OFFICIAL COPY

JUL 1 8 1990

MEMORANDUM FOR: Lawrence P. King, Resident Inspector
North Anna Power Station
Division of Reactor Projects

FROM: Luis A. Reyes, Director
Division of Reactor Projects

SUBJECT: SAFETY CONCERNS AT SURRY

On November 8, 1989, I issued a memorandum (Enclosure 1) to the Region II technical divisions requesting their assistance in resolving four safety concerns, which you brought to your management's attention on September 19, 1989. In parallel with this review, on February 27, 1990, NRR was requested to review two of these concerns (Enclosure 2). After extensive inspection and review by both the Regional and Headquarters staffs, these issues have been resolved to my satisfaction. In addition, the Surry Senior Resident Inspector has reviewed the responses to ensure that any generic hardware information is in fact applicable to Surry. Enclosed are copies of the documents with the detailed reviews and inspection activities conducted to resolve your concerns. In order to keep you informed of the results of the staff's efforts, the following is a summary of the results of these reviews and inspections:

1. We concur with your concern in the matter of flow testing the service water to the recirculation spray heat exchangers (RSHXs). The Region II staff review of this issue (Enclosure 3, Inspection Report 280,281/90-04) has determined that the RSHXs definitely have not been tested subsequent to unit operation. Our review also revealed that, unlike the pump-and-throttle-valve service water system at North Anna, the gravity-fed system at Surry does lend itself to be evaluated for flow requirements through the use of engineering calculations. These calculations do support the position that design basis flow would be achieved even under minimum intake canal level conditions. Nevertheless, in that the RSHXs at Surry have recently been replaced, and that we know that fouling can take place due to the brackish intake water, we informed the licensee of the necessity to perform full-flow service water testing of the RSHXs. The licensee has agreed to perform flow testing commencing with the October 1990 Unit 1 outage. The extent of this testing has not been finalized but will be conducted to the satisfaction of the NRC.

2. Our review has determined that the concern involving post-accident containment iodine removal does not pose a significant safety problem that warrants any further NRC action. Both this issue and Concern 4 were sent to NRR (Enclosure 2) on February 27, 1990, for review. NRR responded with their evaluation (Enclosure 4) on May 17, 1990.

   The NRR evaluation stated that the iodine present in the post-accident containment atmosphere exists in three different forms: elemental, particulate and organic. The containment spray system will effectively remove elemental and particulate forms of iodine. Organic iodide and iodate will be removed only after they decompose in a radiation field. Since it is not known exactly how those organic compounds decompose in a

See Rpts

IE01
1/1

radiation field, no credit is generally taken for removal of the organic form of iodine by the containment sprays. Removal of particulate iodine is controlled strictly by hydraulic mechanisms; chemistry of the spray solution is immaterial. It is only in removal of elemental iodine that the chemistry of sprays plays an important role. The effect of pH is addressed in Section 6.5.2 (Rev. 2) of the Standard Review Plan (SRP). Based on several experimental studies, the revised SRP states that a fresh spray solution, having no dissolved iodine, can effectively scrub elemental iodine at a pH as low as 5. However, higher pH values, usually above 7, are required with spray solutions containing dissolved iodine. This is due to the fact that in acidic solutions absorbed radiation generates hydrogen peroxide in sufficient amount to react with iodide and iodate ions and raise the possibility of elemental iodine re-evolution.

During the injection phase, the solutions coming from the refueling water storage tank (RWST) and from the chemical addition tank (CAT) do not contain dissolved iodine. Although pH of the RWST water containing 2200 ppm of boron (as boric acid) is about 4.9, addition of as little as 1 ppm of 17 percent NaOH solution raises the pH to 5. A spray solution containing only a minimal amount of sodium hydroxide will, therefore, be effective in removal of elemental iodine during the injection phase, and variations in the rate of flow from the CAT will have no significant effect on iodine removal. However, during the recirculation phase, a pH value of 7 or higher should be maintained for the reasons mentioned before and to keep the corrosion down, as required by Section 6.1.1 of the SRP. Since the injection phase lasts a relatively short time, an acidic solution would not produce significant corrosion damage; however, the recirculation phase may last much longer and corrosion control becomes an important issue.

Based on the information provided in Section 6.5.2 (Rev. 2) of the Standard Review Plan, NRR concluded that variation in the rate of feed of sodium hydroxide from the CAT would not affect removal of iodine from the containment atmosphere during the injection and recirculation phases. In the first case, relatively low pH values (above 5) can be tolerated; and in the second, a fully discharged CAT will keep the containment sump water at a pH higher than 7.

3. The general problem that you identified with the component cooling water (CCW) system had been previously known and followed by the Region II staff, specifically, as addressed in Inspection Reports 280,281/89-18 and 280,281/89-19 (Enclosures 5 and 6, respectively). Inspection Report 280,281/90-04 (Enclosure 3) also contains additional information on this issue. We concur in your position with respect to the high reading on the CCW radiation monitor; but your concern over the isolation and pressure capability of the system is not a problem as it is based on an incorrect CCW system description. Although the system still needs some long-term enhancement, the system is presently designed and capable of isolating the high pressure portion of the CCW system, which interfaces with the RCP thermal barrier, from the remaining low pressure portion of the CCW system.

