VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

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U. S. Nuclear Regulatory Commission
Washington, D. C. 20005

Serial No. 98-587A
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Docket Nos. 50-280
50-281
50-338
50-339
License Nos. DPR-32
DPR-37
NPF-4
NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
NORTH ANNA POWER STATION UNITS 1 AND 2
GENERIC LETTER 97-01, DEGRADATION OF CRDM/CEDM NOZZLE AND OTHER VESSEL CLOSURE HEAD PENETRATIONS
REQUEST FOR ADDITIONAL INFORMATION (RAI)

In letters dated April 28, 1997 and July 25, 1997 (Serial Nos. 97-214 and 97-214A, respectively), Virginia Electric and Power Company (Virginia Power) provided our response to Generic Letter 97-01, "Degradation of CRDM/CEDM Nozzles and Other Vessel Closure Head Penetrations," dated April 1, 1997, for North Anna and Surry Power Stations. Our response indicated that we were actively involved in the integrated industry effort to establish the appropriate inspection activities for the reactor vessel head penetrations. In a September 16, 1998 letter, the NRC staff requested additional information within ninety days to complete their review of our response as it relates to the Westinghouse Owners Group's (WOG) integrated program for assessing vessel head penetration nozzles and to the contents of Topical Report No. WCAP-14901.

Virginia Power has continued to be involved with WOG and Nuclear Energy Institute activities to address the NRC staff's inquiries on the inspection activities and WCAP-14901. Due to the concurrent industry activity to generically address the staff's concerns, we requested until January 15, 1999 to provide our response to the request for additional information. This additional one month to respond was discussed with, and agreed upon, by Mr. N. Kalyanam of your staff on December 14, 1998.

No new commitments are intended as a result of this letter. If you have any questions or require additional information, please contact us.

Very truly yours,

L. N. Hartz
Vice President - Nuclear Engineering and Services

cc: U.S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth Street, SW
Suite 23T85
Atlanta, Georgia 30303

Mr. R. A. Musser
NRC Senior Resident Inspector
Surry Power Station

Mr. M. J. Morgan
NRC Senior Resident Inspector
North Anna Power Station

Attachment

NRC Request for Additional Information (RAI) Regarding Utilities Participating in the Westinghouse Owners Group (WOG) Response to Generic Letter 97-01:

"Background and Methodology for Evaluation of Reactor Vessel Closure Head Penetration Integrity for the Westinghouse Owners Group"
Topical Report No. WCAP-14901, Rev. 0

Surry Power Station Units 1 and 2
North Anna Power Station Units 1 and 2

Virginia Electric and Power Company

Virginia Power Response to:
NRC Request for Additional Information (RAI) Regarding Utilities Participating in the Westinghouse Owners Group (WOG) Response to Generic Letter 97-01:
"Background and Methodology for Evaluation of Reactor Vessel Closure Head Penetration Integrity for the Westinghouse Owners Group"
Topical Report No. WCAP-14901, Rev. 0

Applicability of Topical Report No. WCAP-14901, Rev. 0, to the Plant Specific Responses to GL 97-01 for Participating Member Utilities in the WOG

Background:

On September 16, 1998, the NRC staff issued an RAI to Virginia Power (Serial No. 98-587) concerning questions that had arisen in reviewing Virginia Power's response to GL 97-01 (see Virginia Power letters dated April 28, 1997 (Serial No. 97-214) and July 25, 1997 (Serial No. 97-214A)). In those two letters, Virginia Power claimed credit for participation in the WOG integrated inspection program, in lieu of performing additional plant specific inspections. The WOG inspection program is documented in Topical Report No. WCAP-14901, which was submitted to the NRC staff on July 25, 1997. The RAI specifically deals with the applicability of WCAP-14901 in assessing the reactor vessel head penetration (RVP) nozzles at both North Anna 1 & 2 and Surry 1 & 2.

