UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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Docket No. 50-338

MEMORANDUM FOR: John Olshinski, Director
Division of Engineering and Technical Programs
Region II

FROM: Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

SUBJECT: DEGRADATION OF THERMAL SLEEVES-NORTH ANNA POWER STATION UNIT NO. 1 (NA-1)

As discussed in the enclosed Safety Evaluation, we have no objections to NA-1 resuming operations for Fuel Cycle No. 4 with thermal sleeves as reported in the Virginia Electric and Power Company letter dated October 12, 1982. Should our generic review of this matter result in the need for further action on the part of the licensee, we will contact them directly.

By this memorandum, NRR actions requested in the Region II memorandum (J. Olshinski to D. Eisenhut) dated July 9, 1982 regarding NA-1 is complete. You have already received my memorandum dated August 27, 1982 with the enclosed safety evaluation for NA-2 regarding thermal sleeves. Therefore, Task Interface Agreement No. 82-32, Rev. 1, is hereby complete for both NA-1&2.

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

DEGRADATION OF THERMAL SLEEVES

NORTH ANNA POWER STATION, UNIT NO. 1

DOCKET NO. 50-338

1.0 Introduction

During an inspection at an operating Westinghouse plant (Trojan), an underwater television camera revealed a loose metal piece under the reactor internal core plate. Subsequent investigation resulted in the discovery of additional loose parts in the reactor vessel and an eventual conclusion that the sources of the parts were the thermal sleeves from the 10-inch RHR/SIS line nozzles. A similar failure was discovered at another Westinghouse plant where one 10-inch nozzle thermal sleeve of the same design was found in the reactor vessel lower plenum. A third incident of this nature was reported at North Anna, Unit 2 (NA-2), where a radiographic examination by the Virginia Electric Power Company (the licensee), revealed that four thermal sleeves having the suspect design had cracked welds.

For NA-1 also, radiographic examinations during the Summer of 1982 revealed the existence of the thermal sleeve problem. One three (3) inch charging line sleeve (B loop) had a cracked weld and has now been removed during the present extensive NA-1 outage. Also, the six (6) inch safety injection sleeve (4 loop) had broken away. This sleeve has been located in the bottom of the reactor vessel and has now been removed during the present outage. All other six sleeves of similar design have been determined to be intact and the licensee intends to continue operation at NA-1 with the six sleeves in place. To support continued operation, a safety analysis, performed by Westinghouse, was submitted by the licensee on October 12, 1982. The NRC staff evaluation of the potential effects of loose thermal sleeves (should any of the remaining sleeves crack and separate) on the reactor fuel system and operation with thermal sleeves removed at NA-1 is provided below.

2.0 Staff Evaluation

2.1 Mechanical Effects of Loose Thermal Sleeves on the Reactor Fuel System

The material of construction of the thermal sleeves is stainless steel, type 316 or 304. Thermal sleeves of a design different from the failed sleeves are present at the surge line and spray line nozzles at the pressurizer and in the CVCS fill lines on the RCS crossover leg. As there has been no evidence of failure of these types of sleeves, they are not considered in this safety evaluation.

Analyses of the potential effects of loose thermal sleeve parts on the reactor fuel system addressed two concerns: (1) partial flow blockage of fuel assemblies due to wedging of an object in fuel assembly flow paths, and (2) cladding wear caused by pieces lodged within an assembly or between two assemblies. Flow blockage effects are discussed in the following subsection of this evaluation. Potential mechanical effects are addressed below.

As pointed out in the submitted analysis, loose thermal sleeves, if intact, would be stopped by the fuel assembly bottom nozzle or the lower core plate because of dimensional considerations (the fuel assembly bottom nozzle flow holes are less than 1/2 inch in diameter). If the sleeves were to break apart into small pieces, it would be possible for a very small piece to pass through a flow hole and become wedged between fuel assemblies, thereby causing fretting or grid damage. This is considered improbable, however, because (a) the sleeves, which are relatively ductile, are expected to remain essentially intact and (b) the space between the fuel assemblies (~40 mil) is about one fifth the thickness of the thermal sleeve material.