With respect to the radiation monitors, the licensee originally believed the primary cause for the continuous high readings was a small reactor coolant pump (RCP) thermal barrier to CCW leak. They suspect the thermal barrier leak to be from Unit 1, but have not determined which RCP is the problem. In parallel with testing to determine which RCP thermal barrier is leaking, the licensee plans to install ion exchangers by the end of 1990, to filter the CCW flow and reduce the radiation readings. Repairs to the thermal barrier, if identified to be a problem area, will be evaluated for inclusion into an appropriate outage. Also, in late June 1990, the licensee identified another reactor coolant system leak from the Unit 2 non-regenerative heat exchanger to the CCW system. Plans are being made to address the problem.

As far as the inability to auto-isolate the CCW system due to an RCP thermal barrier heat exchanger rupture, a modification to the CCW system was implemented in 1986, installing a check valve to the supply side and a trip valve on the return side of the CCW system from the thermal barrier heat exchangers for each RCP. This modification also installed a relief valve downstream of each unit's thermal barrier in the CCW discharge header for the purpose of providing overpressure protection of the low pressure portion of the CCW system. As the trip valve shuts on high CCW flow, CCW isolation does not depend on any alarm signal from the radiation monitors. In addition, in 1989, the licensee added an additional check valve to each heat exchanger line. Both check valves in each supply line and the CCW relief valve on the discharge header were upgraded to safety-related in 1989. A meeting in NRR was conducted on June 28, 1990 to discuss long-term resolution of this problem. Enclosure 7 is the summary of this meeting. The licensee plans to perform an evaluation of the CCW discharge lines inside containment, and verify they will withstand primary system pressure and to add inside and outside safety-related containment isolation valves with their associated flow instrumentation.

With respect to the potential of the CCW system to rupture on a primary to CCW leak, due to the low pressure rating of a portion of the CCW system, the CCW piping inside the check valve and trip valve isolation boundary is Schedule 160, ASTM GR106, rated to at least 3466 psig. Thus, any RCP thermal barrier rupture which could pressurize the CCW system to reactor coolant system pressure would be isolated and contained within the high pressure portion of the CCW system.

4. Similar to Concern 2, NRR has reviewed the issue involving isolation of steam generator blowdown after a primary-secondary leak (Enclosure 4) and has concluded that the lack of automatic isolation is not a significant concern. Specifically, steam generator blowdown at Surry is continuously monitored for radioactivity, indicative of a primary-to-secondary system leak. The sensitivity of the radiation detectors ensures that abnormal plant conditions will be detected before they cause a hazard to the operators or to the public. Each of the two available channels per unit has a readout and an audible and visual alarm in the control room for radiation levels in excess of preset values. The steam generator blowdown lines contain isolation valves that auto-isolate on a containment

isolation signal and initiation of auxiliary feedwater and also are able to be remote-operated from the control room. A 3-inch line downstream of each steam generator blowdown containment isolation valve carries the blowdown effluent from the auxiliary building into the turbine building. Steam generator blowdown is continuously treated by the blowdown treatment system to control ionic and suspended impurities prior to discharge or return to the condenser hotwell. After cooling, the normal water route is through a prefilter, two mixed-bed demineralizers in series, and a postfilter. A third demineralizer bed, a cation unit, is available to be valved into service for cesium removal during periods of high primary-to-secondary leakage. Monitors and alarms are provided to indicate system malfunctions. The flow rate through the train is manually remote controlled from the control room over the range 30 to 210 gpm. The decontamination factors assigned to the evaluation of this system are as follows: cesium and rubidium, 100; iodine and others, 1000. Radiation monitors are located in the circulating water discharge tunnel beyond the last point of possible radiation material addition. The waste disposal system and radiation monitoring system are designed to satisfy the appropriate GDC and to limit the discharge of radioactive materials to within the limits of 10 CFR Part 20 and the criteria of 10 CFR Part 50, Appendix I and 10 CFR Part 100.

The staff considers that the lack of automatic isolation from high blowdown radiation monitor readings is not a significant safety concern for the following reasons:

a. Adequate treatment of blowdown is normally provided on a continuous basis prior to discharge. This also prevents the discharge of significant quantities of radioactive materials prior to isolation of the blowdown lines during a period of high primary-to-secondary leakage.

b. Detection and alarm provide sufficient time for operator action to valve in the third demineralizer and to isolate the blowdown lines from the control room, if necessary, to prevent the release of significant quantities of radioactive materials during a period of high primary-to-secondary leakage; and

c. Regarding the concern that a significant uncontrolled release could occur prior to manual isolation if the monitor became insensitive, automatic isolation would not alleviate this concern since this could affect not only the proper indication and receipt of alarm signal, but also the receipt of an isolation signal.

If you have any questions regarding the resolution of these concerns, feel free to contact either me, your Branch Chief, or your Section Chief. I will keep you informed if any substantive changes occur to the resolution of these issues.

(Original signed by LAReyes)

Luis A. Reyes

Enclosures: (See page 5)

Enclosures:
1. Reyes to Stohr and Gibson
   Memo Dtd November 8, 1989
2. Region II to NRR Memo Dtd
   February 27, 1990
3. Inspection Report 280,281/90-04
4. NRR to Region II Memo Dtd
   May 17, 1990
5. Inspection Report 280,281/89-18
6. Inspection Report 280,281/89-19
7. NRR Meeting Summary on Surry
   Meeting, Dtd July 9, 1990

cc w/encls:
A. Gibson
P. Stohr

RII:DRP	RII:DRP	RII:DRS	RII:DRSS
PFredrickson	MSinkule	MERSCHOFF	STOHR
07/13/90	07/13/90	7/5/90	7/5/90