UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ON THE SUBJECT OF HIGH PRESSURE INJECTION NOZZLES AND ASSOCIATED THERMAL SLEEVES

SACRAMENTO MUNICIPAL UTILITY DISTRICT

RANCHO SECO NUCLEAR GENERATING STATION

DOCKET NO. 50-312

INTRODUCTION

In April 1982, the licensee shutdown the Rancho Seco facility in order to repair cracks found in the A high pressure injection (HPI) nozzle safe-end. The repairs included installing modified thermal sleeves in the A and B nozzles.

The staff Safety Evaluation Report evaluates:

1) the licensee's corrective actions regarding HPI nozzle cracking and thermal sleeve replacement, and

2) the acceptability of interim reactor operation, until the next refueling outage, with the missing thermal sleeve from nozzle A assumed to be in the reactor vessel.

DISCUSSION

As a result of make-up nozzle cracking experienced at the Crystal River 3 and Oconee plants, the Rancho Seco facility was shutdown in April 1982 to permit ultrasonic and radiographic testing of the four high pressure injection (HPI) nozzles. Nozzles C and D were found to be free of any cracking, and the C and D thermal sleeves were correctly positioned. No cracking was found in nozzle B, but the B thermal sleeve was loose. Only the normal make-up nozzle A contained cracks in the nozzle safe-end, and its thermal sleeve was found to be missing and assumed to be in the reactor vessel.

The licensee replaced the A and B nozzle safe-ends, and the A and B thermal sleeves. The new thermal sleeves incorporated a new design to better secure the thermal sleeve inside the nozzle.

EVALUATION

1) Licensee's Corrective Actions

As discussed above, the licensee has replaced the A and B HPI nozzle safe ends. Additionally, the A and B thermal sleeves were replaced with new, re-designed sleeves that are better retained within the nozzles. Enclosure 1 shows the old thermal sleeve design, while the new design is shown in Enclosure 2. The new design assures better thermal sleeve retention, since the downstream end is contact expanded to the nozzle, and the upstream end is hard-roll expanded. Additionally, the new design incorporates a "lip" on the upstream end, which is of a greater diameter than the sleeve, so that the sleeve will not be able to slide through the nozzle and enter the reactor coolant system.

The C and D thermal sleeves were not replaced since radiographic examination revealed that these sleeves were not loose.

To date the actual cracking mechanism has not been positively identified. A B&W Owner's Group Safe-End Task Force has been established, and met with the staff on May 7, 1982. The Task Force has postulated that the nozzle cracking was caused by loose thermal sleeves. By the end of 1982, the Task Force is expected to issue its report identifying the cracking mechanism, and recommending any required design changes, plant operations changes or augumented inservice inspections.

However, the licensee has independently committed to conduct ultrasonic and radiographic examinations of the four HPI nozzles during the next refueling outage, which starts in January 1983. These examinations will assure that the thermal sleeves are in place and that no new nozzle cracking has occurred.

Accordingly, we find these actions acceptable because the operating information has indicated that no cracking occurred for those nozzles with intact thermal sleeves.

2) Interim Reactor Operation With a Thermal Sleeve in the Reactor Vessel

The Rancho Seco facility will be shutdown in January 1983 for refueling and a 10-year inservice inspection (ISI). The reactor internals will be removed to permit ISI of the reactor vessel, and the thermal sleeve will be removed from the reactor vessel at that time. Reactor operation during the interim (August 1982 to January 1983), with the thermal sleeve in the reactor vessel, has been evaluated by the staff. Two concerns about the thermal sleeve in the reactor vessel were raised by the staff; 1) flow blockage effect on core thermal hydraulics, and 2) damage to reactor internals.

In its letter dated July 21, 1982, Sacramento Municipal Utility District (SMUD) presented its assessment of the potential flow blockage effect on core thermal hydraulics due to the HPI nozzle loose thermal sleeve. The assessment was performed considering two situations for the loose thermal sleeve, i.e., the sleeve either remains intact or may be broken into small pieces.

For the intact sleeve case, it would be too large to pass through the lower end fitting grillage and would, therefore, lodge below and at an angle to the lower grid plate. This condition would result in slight inlet flow maldistribution, which may be accounted for in the design analysis where a 5% inlet flow reduction was assumed for the limiting assembly. In addition, thermal hydraulic analyses using an open lattice crossflow code have shown that inlet flow maldistribution has a small effect on DNBR downstream because of flow redistribution.

For the damaged thermal sleeve case, the large pieces which could not pass through the lower end fitting grillage would result in inlet flow blockage as in the intact sleeve case. Pieces which are small enough to pass through the lower end fitting grillage and lower end spacer grid to gain access to the active fuel region of the core are also too small to produce a significant flow blockage if they lodge in an intermediate spacer grid. However, if many small pieces lodge in the subchannel at the same intermediate grid, a local flow blockage may be formed. Experimental data have shown that the stagnant zone behind the flow blockage essentially disappears a few inches downstream. It is likely that the blockage will occur in a spacer grid near the core inlet where the thermal hydraulic conditions are such that DNB would not occur. Therefore, the staff concludes that DNB due to flow blockage produced by the loose thermal sleeve is highly unlikely and that continued operation of Rancho Seco until the next refueling outage is acceptable with regard to this concern.

Regarding possible damage to reactor internals by the presence of the thermal sleeve in the reactor vessel, the staff has determined that the reactor internals functions will not be impaired by the loose thermal sleeve based on the energy limitations of the loose thermal sleeve; and continued reactor operation until the next refueling outage is acceptable.

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The following NRC personnel have contributed to this Safety Evaluation: M. Padovan, S. Hou, Y. Hsii.