On December 11, 1998, NEI submitted a list of generic responses to the RAI's, developed by the Alloy 600 Issue Task Group with input from the three PWR Owners Groups and EPRI. Efforts were made by the Task Group to use consistent responses when common RAI questions were asked. With the NRC's permission, the Task Group rephrased the common RAI questions in order to further promote commonality and expedite the review process. The paraphrased questions along with pertinent portions of these generic responses are provided below.

Response to RAI Question #1a: *"Provide the latest model susceptibility ranking of your plant based on the model analysis results, including the basis for establishing the ranking of your plant relative to the ranking of other plants in your owners group analyzed using your model."*

For industry planning purposes, plants have been grouped into three categories based on the predicted time to reach the allowable flaw depth limit. These results are provided in the industry histogram provided as Figure 1 of the December 11, 1998 NEI submittal (attached).

North Anna 1 and Surry 1 are grouped in the composite model industry category calculated as most susceptible to reactor vessel head penetration (RVP) cracking, whereas North Anna 2 and Surry 2 reside in the intermediate cracking susceptibility group. However, North Anna 1 performed limited volumetric inspections in February 1996 at the most susceptible locations and found no cracks. Moreover, as summarized earlier in Virginia Power's initial response to GL 97-01 (Serial No. 97-214A), WCAP-14552 (a plant specific RVP analysis at North Anna and Surry) showed that cracking at all four Virginia Power units should not lead to leaks. That is, the driving force for RVP cracking propagation drops to zero before through wall cracks can extend above the RVP partial penetration weld, the pressure boundary.

Response to RAI Question #1b: *"Describe how the crack initiation and growth models for assessing postulated flaws in the nozzles were benchmarked, and list and discuss the standards that the models were benchmarked against."*

The Westinghouse models and software used for the probabilistic analysis of reactor vessel head penetration nozzles were developed using the structural reliability and risk assessment (SRRA) methodology. The application of this SRRA methodology to piping for risk-informed ISI was extensively benchmarked against hand calculations, available failure data and alternative calculations as described in WCAP-14572, Revision 1, Supplement 1 (October 1997). According to NEI, the NRC is currently planning to issue a SER accepting this application of SRRA shortly.

As described in Table 4-2 of WCAP-14901 (July 1997), the SRRA probabilities for Alloy 600 primary water stress corrosion cracking (PWSCC) compare very well with inspection observations at four plants, where sufficient information existed to perform calculations for the worst head penetration nozzle at the time they were first inspected. While two of the plants (D. C. Cook 2 and Ringhals 2) with relatively high calculated probabilities had observed flaw indications, two other plants with lower calculated probabilities (Almaraz 1 and North Anna 1) did not. The initial WOG probabilistic model was revised as a result of the North Anna 1 inspection observations and an independent peer review was performed by Alloy 600 PWSCC specialists (Jim Begley and Brian Woodman) at APTECH Engineering in the spring of 1997.

Response to RAI Question #1c: *"Provide additional information regarding how the model will be refined to allow the input of plant-specific inspection data into the model's analysis methodology."*

There are two kinds of variations that are considered in the Westinghouse probabilistic analysis: random and systematic. The random variation is that due to localized material variability and other effects with insufficient information available to completely characterize them. This could include the effect of the variation in surface roughness on crack initiation and the variation in the actual weld size on the local stress. For these types of uncertainties, a Bayesian updating process has been developed by Westinghouse that could be used to combine the prior distribution on time to failure, which gives the initial calculated probability of failure with time, with the observations from the inspection. The updated posterior distribution that is generated in this manner can then be used to generate an updated estimate of the probability of failure with time for each penetration that was inspected.

The systematic or mechanistic type variations, such as the time to crack initiation being inversely proportional to the stress to the 4th power, are included directly in the Westinghouse probabilistic model. If the observations from an inspection would differ significantly from what was calculated, then the basic model would need to be revised. This in fact has already occurred based upon the observations from the North Anna Unit 1 inspections. The revised model now provides calculated probabilities that are consistent with the current inspection observations (see previous response to question 1b).

