VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

December 21, 1990

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

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Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNIT 1 AND UNIT 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING AMENDMENT REQUEST TO DELETE THE RESIDUAL HEAT REMOVAL (RHR) AUTOCLOSURE INTERLOCK (TAC NOS. 76956 AND 76957)

By letter dated October 16, 1990 the NRC staff requested additional information to complete its review of our Technical Specification amendment request deleting the Residual Heat Removal autoclosure interlock. The additional information requested is provided in the attachment.

Should you have further comments or questions, please contact us.

Very truly yours,

RJ Saunders
for W. L. Stewart
Senior Vice President - Nuclear

Attachment

A061
1/1

cc: U. S. Nuclear Regulatory Commission
101 Marietta Street, N.W.
Suite 2900
Atlanta, Georgia 30323

Mr. M. S. Lesser
NRC Senior Resident Inspector
North Anna Power Station

Mr. Leon B. Engle, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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ATTACHMENT

NRC Question (1):

The staff position taken in WCAP-11736-A was that an alarm shall be added to each RHR suction valve which will actuate if the valve is open and the pressure is greater than the open permissive setpoint and less than the RHR system design pressure minus the RHR pump head pressure [justified by WCAP-11736]. The PRA analysis in the WCAP-11736 includes these alarms. Your proposed changes do not include alarms to each RHR suction valve and your procedural arguments do not provide an acceptable alternate. Justification that your proposal provides an overall reduction in risk needs to be provided, or the aforementioned alarms need to be provided.

Response:

WCAP-11736-A provides justification for the removal of the Residual Heat Removal (RHR) suction valve Autoclosure Interlock (ACI) based on a decrease in core damage frequency. The WCAP utilizes probabilistic risk analysis (PRA) to address three areas of concern: 1) the likelihood of an interfacing system LOCA, 2) RHR availability, 3) low temperature overpressurization concerns. These PRA analyses assume that an annunciator is added to warn of leaving one RHR suction valve open when the RCS is fully pressurized. In response to NRC questions the Westinghouse Owners' Group (WOG) (Letter OG-89-01, dated January 3, 1989), states the WOG position that an alarm is not required for plants whose RHR system is located inside the containment. The reason is that the likelihood of an interfacing system LOCA with severe consequences is precluded.

We agree with the WOG position in that the questioned alarm provides only an additional means of protection against potential RHR overpressurization. The impact of the annunciator has been examined by reviewing the three areas of PRA analysis as detailed in Appendix B of the WCAP. North Anna was the lead plant examined for one of the RHR design groupings. The proposed annunciator has no effect on the calculated RHR availability or on the low temperature overpressurization concerns. The only PRA area affected by the proposed annunciator is for the likelihood of an interfacing system LOCA.

The North Anna interfacing system LOCA, however, is modeled different from most Westinghouse plant models, since the North Anna RHR system is entirely within the containment. A failure of the RCS/RHR interface is not an event V-sequence (interfacing LOCA outside containment) initiator, but rather, is enveloped within the LOCA initiators inside containment (i.e., "A" - large LOCA, "S1" - medium LOCA, and "S2"&"S3" - small LOCA for NUREG/CR-4550 Vol. 3). The North Anna V-sequence interfacing LOCA outside containment initiator generally involves the RCS/LHSI (Low Head Safety Injection system) interface where LHSI is a separate system from RHR, with separate pumps, valves, piping etc. It is important to note that a LOCA inside containment is quite different from an interfacing LOCA outside containment by three major factors:

1) radiological releases can be controlled by the containment,
2) water inventory remains within the containment for long term core cooling, and therefore,
3) core damage can be mitigated by engineered safety features.

The calculation for the frequency of an intersystem LOCA between the RCS and RHR systems is described in Appendix B of WCAP-11736-A. Two cases are calculated. One case is for the current configuration with the ACI, 9.28E-7/yr, and the other is for no ACI and a new annunciator, 5.77E-7/yr. The frequency for the ACI without adding the new annunciator was not evaluated by the WCAP. We have calculated the frequency of an intersystem LOCA between the RCS and RHR systems to be 3.05E-6/yr without the new annunciator. This slight increase in frequency for this scenario does not have a significant impact on the total core damage frequency for North Anna LOCAs inside containment.

We have evaluated the impact of the RCS/RHR interfacing LOCA inside containment using the Surry NUREG/CR-4550 Vol. 3 Rev. 1 PRA. Surry and North Anna have similar RCS and RHR system designs, and the front line systems for LOCA modeling are similar. The generic LOCA initiator frequencies for the Surry analysis are taken from NUREG/CR-4550 Vol. 1 Rev. 1, Table 8.2-4, and are based upon a variety of PRA models for Westinghouse (4 loop and 3 loop plants), Babcock and Wilcox and Combustion-Engineering PWRs. These LOCA initiator frequencies are considered estimates, as LOCA events are rare. Small break LOCAs have occurred infrequently while medium to large LOCAs have not yet occurred. The Surry or generic NUREG/CR-4550 LOCA frequencies range from 5E-4/yr for large LOCA to 2E-2/yr for the small-small LOCA (S2). The change frequency for the North Anna RCS/RHR interfacing LOCA inside containment event from 9.28E-7/yr to 3.05E-6/yr is insignificant compared to the inherent uncertainty and magnitude of the generic NUREG/CR-4550 frequencies.

