units
Salem Nuclear Generating Station

1. Salem 2 FSAR Amendment 43 addresses branch technical position RSB 5-1, Cold Shutdown Requirements. We note that the long-term auxiliary feedwater supply source (service water system) may contain salt in solution. Justify the salt content of this water (for the time period required to satisfy RSB 5-1) against the following considerations:

   a. degradation of steam generator shell side heat transfer due to salt plating or tubes,

   b. clogging of steam generator secondary side flow paths by clumps of accumulated salt,

   c. corrosion effects of the salt.

   Also describe provisions (procedures, special equipment, etc.) which would be used to attenuate the effects of the salt (a,b,c, above).

Response

1. ASSUMPTIONS

   Based upon the information contained in the response to Branch Technical Position RSB 5-1 (FSAR Amendment 43, Question 9.61), the following scenario has been evaluated:

   A. There has been a loss of offsite power. Coincident with the loss of offsite power the turbine and reactor systems are tripped, and the plant begins decay heat removal using emergency supplies.

B. Auxiliary Feedwater is initially provided from the 220,000 gallon Seismic Category 1 Auxiliary Feedwater Storage Tank (AFST). This tank contains condensate grade make-up water. When the supply of water in the AFST has been depleted, Auxiliary Feedwater pump suction is transferred from the AFST to the Seismic Category 1 Service Water System.

This system is supplied directly from the tidal section of the Delaware River. The salinity of the river at this point will vary with the overall hydrological environment of the area. Plant records show that normally the Total Dissolved Solids (TDS) content of the river is in the range 9,000 - 13,000 ppm at Salem. In the absence of a reliable value for the upper limit of TDS previously experienced in this part of the river, and to provide maximum conservatism in evaluating the effects of the postulated incident, it is assumed that at the time of the incident the river will have a TDS value of 35,000 ppm as sodium chloride. This value corresponds to the salt concentration accepted as being representative of "international sea water", and will have a higher TDS content than the most concentrated brackish water postulated for the Delaware River in the area of the Salem Station.

C. Decay heat removal is achieved using three of the four steam generators. Cooling is achieved by dumping steam to atmosphere via Power Operated Relief Valves (PORV).

D. The three available steam generators are maintained at Normal Working Level (NWL) throughout the major part of the scenario. Level is maintained using only Seismic Category 1 water supplies.

E. Following the loss of offsite power the unit is held at the Hot Standby condition. After 43 hours at the no load condition, Reactor Coolant System cooldown is initiated. After 5 hours of cooldown the plant is at 350°F and the Residual Heat Removal System is valved in-line. At this point, decay heat removal via the steam generators ceases, and the RHR system is used to continue the heat removal process.

2. <u>PLANT PARAMETERS</u>

Number of Loops	=	4
Power Rating (MWTh)	=	3,423
Temp., No Load (°F)	=	547
Temp., RHR On (°F)	=	350
Available Steam Generators	=	3
SG Volume at NWL (FT^3)	=	3,425
Volume AFST (gallons, demin.)	=	220,000
TDS of Service Water (ppm)	=	35,000

3. TOTAL WATER REQUIREMENT

HOURS AT NO LOAD	NO LOAD WATER (GALLONS)	5 HOUR COOLDOWN (GALLONS)	TOTAL WATER (GALLONS)
2	82,075	138,087	220,162
12	212,491	119,103	331,594
24	332,756	111,287	444,043
168	1,280,973	93,030	1,374,003

4. SALT WATER CONCENTRATION

Figure 1 shows that 0.602×10^6 gallons of water will be required to maintain the plant at the no load condition for 43 hours plus at least a 5 hour cooldown to begin RHR operation. 0.22×10^6 gallons of this water will be supplied from the AFST, and should not contribute significantly to the salt burden of the Steam Generators. Thus the salt water requirement to meet RHR operation within 48 hours is

$$(0.602 - 0.22)\ 10^6 \text{ gallons}$$
$$= 0.382 \times 10^6 \text{ gallons}$$

Based upon the assumption that this salt water will contain 35,000 ppm TDS as sodium chloride, then the weight of salt introduced into the three available Steam Generators will be 50.75 tonnes.

At 350°F, three Steam Generators will contain 258.67 tonnes of water.