Response to RAI Question #1d: *"Describe how the variability in the product forms, material specifications, and heat treatments used to fabricate each CRDM/CEDM penetration are addressed in the crack initiation and growth models."*

Since the Westinghouse probabilistic analysis models are mechanistically based, uncertainties are provided to directly account for the variability in such fabrication related input parameters as nozzle wall thickness, material grain boundary carbide coverage and monotonic yield strength. The Westinghouse mechanistic model also accounts for the variability in indirect fabrication related effects, such as the variation in surface roughness on crack initiation and the variation in the actual weld size on the local stress, where there is insufficient information to describe the causes and effects in a statistically significant manner. Specifically, the model input also includes the observed uncertainties on the coefficients used to calculate residual stress, initiation time and crack growth rate.

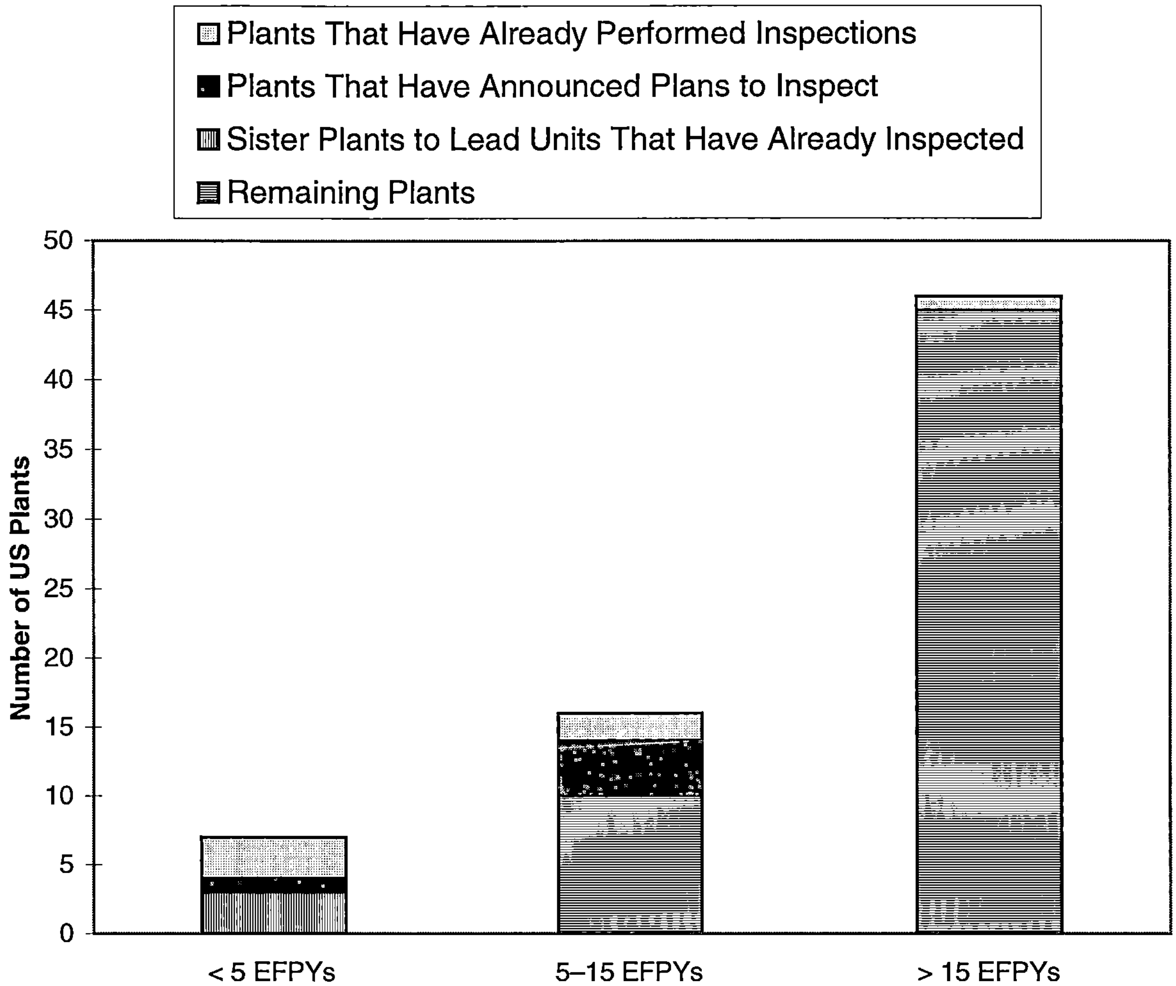
Response to RAI Question #2: *"Table 1-2 in WCAP-14901 provides a summary of the key tasks in the WEC vessel head penetration nozzle assessment program. The table indicates that tasks for 1) evaluation of mitigation methods, 2) crack growth data and testing, and 3) crack initiation characterization studies are still in progress. In light of the fact that the predictive models appear dependent in part on crack initiation and growth estimates, provide your best estimate of when these tasks will be completed, and describe how these activities relate to and will be used to update the susceptibility assessments at your plant."*

