Enclosure 1

Evaluation of Open Issues on Three-loop Operation in Millstone 3 (TAC 61720)

In Supplement 4 to the Millstone 3 SER (NUREG 1031) the staff identified certain shortcomings in the analysis of N-1 loop operation for Millstone Unit 3 and requested that the licensee provide additional information. The licensee has provided the requested analyses and the Reactor Systems Branch has prepared the following evaluation. The Electrical/Instrumentation and Control Systems Branch is reviewing the implementation of the required protection system adjustments and will provide a separate evaluation.

1. Transition from 4-loop to 3-loop Operation at 25% power

The staff had expressed concern that the existing safety analyses might not bound scenarios involving transients while in the transition phase from four loop to three loop operation. By letter dated January 17, 1986 the licensee provided a justification for the proposed transition at 25 percent power.

The switchover procedure is established to assure that the safety analyses remain bounding during the transition. While operating with four loops, core power is reduced to less than or equal to 25% rated (at this level no reactor trip is required on loss of flow in one loop). The appropriate reactor trip setpoints are reset to three loop operation values. Finally, the loop is isolated and the coolant pump is tripped.

In summary, the initial and final conditions are bounded by safety analyses presented in the FSAR. Since the setpoints are reduced to their three-loop values before the switchover is made, operation is also bounded during the transition. The licensee has examined each of the Chapter 15 events and confirmed that the analyses remain valid and bound the transition. We conclude that the switchover from 4-loop to 3-loop operation at 25% power is acceptable.

2. Reactor Coolant Pump Rotor Seizure and Shaft Break

The staff required that the licensee either justify the absence of fuel failure following DNB for this event or evaluate the consequences of fuel failures. By letter dated December 23, 1985 the licensee provided such analyses evaluating the consequences of fuel failures and the issue was closed in Supplement 5 to the Millstone 3 SER (NUREG 1031).

3. Three Loop-Main Steam Line Break Flow Modelling

The staff's SER of October 25, 1985 expressed concern that the licensee had not clearly stated that conservative assumptions were used for the main steam line rupture analysis. In particular, the use of homogeneous complete coolant mixing in the lower plenum was questioned. By letter dated May 29, 1986 the licensee addressed the staff's concern. The licensee pointed out that homogeneous mixing was not assumed in the analysis. The LOFTRAN code which was used in the analysis permits the degree of mixing to be specified by use of mixing coefficients. These coefficients were chosen so that the resultant mixing was conservative compared to test results from the Indian Point One-Seventh Scale Vessel Model Test. Other conservative assumptions and methodology described in WCAP 9226 were used.

We conclude that mixing was appropriately treated in the analysis and that the steamline break analysis is acceptable.

4. Justification that AOOs and Postulated Accidents for Modes 1 and 2 are Bounding

The staff requested the licensee to review each Anticipated Operational Occurrence (AOO) and Postulated Accident (PA) to assure that all equipment and systems specifically required by Millstone 3 Technical Specifications to mitigate those events during Modes 1 and 2 would be available and provide the same level of protection for Modes 3 and 4. Otherwise the licensee was requested to provide additional information to show that the proposed N-1 loop Technical Specifications for Modes 3 and 4 would not allow the plant to operate in a unanalyzed condition and that they are consistent with the safety analyses.

By letter dated July 1, 1986 the licensee provided information addressing this request. Each of the Chapter 15 events described in the FSAR was examined to verify that protection equivalent to that in Modes 1 and 2 is provided in Modes 3 and 4. In this evaluation the licensee considered the effect of the reduction in protective function in the subcritical modes. This reduction is offset by the reduction in severity of transients and by the introduction of other protective functions (e.g., the source range reactor trip). In addition the removal from service of components and systems also removes the initiating equipment for some of the Chapter 15 events (e.g., the feedwater heaters are turned off below mode 2). The licensee concludes, and we concur, that the level of protection is not reduced in the subcritical modes from that which is present in Modes 1 and 2.

Coincident with the licensee's submittal of July 2, 1986 Westinghouse informed the staff of a potential for degradation of protection in the event of a LOCA in Mode 4 when the reactor is being cooled by the RHR system. As part of the response to such an event the operator is required to assure that the RHR pumps are not damaged should the hot leg uncover during the blowdown phase. If the operator should fail to do so the RHR pumps could be damaged and would not be available for the recirculation phase of the event. The Westinghouse Owners Group, of which the licensee is a member, is currently developing a resolution to the problem which is expected to the completed in April 1987. The licensee has committed to conform to the actions agreed to by NRC and the Owners Group. In addition an interim resolution involving changes in administrative requirements and procedure is being developed and will be submitted on a shorter schedule. The staff considers the occurrence of a LOCA in Mode 4 to be unlikely since the system pressure is low (less than 450 psig). In view of this fact we conclude that Millstone Unit 3 may be safely operated while awaiting the implementation of the interim solution and final resolution of this issue.

5. Steam Generator Tube Rupture (SGTR) for 3-loop Operation

As stated in Supplement 5 of this SER, by letter dated October 25, 1985, the licensee proposed to use the results of the Westinghouse Owners Group (WOG) generic program to resolve the Millstone 3 SGTR licensing issue for both four-loop and three-loop operation. In Supplement 5 to NUREG 1031 the staff concluded that the proposed methodology to evaluate the N-1 loop SGTR event is acceptable pending the results of the review of WCAP-10698, although the overfill margin for 3 loop operation may be slightly less than for 4 loop operations.

Staff review of the SGTR generic reports WCAP-10698 and WCAP-11002 is continuing. These reports are pertinent to both four-loop and three-loop plants. Therefore the staff does not require additional submittals for N-1 loop operation. The staff, however, will require that each facility submit plant specific information on the following:

1. Steamline static load analysis in the event of overfill,

2. Site specific offsite radiological consequences.

3. Justification that systems and components used to mitigate the consequences of the accident are safety related.

This information will be required at least six months prior to the first refueling outage in order to permit preparation of an SER prior to startup of the second cycle.

The staff concludes that there is reasonable assurance that Millstone Unit 3 can operate for the first fuel cycle before this issue is resolved, because: (1) the probability of a design basis SGTR during the first fuel cycle is low, (2) in the event of the design basis SGTR, the offsite consequences can be expected to be within 10 CFR Part 100 guidelines, particularly if no liquid release occurs, (3) the core melt probability from a single tube rupture is very low (Reference 1). While 3 loop operation may result in less margin to overfill than 4 loop operation in the event of SGTR, the iodine fission product inventory is lower due to operation at reduced power (see supplement #4 of this SER)

In summary, we conclude that the five open issues identified in Supplement 4 to NUREG-1031 are closed, pending compliance with the conditions imposed by the review of the generic SGTR Topical reports and the plant specific information listed above.

References

(1) NUREG 0844 "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4 & A-5 Regarding SG Tube Integrity" Draft Report, April 1985.