The most limiting case is the large LOCA sequence A with a frequency of 5E-4/yr. At 3.05E-6/yr, the North Anna RCS/RHR interfacing LOCA inside containment frequency is less than 1% of the sequence A frequency. Considering the variation in Westinghouse plant configurations including 4 loop vs 3 loop, with and without loop stop valves, etc., a contribution of 1% to the smallest LOCA initiator frequency is felt to be insignificant. Furthermore, LOCAs inside containment can be mitigated with plant engineered safety features to prevent core damage.

Additional system failures must be combined with the initiator frequency to estimate core damage frequency. The Surry NUREG/CR-4550 LOCA core damage frequencies of 2E-6/yr for large LOCAs and 3.1E-6/yr for medium LOCAs suggest that the North Anna RCS/RHR interfacing LOCA inside containment event, which contributes less than 1% to initiator frequency, would impact overall core damage frequency in the range about 1 E-8/yr. We feel this contribution to overall core damage is insignificant.

Additional perspective can be gained by comparing the core damage frequencies of the North Anna RCS/RHR interfacing LOCA inside containment and the generic Westinghouse RCS/RHR interfacing LOCA outside containment. Generally, the Westinghouse frequency of 5.77 E-7/yr equates to a core damage frequency of 5.77 E-7/yr, as there is no mitigative action to prevent core damage for the event. The North Anna core melt frequency of about 1 E-8/yr is much less due to the ability to mitigate LOCAs inside containment with engineered safeguards systems. Also, these core melt frequencies are for evaluations at power. The overall change in the core damage frequency should consider the reduction of loss of RHR system event during shutdown conditions, which is generally a qualitative argument at this time.

In summary, the removal of the RHR ACI without installing the sub[illegible] annunciator will result in a core damage frequency at about 1E-8/yr, which is much less than the 5.77E-7/yr core damage frequency for Westinghouse plants with either ACI or the new annunciator (with the RHR system located outside of the containment) even though there is an increase the North Anna initiating event frequency for a LOCA between the RCS and RHR system from the previous value of 9.28E-7/year to 3.0E-6/year. Also, the North Anna RCS/RHR interfacing LOCA inside containment is an insignificant contributor to the overall core damage frequency, using estimates of core damage from the Surry NUREG/CR-4550 PRA. Coupling this insignificant change in core damage frequency to the gains in RHR system availability during shutdown will result in an overall reduction in the likelihood of core damage when all modes of operation, both power and shutdown, are considered.

NRC Question (2):

If suction valve alarms are to be provided, the valve position indication to the alarm must be provided from the stem mounted switches (SMLSs) and power to the SMLSs must not be affected by power lockout of the valve [justified by WCAP-11736]. If suction valve alarms are provided, will valve position indication be maintained available in the control room upon removal of power to the RHR suction MOVs?

Response:

No new suction valve alarms or Control Room annunciators will be added. Current RHR high pressure annunciators provide adequate warning of both RHR valves left in the open position when the RCS is being pressurized above the RHR design pressure and do not require valve position information in order to alert the operators of this undesirable condition. Other annunciators providing indirect indication that RHR pressurization is occurring and that one or both of the RHR high volume suction relief valves are relieving include, high Pressurizer Relief Tank (PRT) temperature, high PRT pressure and high PRT level. Surveillance testing will be reviewed to ensure adequate testing exists for these annunciators.

NRC Question (3):

If suction valve alarms are to be provided, procedural improvements associated with the alarm should be implemented. These should include procedures to: (a) confirm operability of the alarm circuitry including a surveillance procedure to make sure the alarm is operable, (b) recognize the alarm so that when there is an alarm, it will be known what it is for and what should be done, and (c) avoid pressurizing until the problem is ascertained and proper steps taken. If suction valve alarms are to be provided, please describe your procedure improvements relative to the above.

Response:

No new suction valve alarms or Control Room annunciators will be added. Current annunciators provide adequate warning of both RHR valves left in the open position when the RCS is being pressurized above the RHR design pressure. The appropriate annunciator response procedures will be reviewed to ensure the Control Room Operators have sufficient guidance to respond to all annunciators which might indicate that both RHR suction valves have been left open during RCS pressurization. The operator action included within the procedure will be to close the RHR suction valves, stop pressurization and return to a safe-shutdown mode of operation.

NRC Question (4):

You have stated that you will verify isolation of the RHR system by closing and de-energizing both remote operated RHR suction isolation valves and locking the associated breakers. This is done prior to exceeding 500 psig RCS pressure. What means are used to assure the valves are closed subsequent to de-energizing the breakers at times when a leak check is not performed.

Response:

The RHR suction valves are confirmed to be closed prior to de-energizing the breakers by utilizing the valve position indication lights in the Control Room. These lights provide direct indication that the valve is either full closed, mid position or full open. Once the breakers have been de-energized, the valves cannot be opened remotely and the valve operators are sized so that they cannot be opened manually if the differential pressure is greater than 600 psi. At normal operating pressure, the minimum differential pressure would be greater than 1700 psi. Independent verification of the MOV position prior to de-energizing the breaker has been added to the operating procedures.

NRC Question (5)

In a telecon you indicated that the RHR suction valve operators are sized so that the valves cannot be opened against full system pressure. Please confirm this.

Response:

The RHR suction valves are designed to stroke against a differential pressure of 600 psi. The valve operators are sized so that the valves cannot be opened at nominal RCS operating pressure.

References:

(1) N. B. Closky, et al.: "Residual Heat Removal System Autoclosure Interlock Removal Report For The Westinghouse Owners Group," WCAP-11736-A, Volume I and II, Westinghouse Electric Corporation, Power Systems, P.O.Box 355, Pittsburgh, Pennsylvania 15230 (October 1989).

(2) NUREG/CR-4550.