With respect to the potential for transport of loose thermal sleeve pieces into the fuel assemblies, this is considered even less likely due to the very small spacing (pitch) between the fuel rods relative to the size of fragments expected from the thermal sleeves. (The smallest fragment found from a 10-inch sleeve in the Trojan plant was ~1/3 x 1/2 x 1/7 inch whereas the spacing between Westinghouse 17x17 fuel rods is roughly 120 mils.)

Even if fretting wear of fuel rod cladding were to occur, the worst result would be the release of fission products, which would be detected and monitored by the activity monitors. Such fretting failures, however unlikely, would be apt to start as small perforations, which would not be expected to worsen in a sudden or catastrophic manner or to propagate from rod-to-rod. Any attendant fission product release would be gradual and would not be permitted to exceed Technical Specification values on coolant activity. We conclude, therefore, that there is reasonable assurance that the potential effects of loose thermal sleeve parts would be tolerable and manageable with respect to fretting wear.

The impact and wedging effects of potential loose parts generated from remaining thermal sleeves whose welds are presently intact were evaluated by the licensee. Although under most unfavorably assumed circumstances coolant leakage and minor damage to steam generator, coolant pump and primary stop valve may occur, we concur with the conclusion that such leakage is likely to be small and detectable, and effects would not cause unacceptable safety concerns due to the limited available impact energy which can be impacted on randomly targeted mechanical components.

2.2 Potential Flow Blockage Thermal Hydraulic Effects

With regard to flow blockage effects of loose thermal sleeves, it is assumed that (a) the 14" sleeve from the pressurizer surge line becomes loose, travels with the hot leg flow, and finally lodges and blocks flow at the steam generator tube sheet; and (b) the remaining 3", 6" and 12" sleeves protrude into .ie cold leg flow and finally lodge in the reactor lower internals and block flow at the lower core plate. Conservatively assuming that the 14" sleeve completely blocks flow in 10% of the steam generator tubes, Westinghouse analysis showed that the total RCS flow reduction was approximately 1.13%. This RCS flow reduction will not affect thermal margin because the TCS flow will still be higher than the thermal design flow used in the transient analyses.

Concerning the core flow blockage effect, the evaluation consists of three conditions, i.e., (a) the effects of sleeve segments entrapped by the lower core plate, (b) the effects of sleeve segments entrapped by the bottom nozzle plate and (c) the effects of sleeve segments carried upward into the fuel assemblies. For the first two cases, the sleeve segments are entrapped by the lower core plate or the bottom nozzle plate. These will result in core inlet flow blockage. The flow redistribution downstream of the blockage will result in full recovery of flow in the lower portion of the active core region as predicted by open-lattice thermal hydraulic codes, such as THINC-IV. Thus inlet blockage effects would be limited to the lower portion of the core where DNB and LOCA are not limiting concerns. This is because an increase in enthalpy and decrease in mass velocity in this region would not result in the fuel rod reaching design DNBR limit and peak cladding temperature exceeding the LOCA limit. For the third case, the smaller sleeve segments are carried into the fuel assemblies, entrapped by the lower grid which has tighter clearance, and results in local flow blockage. Test data (Basmer, P., et al, "Investigation of the Flow Pattern in the Recirculation Zone Downstream of Local Coolant Blockage in Pin Bundles," ATOMWIRTSHAFT, 17, No. 8, P. 416-417, 1972) performed on open lattice fuel assemblies shows that with 41% of the subchannels completely blocked, the stagnant zone behind the flow blockage essentially disappears after 1.65 l/De (the ratio of the actual length of the fuel channel versus the hydraulic diameter). It was also found that leakage flow through the blockage tends to shorten the stagnant zone. Thus, the local flow blockage has little effect on subchannel enthalpy rise and minor perturbation in local mass velocity. Therefore, we conclude that the local flow blockage would not affect DNB.

