January 21, 1999

Mr. J. P. O'Hanlon
Senior Vice President
Virginia Electric and Power Company
5000 Dominion Boulevard
Glen Allen, VA 23060

SUBJECT: CLOSURE OF THE TASKS REGARDING GENERIC IMPLICATION OF THE PART-LENGTH CONTROL ROD DRIVE MECHANISM HOUSING LEAK AT PRAIRIE ISLAND UNIT 2 FOR NORTH ANNA POWER STATION, UNITS 1 AND 2 (NAPS 1 AND 2)

Dear Mr. O'Hanlon:

By the enclosed letters dated August 11 and December 23, 1998, the NRC has responded to Westinghouse Owners Group (WOG) positions regarding corrective actions to address generic aspects of the part-length control rod drive mechanism housing issue that originated as a result of the leak at Prairie Island, Unit 2, on January 23, 1998. The WOG program is a voluntary industry initiative to address this issue.

By letter dated August 14, 1998, Virginia Electric and Power Company discussed its participation in the WOG initiative and the activities at NAPS 1 and 2 resulting from that initiative. The NRC staff's review of these letters and results has been completed.

As discussed in the December 23, 1998 letter, the NRC staff has concluded that, given the marginal increase in risk and the small number of welds with potentially reduced safety margins, the actions taken under the industry initiative are acceptable for protecting public health and safety. Accordingly, our review of these tasks is considered complete.

If you have questions regarding this letter, you can contact me at (301) 415-1480.

Sincerely,
ORIGINAL SIGNED BY:
N. Kalyanam, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Letter from B. Sheron to L. Liberatori dated December 23, 1998
2. Letter from B. Sheron to L. Liberatori dated August 11, 1998

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 21, 1999

Mr. J. P. O'Hanlon
Senior Vice President
Virginia Electric and Power Company
5000 Dominion Boulevard
Glen Allen, VA 23060

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cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 23, 1998

Mr. Lou Liberatori, Chairman
WOG Steering Committee
Indian Point Unit 2
Broadway & Bleakley Ave.
Buchanan, NY 10511

SUBJECT: PART-LENGTH CONTROL ROD DRIVE MECHANISM HOUSING ISSUE

Dear Mr. ~~Liberatori~~: Lou

This letter provides the staff's response to your letter of October 15, 1998, transmitting WCAP-15126, "Technical Assessment of the Part Length CRDM Housing Motor Tube Cracking In Westinghouse Owners Group Plants." Your October 15, 1998, letter was in response to the August 11, 1998, NRC letter on this subject and contains the Westinghouse Owners Group (WOG) position regarding corrective actions to address generic aspects of the part-length control rod drive mechanism (CRDM) housing issue that originated as a result of the leak that occurred at Prairie Island, Unit 2, on January 23, 1998. The staff considers the WOG program as a voluntary industry initiative in lieu of a regulatory action to address this issue. The staff notes that affected licensees have provided commitments to follow the recommendations of the WOG in addressing this issue.

In our August 11, 1998, letter we requested that the WOG address whether it agreed with the staff's statistical analysis regarding the potential number of defective welds that could be left in service. If WOG agreed with the staff analysis, then we requested that the WOG address why it believes leaving up to six defective welds in service is acceptable. Finally, we asked what modifications WOG would propose to the inspection program to address the staff concerns. WCAP-15126 contains conclusions similar to staff conclusions regarding the potential number of defective welds that could be left in service. However, to address the latter two questions, the WCAP refers to USNRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" and contains an assessment of the probability of core melt as the basis for your conclusion that no further actions beyond the approximately 36% sample of welds inspected or replaced are necessary.

From the review of the information provided regarding fabrication history and metallurgical root cause analysis, it cannot be precluded that additional cracked housings remain in service. Further, if cracks similar to those found at Prairie Island were in service, safety margins would be significantly less than specified by 10 CFR 50.55a through its implementation of Section XI of the ASME B&PV code for the CRDM housings. The sampling based inspection program for Type 309 welds performed by the WOG provides a 95% confidence that less than about 3% of the uninspected welds are likely to be defective. We agree this would limit the potentially significant number of severely degraded components in service to that assumed in the WOG risk assessment.