The programs on crack growth testing and crack initiation have been essentially completed, and the program on mitigation is now underway and targeted for completion in mid-2000. These programs have thus far served to confirm the assumptions used in the original safety evaluations and models. As additional information becomes available from the referenced testing, the models will be reviewed and updated as necessary. No major changes are anticipated.

Response to RAI Question #3: *In the NEI letters of January 29, 1998, and April 1, 1998, NEI indicated that inspection plans have been developed for the RVP nozzles at the Farley Unit 2 plant in the year 2002, and the Diablo Canyon Unit 2 plant in the year 2001, respectively. The staff has noted that although you have endorsed the probabilistic susceptibility model described in WCAP-14901, Revision 0, other WOG member licensees have endorsed a probabilistic susceptibility model developed by an alternate vendor of choice. The WOG's proposal to inspect the RVP nozzles at the Farley Unit 2 and Diablo Canyon Unit 2 plants appears to be based on a composite assessment of the RVP nozzles at all WOG member plants. Verify that such a composite ranking assessment has been applied to the evaluation of RVP nozzles at your plant(s). If composite ranking of the RVP nozzles at WOG member plants have been obtained from the composite results of the two models, justify why application of the probabilistic susceptibility model described in WCAP-14901, Revision 0, would yield the same comparable rankings of the RVP nozzle for your plant(s) as would application of the alternate probabilistic model used by the WOG member plants not subscribing to WCAP-14901, Revision 0. Comment on the susceptibility rankings of the RVP nozzles at your plant(s) relative to the susceptibility rankings of the RVP nozzles at the Farley Unit 2 and Diablo Canyon Unit 2 plants.*

The announcement of inspection plans by individual WOG plants is the result of each individual plant's economic situation, along with their future operational plans. The individual plant results are all compared in the histogram in Figure 1. An individual plant's category in the histogram is one of the many considerations which must be evaluated in making inspection decisions.

North Anna 1 and Surry 1 are grouped in the same composite model industry category calculated as most susceptible to RVP cracking as is Farley 2. North Anna 2 and Surry 2 reside in the intermediate cracking susceptibility group with Diablo Canyon 2. However, North Anna 1 already performed limited volumetric inspections in February 1996 at the most susceptible locations and found no cracks. Moreover, as summarized earlier in Virginia Power's initial response to GL 97-01 (Serial No. 97-214A), WCAP-14552 (a plant specific RVP analysis at North Anna and Surry) showed that cracking at all four Virginia Power units should not lead to leaks. That is, the driving force for RVP cracking propagation drops to zero before through wall cracks can extend above the RVP partial penetration weld, the pressure boundary.



Effective full power years (EFPYs) from 1/1/97 until probability of having a crack at the allowable depth matches DC Cook 2 probability of one 75% through-wall crack at time of its 1994 inspection

Figure 1. Industry Histogram for Reactor Vessel Head Penetration Nozzle Primary Water Stress Corrosion Cracking (PWSCC)