

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

10 CFR 100, Appendix A

October 10, 2011

U.S. Nuclear Regulatory Commission
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Washington, DC 20555

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VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
RESTART READINESS DETERMINATION PLAN

By letters dated September 14, 26, 28 and 30 and October 6, 2011, the NRC requested additional information (RAI) regarding Dominion's Restart Readiness Determination Plan for North Anna Power Station following the August 23, 2011 Central Virginia earthquake. By letters dated September 27, 2011 (Serial Nos. 11-520A and 11-544) and October 3, 2011 (Serial Nos. 11-544A and 11-566), Dominion responded to several of the RAI questions provided by the NRC technical review branches. However, the responses to a number of RAI questions were being developed and were therefore not included in the previous responses. As a result, Dominion is providing its responses to several of the remaining questions, as well as clarifications requested by the NRC staff on previously submitted responses, in the attachment to this letter. The specific technical review areas and the associated questions being answered are provided below for reference:

Electrical	Questions 4 and 5
Steam Generators	Questions 1 and 2
Snubbers	Questions 2 and 4
Reactor Vessel Internals	Questions 2 and 3

Dominion's response to the remaining unanswered RAI questions will be provided in subsequent correspondence.

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will

If you have any questions or require additional information, please contact Thomas Shaub at (804) 273-2763 or Gary D. Miller at (804) 273-2771.

Sincerely,



E. S. Grecheck
Vice President – Nuclear Development

Attachment:

Response to Request for Additional Information - Restart Readiness Determination Plan

There are no commitments made in this letter.

COMMONWEALTH OF VIRGINIA)

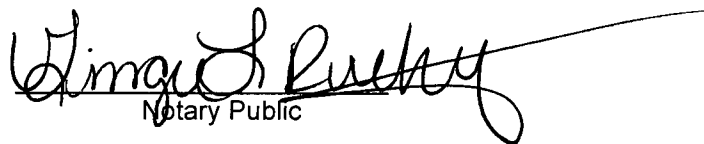
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by E. S. Grecheck who is Vice President – Nuclear Development, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 10th day of October, 2011.

My Commission Expires: 4/30/2015.

Ginger Lynn Rutherford
NOTARY PUBLIC
Commonwealth of Virginia
Reg. # 310847
My Commission Expires 4/30/2015



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Response to Request for Information - Restart Readiness Determination Plan

Attachment

**Response to Request for Additional Information
Restart Readiness Determination Plan**

**Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 and 2**

BACKGROUND

By letters dated September 14, 26, 28 and 30, 2011, the NRC requested additional information regarding the Restart Readiness Determination Plan to facilitate review of Dominion's restart activities. In letters dated September 27, 2011 (Serial Nos. 11-520A and 11-544) and October 3, 2011 (Serial Nos. 11-544A and 11-566), Dominion provided an updated status of the Restart Readiness Determination Plan and a partial response to the September 14, 2011 RAI questions.

The information provided in this attachment provides responses to and clarifications for questions in the following technical review areas: Electrical, Steam Generators, Snubbers and Reactor Vessel Internals. The NRC questions are in italics.

NRC REQUEST FOR INFORMATION

Electrical

- 4. Explain how VEPCO has evaluated the EDG and the support systems (cooling water, starting air and fuel oil) to ensure maintenance of their required safety-functions during all design basis events.*

Dominion Response

The Emergency Diesel Generators (EDGs) started upon the loss of offsite power at the beginning of the seismic event and powered their respective emergency busses until offsite power could be restored, with one exception. The 2H EDG was secured due to a coolant leak approximately 50 minutes into the event and was subsequently repaired. The cause evaluation for the 2H EDG determined the coolant leak was not caused by the seismic event.

Consistent with the EPRI NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake," North Anna developed a methodology for performing inspections to assess significant physical or functional earthquake-related damage to structures, systems, and components (SSCs). Using this methodology, inspections were performed on the EDGs and the Station Blackout (SBO) diesel generator (DG) and their support systems. The inspections did not identify any earthquake-related physical or functional damage to the EDGs, the SBO DG or their support systems that would render them incapable of performing their design functions. A more detailed discussion of the attributes of the inspections and tests performed on North Anna plant SSCs to assess any potential earthquake damage is contained in Enclosure 2 of Dominion's letter dated September 17, 2011 (Serial No. 11-520). The following equipment associated with the EDGs, as well as the SBO DG, was inspected in accordance with the guidance noted above: EDGs including the cabinets, relays, voltage regulators and breakers; the EDG support systems (cooling water, starting air, fuel oil (FO) and batteries); the SBO DG engine,

support systems, electrical breakers and busses. The FO transfer system was inspected and performance tests were run to verify the pumps supplied fuel oil to the EDGs as designed.