9812310167

Enclosure 1

We compared the WOG's resolution of this case, including its use of probabilistic risk assessment, with the guidance provided in Regulatory Guide 1.174. As noted above, with 95% confidence safety margins should be unaffected for all but as few as 3% of the uninspected welds.

We agree that the incremental core damage frequency for the range of defects that might be present is of the order of 10^{-6} per reactor year. Given this marginal increase in risk and the small number of welds with potentially reduced safety margins, we conclude that the actions taken are acceptable for protecting public health and safety.

Sincerely,

[original signed by:]

Brian W. Sheron, Acting Associate Director
for Technical Review
Office of Nuclear Reactor Regulation

cc: N. Liparulo
A. Drake
J. Bastin
H. Sepp

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 11, 1998

Mr. Lou Liberatori, Chairman
WOG Steering Committee
Indian Point Unit 2
Broadway & Bleakley Ave.
Buchanan, NY 10511

SUBJECT: PART-LENGTH CONTROL ROD DRIVE MECHANISM HOUSING ISSUE

Dear Mr. Liberatori:

This letter contains the NRC staff's evaluation of the Westinghouse Owner's Group (WOG) proposed resolution of the part-length control rod drive mechanism (CRDM) housing issue that originated as a result of the leak that occurred at Prairie Island Unit 2 on January 23, 1998. Following the staff's review of the initial information on this event, the NRC formally requested WOG to activate the Regulatory Response Group on February 20, 1998. The staff met with the RRG on February 27, 1998 to discuss this issue. On March 6, 1998, RRG issued a letter that requested the affected owners to docket their plans for addressing the issue within 30 days and initiate compensatory measures for RCPB leakage. The options identified by RRG for the plan were:

- Remove the housings and cap the reactor head penetrations
- Perform non-destructive examinations to confirm the absence of any cracking
- Perform additional records search to better identify applicability and obtain other data to confirm the absence of any cracking
- Address the capability of using additional RCS leakage monitoring awareness while the issue is being resolved

The NRC found these recommendations an appropriate and acceptable response to the identification of the QA breakdown at the vendor's shop and as suitable corrective actions for the potential very large defects that jeopardize RCS integrity.

The staff met with WOG representatives in a number of public meetings, the most recent of which was held on June 11, 1998. During this meeting WOG provided its conclusions based on the weld inspections, fabrication records review, safety assessment, and statistical evaluation of the inspections planned and performed (assuming that no additional flaws are identified in the planned inspections). Its conclusions are that (1) the Prairie Island flaw was an isolated event, (2) there is 95 percent confidence that about 95 percent of the remaining welds do not have flaws, and (3) continued operation of plants will not result in a significant increase in risk. WOG plans to close this part-length CRDM housing issue, if no further unacceptable flaw is identified in the currently planned weld inspections.

9808120260

Enclosure 2

After completing its evaluation, the staff disagreed with WOG's conclusions. The staff conveyed this determination to WOG by telecon on June 25, 1998. Specifically, the staff determined that the inspections performed to date and inspections currently planned are not adequate to assure that a similar CRDM housing weld flaw found at Prairie Island would not be present at another facility. The staff disagreed with aspects of the mechanistic, statistical, and risk evaluations presented by WOG. In light of the break down in the quality assurance program at the vendor's shop and the need to maintain the pressure boundary integrity, the staff disagreed with WOG's approach of using the 95/95 criterion of 95 percent confidence that 95 percent of the welds would not have flaws of interest to justify the sampling size of the weld inspection. This approach does not provide high assurance that the Type 309 weld buttered 403 components manufactured at Royal Industries satisfy the applicable regulation, including the required specified margins for structural integrity. In its evaluation, the staff determined that use of the acceptance criterion suggested by WOG would be inadequate to catch (with 95% confidence) as many as six defective welds in the population of 182 uninspected welds even if no additional flaws are found in the proposed WOG sample. The detailed staff evaluation is enclosed.