With regard to the effect on LOCA performance due to a small broken sleeve piece being lodged in the fuel region, the Westinghouse analysis has concluded that the currently-docketed North Anna LOCA analysis (Letter from R. H. Leasburg (VEPCO) to H. R. Denton (NRC) Serial No. 627, November 12, 1981) provides margin that can offset this additional flow blockage. In that analysis, a flow reduction penalty based on 75%

blockage in the hot assembly was assessed. The maximum blockage given in the NUREG-0630 report is 71.5%. The excess margin (i.e., 3.5%) is sufficient to offset the adverse Peak-Cladding Temperature (PCT) increase due to the presence of a small piece within the hot assembly.

Inasmuch as it is quite unlikely that such a small piece would (a) be produced, (b) travel to and become lodged in the maximum blocked section of the hot assembly, and (c) have a significant impact on the PCT, we conclude that there is reasonable assurance that this type of flow blockage will not be a problem in NA-1.

2.3 Materials Considerations

No unacceptable material would be introduced into the reactor systems as a result of the failure of a thermal sleeve. Minor clad damage could occur on the surfaces of carbon steel components; however, this would present no safety or operational concern due to the very slow corrosion rate of the carbon steel in the reactor coolant environment.

3.0 Operation with Thermal Sleeves Removed

The licensee's submittal indicated that a detailed stress analysis was performed to verify the structural integrity of the 3 inch charging line sleeve (B loop) and the 6 inch safety injection sleeve (A loop) with removal of the thermal sleeves. Finite element techniques were used to calculate thermal stress effects under the most adverse operating transients, including their contribution to cumulative fatigue damage.

Such analysis was performed by Westinghouse and is similar to the analysis performed for the McGuire, Trojan, and NA-2 facilities, which have been reviewed and accepted by the staff. The analysis includes the whole nozzle structure, connecting weld and affected portions of the connected pipe. In addition to operating transients, all other mechanical loads were included. The submittal also indicated that the stress and fatigue damage are within the allowable limits set by subsection NB of Section III of the ASME Code.

4.0 Augmented Surveillance

Attachment V of the licensee's submittal describes the NA-1 loose part surveillance program. In addition to the existing Loose Parts Monitoring System (LPMS) the licensee is installing a supplemental LPMS on NA-1 to monitor thermal sleeves remaining in the reactor coolant system. The supplemental LPMS consists of installing accelerometers on each loop cold leg between the 12-inch accumulator discharge line and the reactor vessel, and an accelerometer located on Loop C hot leg between the 14-

inch surge line and the loop isolation valve. All accelerometers will be installed one to two feet downstream from the thermal sleeves to be monitored. In the control room, a Westinghouse control cabinet will provide continuous alarm monitoring.

Prior to and during plant startup, the control room operators will be instructed about the potential for loose parts in the RCS. Specific instruction will be provided by Westinghouse in the use of the supplemental LPMS. The LPMS will be functionally tested at least once every 31 days. Audible checks of the LPMS are being performed by the Shift Technical Advisor on a daily basis. Control rod exercises will be performed on a weekly basis to verify the free movement of the control rod assemblies. Reactor coolant activity will be monitored at least once per 72 hours to detect any potential clad damage, and the incore instrumentation thimbles will be verified as operable on a weekly basis. In addition, NA-1 already has incorporated limiting conditions for operation and surveillance requirements for the LPMS in the Technical Specifications. The staff concludes that the surveillance program is acceptable.

5.0 Conclusions

Based on the above, we find that operation of NA-1 for fuel cycle 4 with thermal sleeves as described above is acceptable. The staff is presently proceeding with a generic review of the thermal sleeve issue which should be concluded in the next few months. The results of this study and the long-term resolution of the thermal sleeve issue will be made known to all Westinghouse reactor licensees should it require further licensee action.

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