Thus, the concentration of salt in the SG water will be

$$\frac{50.75}{258.67} = \underline{0.20 \text{ t/t}}$$

From standard solubility tables, and using the same notation as above, the solubility of sodium chloride at room temperature is 0.36 t/t, increasing to 0.58 t/t at the no load temperature. (References 1 and 2 and Figure 2.)

5. RESPONSE TO NRC CONSIDERATIONS

A. Degradation of heat transfer due to salt plate-out.

Salt water will be first admitted to the steam generators twelve hours after the loss of offsite power incident. By this time the thermal flux at the steam generator tube wall will be relatively low, and there will be no tendency to hide out normally soluble ionic species. It is anticipated that steam release from the PORVs will tend to be increasingly intermittent as decay heat decreases. Thus, the net effect will be similar to bulk boiling, rather than the superheat driven mode of boiling normally associated with steam generation for power production purposes.

In addition, the data provided in 4 above, shows that the concentration of salt in the bulk water is well below the solubility limit of sodium chloride throughout the range of temperatures expected during operation at the hot standby condition (0.20 vs. 0.44 at 350°F, and 0.58 at 547°F).

There is, therefore, no evidence to suggest that there will be any decrease in heat transfer capability due to the deposition of salt on the steam generator tube surfaces during the postulated period of hot standby operation.

B. Clogging of SG Flow Paths by Salt Accumulation

The concentration factor and sodium chloride solubility data discussed in previous parts of this response indicate that the internals of the steam generator totally immersed in the bulk liquid phase should not be subject to salt precipitation and clogging. Similar consideration of the steel surfaces at the water/steam interface suggests that although some salt deposition is likely, the deposition will not be at a fixed boundary because of level changes caused by intermittent steam release via the PORVs. In addition, any re-wetting will be carried out with a liquid which has considerable salt solubilization capability. Based upon these considerations, it is unlikely that salt bridging wil occur at the water/steam interface. Zones above this interface are not expected to become clogged with salt since evaporation is a process which purifies the solvent phase. Thus any salt carried upward by liquid phase entainment should be washed back down into the bulk liquid by relatively pure water (condensed steam).

C. Corrosion Effects of Salt

The corrosion of carbon steels in hot concentrated sodium chloride solutions is controlled by the metal alloying constituents, solution flow rate, oxygen concentration, temperature, and solution pH. Literature values for carbon steel corrosion rates at steam generator operating temperatures vary from 0.012 inches/week (Reference 3) to 0.21 inches/week (Reference 4). Tests carried out on ASTM SA 285 grade C steel, used by Westinghouse as Tube Support Plate material, show that a corrosion rate of approximately 0.015 inches/week can be expected when this material is exposed to concentrated sea water at 540°F (Reference 5). The ligament between the tube and flow hole in the Salem steam generators is 0.083 inches wide. Based upon these data, tube support plate corrosion must be expected. Other carbon steels present in the steam generator will also be corroded by the hot concentrated sea water. The condition of these carbon steel internals under hot standby conditions does not compromise the integrity of the primary to secondary steam generator boundary, since their function is to act as anti-vibration support or steam separation equipment during normal power operation.

Data generated in single tube model boiler tests performed by Westinghouse indicate than Inconel 600 will pit at about 0.005 inches/week in hot concentrated sea water environments. Corroborative information of this behavior is contained in Reference 6. Nominal tube wall thickness in the Salem steam generators is 0.050 inches. Thus only shallow tube wall penetration should be anticipated as a result of 36 hours of operation at hot standby with concentrated sea water in the steam generators. No perforation of tube wall is anticipated during the period of this scenario.

6. CONCLUSION

The above discussion evidences that a loss of offsite power incident, followed by a 48 hour period when only Seismic Category 1 water supply systems are available, will have no impact upon safety concerns at Salem. Both primary circuit integrity and heat removal capability will be maintained throughout the postulated period of operation at hot standby, until the plant can begin heat removal via the RhR system. From the safety standpoint, no provisions are necessary to attentuate the effects of salt wter ingress under the conditions postulated for this scenario.

REFERENCES

1. CRC Handbook of Chemistry and Physics
2. Communication from W. T. Lindsay, Jr., Westinghouse Chemical Service Consultant
3. Potter and Tease - Corrosion Science Vol. 12, No. 4, April 1972.
4. W. M. Huijbregts - VGB Speiswassertagung 1970.
5. M. J. Wootten - Westinghouse Class 2 Research Report 77-1B6-DENTS-R1.
6. D. J. Roberts et.al. - AEC Research and Development Report GA-9299.