Based on the staff evaluation results described above, the inspection program for Type 309 welds proposed by the WOG appears to leave a potentially significant number of severely degraded components in service. An inspection program that results in high assurance that no degraded components are left in service is the appropriate goal. To accomplish this, it thus would appear necessary to either inspect essentially all the components with a qualified ultrasonic examination, or remove the components.

The staff finds the statistically based inspection program proposed by WOG for part-length CRDM that used Inconel weld filler (Alloy 82) is acceptable. The staff's basis for this is that no failures have been identified with these components and they are considered to be less susceptible to the mechanism that generated the flaw in the Type 309 weld; therefore, an inspection program based on the 95/95 criterion is acceptable for sampling the population.

I would appreciate if you would address the concerns described above. Specifically, the WOG should address (1) whether they agree with the staff statistical analysis regarding the potential number of defective welds that could be left in service, (2) if you agree with the staff analysis, why you believe leaving up to six defective welds in service is acceptable and (3) what modifications you would propose to your inspection program to address the staff concerns.

Please provide your response within 14 days of receipt of this letter so that the staff can resolve this issue in the near term and take any regulatory action deemed necessary.

Sincerely,

J. C. Lainas for

Brian W. Sheron, Acting Associate Director
for Technical Review
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: As stated

cc w/encl: See next page

Staff Evaluation on WOG's Proposed Inspection Program

for Part-Length CRDM Housing Issue

1.0 BACKGROUND

On January 23, 1998, a non-isolable reactor coolant pressure boundary leak of 0.26 g.p.m. was discovered in a part-length CRDM housing at the G-9 core location of the Prairie Island Unit 2 reactor while it was operating. Metallurgical evaluation of the failed housing confirmed ultrasonic (UT) examination results that a very deep 360° long, partial through-wall crack was present. The metallurgical evaluation results showed the flaw had been undersized by UT examination results. The failure mechanism was identified as hot tearing associated with the fabrication of the Type 309 austenitic stainless steel (309) weld buttering at the Type 403 martensitic stainless steel (403) forging. Chemical analysis results identified the following contaminants on the fracture face of the failed component: sulfur, copper, boron, and zinc.

Following staff's review of the information provided by the licensee, the NRC formally requested WOG to activate the RRG on February 20, 1998. The staff met with RRG on February 27, 1998, to discuss this issue. On March 6, 1998, RRG issued a letter that requested the affected owners to docket their plans for addressing the issue within 30 days and initiate compensatory measures for RCPB leakage. The options identified by RRG for the plan were:

- Remove the housings and cap the reactor head penetrations
- Perform non-destructive examinations to confirm the absence of any cracking
- Perform additional records search to better identify applicability and obtain other data to confirm the absence of any cracking
- Address the capability of using additional RCS leakage monitoring awareness while the issue is being resolved

The NRC found these recommendations an appropriate and acceptable response to the identification of the QA breakdown at the vendor's shop and as suitable corrective actions for the potential very large defects that jeopardize RCS integrity.

To date, affected WOG member utilities have inspected or repaired or committed to inspect or repair 102 CRDM 308/309/403 weldments on 51 assemblies at nine operating plants. There is a total population of 284 welds of the type of interest (i.e.Type 309) in 137 installed assemblies and five spare assemblies at 21 operating plants.

On June 11, 1998, representatives of WOG summarized this issue at a public meeting with the staff. Based on the weld inspections, fabrication records review, safety assessment, and statistical evaluation of the inspections planned and performed (assuming no flaws are identified in the planned inspections), WOG concluded that (1) the Prairie Island flaw was an isolated event, (2) there is 95 percent confidence that about 95 percent of the remaining welds do not have flaws, and (3) continued operation of plants will not result in a significant increase in risk. WOG indicated it plans to close this part-length CRDM housing issue, if no further unacceptable flaw is identified in the currently planned weld inspections.