HOT STANDBY WATER REQUIREMENTS

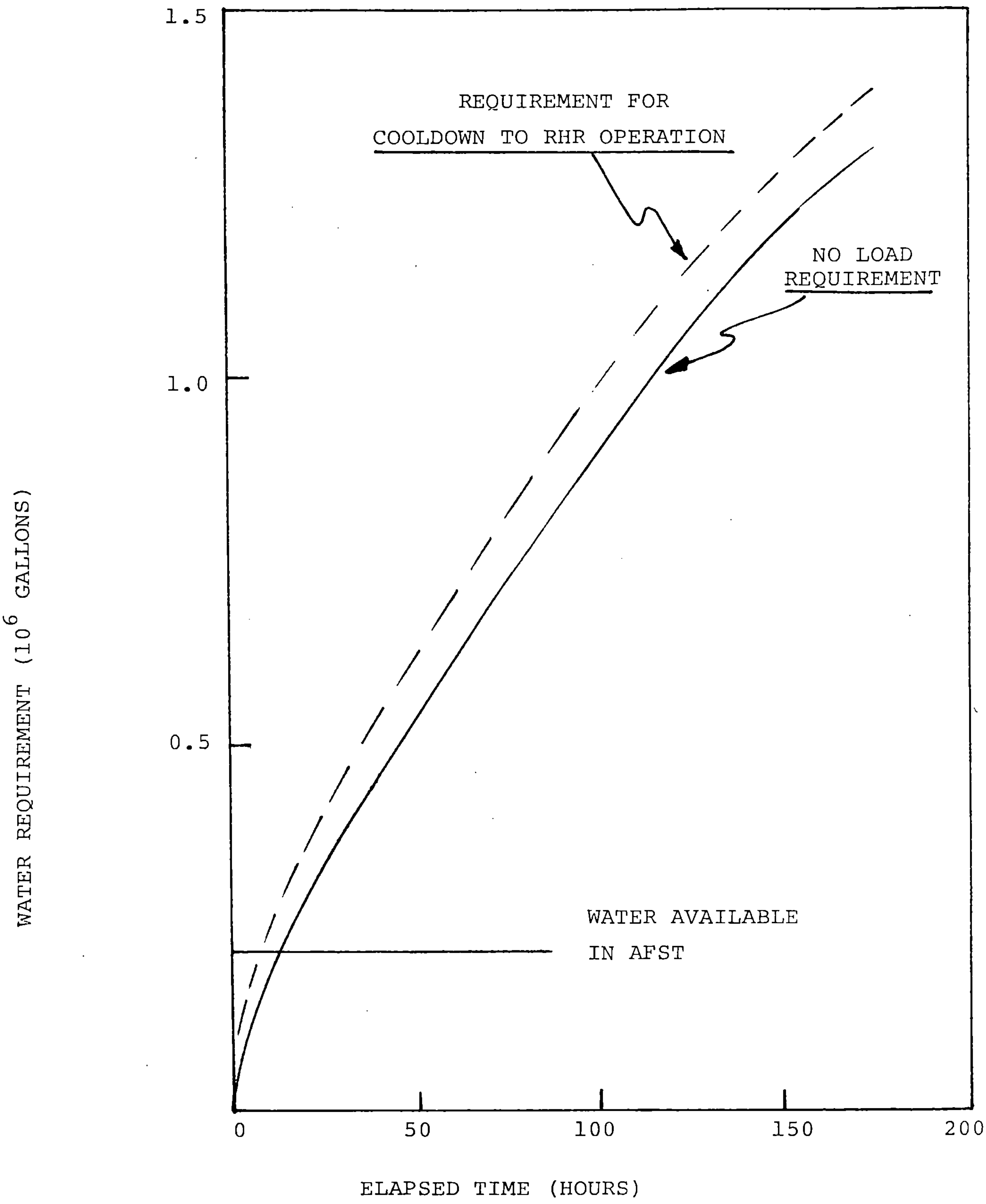

FIGURE 1

SODIUM CHLORIDE SOLUBILITY

(BASED ON DATA SUPPLIED BY W. T. LINDSAY, JR.)