2.0 EVALUATION

Materials Engineering

The staff agrees with the industry finding that the significant cracking at Prairie Island Unit 2 was fabrication-related. From the contaminants found on the failed component's fracture faces, it appears that the failed component was probably inadequately cleaned prior to weld buttering. Some of the elements found on the fracture faces are usually contained in commercial cutting lubricants. The staff agrees that the hot tearing most likely occurred during solidification of the weld butter and that subsequent post-weld heat treatment (PWHT) may have extended the cracking. Further, the heavy oxide scale found on fracture faces indicates that the open crack was subjected to the high temperatures of the PWHT that are well above plant operating temperatures. Because of the large thermal expansion mismatch between the 309 and 403 materials, care must be taken to minimize solidification cracking. The presence of contaminants, from perhaps residual cutting lubricant, would increase the chances for solidification cracking.

The CRDM housing is a safety-related Code component that was manufactured under a quality assurance program that was intended to satisfy 10 CFR 50, Appendix B. The cleaning prior to weld buttering was specified in the controlling procedure. The surface and volumetric examinations performed failed to assure quality in that a component with severe cracking was not rejected. Therefore, it is clear that the quality assurance program broke down for the failed component, in particular, with respect to Appendix B, Criterion IX-Control of Special Processes, in that cleaning prior to welding appears to be not as specified and Criterion X-Inspection, in that the examinations performed for the work operation did not identify the unacceptable defect.

The severity of the cracking found in the Prairie Island Unit 2 part-length CRDM was among the worst identified in a safety-related component at an operating nuclear power plant. The cracking found was well in excess of the depth that could be accepted by analysis pursuant to Section XI, IWB 3600. One portion of the 360° circumferential crack was through-wall and other portions of the crack were in excess of the approximately 75% Code maximum flaw depth limitation. Nonetheless, limit load fracture analysis was performed by WOG to determine the margins that existed in the flawed component. WOG stated that the average remaining ligament in the cracked component from metallography was about 25%. The failure pressure was calculated to be 2900 psi for the 309 weld. Based on WOG analysis, there was a margin of about 1.3 (2900/2250) to failure for normal and upset conditions; the ASME code-required margin of safety is 2.77. However, the staff was unable to confirm that the average uncracked ligament was 25%. It is not clear, from review of the metallography presented (WCAP-15054), if the 25% average ligament includes a "mixed zone" of small ligaments across the fracture face. WOG stated that additional capacity existed because the actual strength for the 309 weld is about 10% higher than used in the analysis. Arguments regarding the margins available for component integrity for OBE and SSE loadings are based on a calculation using material allowables higher than the design allowable. Further, WOG argues that since the flawed component passed a hydro test at 3450 psi and crack growth in service has not been identified, a margin of 1.5 is inferred. The staff does not agree that a margin based on pressure alone is indicative of component integrity structural margins. The staff's view is that the actual margin to failure is smaller than claimed, but the margin is essentially indeterminate. In part, this comes from the staff's review of the metallography and from a review of operating history. Prior to the previous refueling no leakage had been reported for the component. A leak in this reactor coolant pressure boundary (RCPB) component was discovered while the plant was operating, and the plant was taken out of service as required by the technical specifications (TS). To the staff's knowledge no unusual loadings from transients or other events had occurred prior to the discovery of the leak. The staff understands that some work was done on the reactor head during the last refueling outage and that the head was removed from and replaced on the reactor vessel during the refueling. It is possible that a load from either bumping the head during movement or when work was being performed was of sufficient magnitude to cause the crack to open and leak during the subsequent cycle of operation.

The regulations applicable to this issue are as follows:

1. 10 CFR 50.55a(g)(4) requires that throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components that are ASME Code Class 1, 2, or 3 must meet the requirements set forth in the applicable edition and addenda of Section XI for the facility.

2. 10 CFR 50, Appendix B, Criterion XVI, states that measures will be established to assure that conditions adverse to safety, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly detected and corrected.

The leaking component at Prairie Island did not satisfy the above requirements in that Section XI margins to failure were not maintained and there was through-wall leakage in the RCPB. This resulted in a reduction in defense in depth since the reactor coolant pressure barrier was breached. As explained in the following section on the staff's statistical evaluation, the inspection program proposed by WOG is inadequate.

Statistical Evaluation

At the June 11, 1998, WOG/NRC meeting, WOG reported that 35 weld inspections were performed, and found no defective welds. Based on this information, the staff performed its independent statistical evaluation, and found that this inspection would not catch (with 95% confidence) as many as 21 defective welds in the remaining population of 248 uninspected welds.

The inspection program proposed by WOG is inadequate from a statistical point of view. It does not provide adequate confidence that appropriate corrective actions are taken to ensure the Type 309 weld buttered 403 components manufactured at Royal Industries satisfy the applicable regulation, specifically the required margins for structural integrity. In order to attain this goal, it would be necessary to demonstrate with 95% confidence that there are no flaws remaining in the uninspected welds. Even if no defective welds are found in the 66 additional welds which are to be inspected, and accounting for the 35 welds already inspected, what can be demonstrated with 95% confidence, is only that there are less than seven defective welds in the remaining population of 182 uninspected welds.

The staff has the following additional comments on WOG's statistical evaluation presented at the 5/6/98 meeting (Statistical Evaluation viewgraphs):

<u>Page 3</u>: The inspection results to date are not "1 flaw in 36 welds inspected" but rather zero flaws in 35 welds inspected. The one flaw found was not the result of a random inspection and should therefore not be counted.

<u>Page 8</u>: 1. From the "82" Sub-lot column, it appears that $p \leq .0271$ with at least 95% assurance. However, there are two problems with this conclusion. First, from page 6, the value .0271 is the posterior mean. However, the Perdue-Abramson method does not use mean values. It uses the posterior distribution which is given on page 6. From this distribution, the probability of $p \leq .0278$ is .235+.637 = .872. Thus, $p \leq .0278$ with only 87 percent assurance (not with 95 percent as claimed by WOG). The only statement that can be made with at least 95% assurance (actually, 100%) is that $p \leq .0729$.

2. From the "309" Sub-lot column, it appears that $p \leq .0278$ with at least 95% assurance. Because this population is described by the prior distribution, from page 6 the probability that $p \leq .0278$ is .185 + .630 = .815. Thus, $p \leq .0278$ with only 81 percent assurance (not with 95 percent as claimed by WOG).

<u>Risk Assessment</u>

WOG risk analysis is the product of three numbers:

- the probability that a reactor will have at least one flaw,
- the frequency of operational events that might cause the flaw to fail catastrophically enough to create a LOCA, and
- the probability of failing to mitigate the LOCA.

The staff disagrees with the Westinghouse analysis on the first two values.

The Westinghouse analysis uses a probability of "~0.05 for a flaw to exist," presumably in a single plant. However, that is apparently taken from its statement that there is 95% confidence that the whole population of welds is less than 5% flawed. For a PRA, the appropriate value is the probability that one or more of the 8 to 16 welds in the plant is flawed. To determine that probability properly, the mean or "best estimate" value of the flaw rate should be used, not the 95th percentile value. Assuming that the inspection of the sample of 101 welds is completed without discovery of another severe flaw, there is 50% confidence that the rate of flaws in the remaining population is less than 0.7/101=0.0069; so there is 50% confidence that there are no more than 1.26 flaws in the remaining uninspected population of 182 welds. Together with the known flaw, that makes a total flaw occurrence rate of 2.26/284= 0.008 for the whole population. At this rate, the probability that one of the welds in a plant will be flawed is between 0.062 for 8 welds and 0.12 for 16 welds.