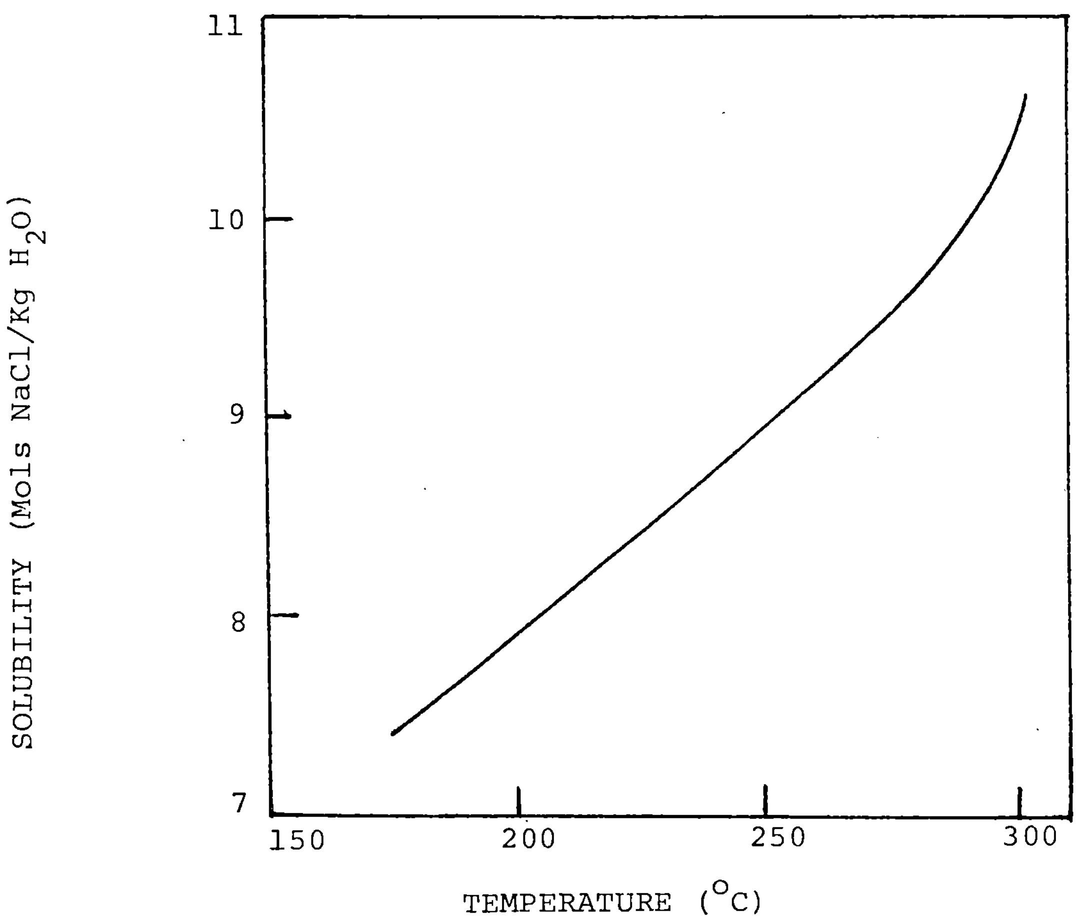

FIGURE 2

2. SERVICE WATER SPOOL PIECE CONNECTION

The Station Operating Procedures will be revised to provide for installation of the spool-piece connection between the SWS and the AFW System upon receipt of a "Tornado Warning" in the vicinity of the Salem Station.

3. AFW ALTERNATE SUCTION

The Alternate AFW suction line from Demineralized Water Storage Tank passes through the switchgear room. This line is normally maintained in an isolated and dry condition. Modifications will be made to the normally-open drain valve on this isolated section of pipe such that it will be piped directly to a floor drain to preclude any potential for flooding the switchgear room.

Additionally, the Station Alarm List provides for changeover to an alternate suction path for the AFW pumps upon initiation of a low level alarm in the AFST and an inability to correct the low level condition.

4. SINGLE SUCTION PATH FROM AFST

In order to provide assurance that inadvertent failure (or closure) of the suction valve from the AFST (AF1) will not result in a degraded condition of the AFW System, PSE&G will remove the intervals from AF1. This modification was discussed with and approved by the NRC staff and will provide conformance with recommendation GL-2 in NUREG-0611.