The Westinghouse analysis uses "~1E-03 - 1E-05/year abnormal event frequency" as the frequency of occurrence of events that might cause catastrophic failure of a flawed weld in these CRDM housings. That is apparently based on the facts that the flawed housing survived a 3450 psi hydrostatic test after fabrication and was analyzed by Westinghouse to be capable withstanding an operating basis earthquake. However, for reasons addressed elsewhere in this review, the staff is not confident in that part of the Westinghouse analysis. It is known that the flawed weld survived 23 years of service at Prairie Island Unit 2, so one estimate is that the frequency of occurrence of events that would cause weld failure is probably less than 1/23 years = 4.3×10^{-2}/reactor-year. Since none of the operating PWRs have experienced pressure transients exceeding 3450 psi or earthquakes exceeding the magnitude of operating basis earthquakes in approximately 1500 combined years of operation, one could estimate the occurrence rate of events that could fail this flawed weld as less than 1/1500 years = 6.7×10^{-4}/reactor-year. This value is just inside the upper range suggested by Westinghouse. However, it is not clear that the flawed weld at Prairie Island actually would have survived all of the operational occurrences experienced at the other PWRs to date. Although corrosive degradation of the weld during its service life was not evident, it was observed to begin leaking noticeably during the current cycle operation. Some sort of stress imposed during the outage is suspected of producing the leak that occurred later, although no actual stress inducing occurrence was noted. However, other plants have experienced such events as moderate earthquakes, cable snags and impact loading while moving the upper heads and other loads. It is not clear how these occurrences at the other plants would have affected the flawed CRDM housing. Degradation during an outage may potentially make the flawed weld more susceptible to failure during operational events.

The Westinghouse analysis used a "LOCA CCDP ~ 1E-02 -1E-04." The staff agrees that the probability is in this range for failing to prevent core damage, given a LOCA of this size. The staff's analysis uses a value of 1×10^{-3} for the conditional core damage probability due to small to medium LOCAs. This is consistent with the results of a variety of NRC and industry PRAs.

Combining the staff's factors provides a range from 5×10^{-6} to 4×10^{-8}/year over which the staff's confidence varies from good to poor that the core damage frequency due to failure of a flawed CRDM housing has been bounded. The staff's range generally overlaps and slightly exceeds the upper part of the range suggested by Westinghouse, which is "1E-06/yr to1E-10/yr."

3.0 CONCLUSION

The staff has concluded the following:

- The leaking component at Prairie Island did not meet the current regulation. In order to assure that the remainder of the population of the 309 weld buttered 403 components manufactured at Royal Industries have the required specified margins for structural integrity, and to satisfy applicable quality assurance requirements, corrective actions are necessary to provide a high confidence that the deficiencies revealed by the discovery of the weld flaw at Prairie Island Unit 2 did not result in a similar CRDM housing weld flaw at another facility.

- The inspection program proposed by WOG for the Type 309 welds is inadequate from a statistical point of view. As stated in the staff's statistical evaluation, WOG's current inspection plan is inadequate to catch (with 95% confidence) as many as six defective welds in the remaining population of 182 uninspected welds, even if no additional defective welds are found in the sample. There would be only 36% confidence that no defective weld remains in the uninspected population. Based on the staff's evaluation a combination of inspection o. repair of 100% of the 309/403 partial length CRDMs is appropriate.

- The level of risk associated with this issue at plants which have not yet inspected similar welds may be small (i.e., the CDF increment is in or below the mid-10^{-6}/reactor-year range). Considering this level of risk, the staff has concluded that it is not appropriate to require immediate action that would subject plants to additional startup and shutdown activities. The staff considers it more prudent to implement the necessary inspection or repair during the next refueling outage. This should allow planning and qualification of inspection and repair methods that will minimize personnel exposure and best integrate with other refueling activities.

Westinghouse Owners Group
Project No. 694

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