The following Technical Specifications (TS) surveillances will be completed prior to unit restart:

- SR 3.8.1.2: Verify each required EDG starts from standby conditions and achieves steady state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 59.5 Hz and ≤ 60.5 Hz.
- SR 3.8.1.3: Verify each required EDG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2500 kW and ≤ 2600 kW.
- SR 3.8.1.6: Verify each required fuel oil transfer pump operates to transfer fuel oil from the storage tank to the day tank.
- SR 3.8.1.7: Verify each required EDG starts from standby condition and achieves
 - a. In ≤ 10 seconds, voltage ≥ 3960 V and frequency ≥ 59.5 Hz; and
 - b. Steady state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 59.5 Hz and ≤ 60.5 Hz.
- SR 3.8.1.9: Verify each required EDG rejects a load greater than or equal to its associated single largest post-accident load, and:
 - a. Following load rejection, the frequency is ≤ 66 Hz;
 - b. Within 3 seconds following load rejection, the voltage is ≥ 3740 V and ≤ 4580 V; and
 - c. Within 3 seconds following load rejection, the frequency is ≥ 59.5 Hz and ≤ 60.5 Hz.
- SR 3.8.1.12: Verify each required EDG's automatic trips are bypassed on actual or simulated automatic start signals except:
 - a. Engine overspeed; and
 - b. Generator differential current.
- SR 3.8.1.15: Verify each required EDG:
 - a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power;
 - b. Transfers loads to offsite power source; and
 - c. Returns to ready-to-load operation.
- SR 3.8.1.17: Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal:
 - a. De-energization of emergency buses;
 - b. Load shedding from emergency buses; and
 - c. Each LCO 3.8.1.b EDG auto-starts from standby condition and:
 - 1. energizes permanently connected loads in ≤ 10 seconds,
 - 2. energizes auto-connected emergency loads through load sequencing timing relays;
 - 3. achieves steady state voltage ≥ 3740 V and ≤ 4580 V,

4. achieves steady state frequency ≥ 59.5 Hz and ≤ 60.5 Hz, and
5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.

Successful operation of the EDGs and the SBO DG in response to loss of offsite power, as well as the lack of damage found during inspections of the EDGs, SBO DG, and the associated support systems, ensures the EDGs and SBO DG will continue to perform their design functions. Functionality of the support systems will be further established during the performance of the EDG surveillance testing noted above.

5. *Explain how electrical systems were declared operable including any integrated tests performed. Was any maintenance or operator action required post seismic event to restore the integrity of any equipment required for plant safe shutdown?*

Dominion Response

Inspections were performed on plant electrical systems. The inspections did not identify any physical or functional damage to safety related or non-safety related SSCs that would have prevented any structures/components from performing their design functions as a result of the earthquake. A more detailed discussion of the attributes of the inspections and tests performed on North Anna plant SSCs to assess the potential earthquake damage is contained in Enclosure 2 of Dominion's letter dated September 17, 2011 (Serial No. 11-520). The following is an excerpt concerning inspection of electrical equipment:

For the 4160VAC, 480VAC, Vital/Semi-Vital 120VAC, and 125VDC equipment, the areas of focus consisted of four systems: Emergency Electrical (EE), Vital Bus (VB), Battery (BY), and Electric Power (EP). Comprehensive external inspections were performed in accordance with station procedure 0-GEP-30, "Post Seismic Event System Engineering Walkdown." Attachment 1, "Post Seismic Event Walkdown Checklist," of 0-GEP-30 contains the focus areas of these inspections for each type of equipment. In addition to the external inspections, an internal inspection was performed on the above mentioned equipment. This inspection was divided into categories of safety related systems and non-safety related systems. Safety related systems received nearly 100% internal inspections. For the non-safety related systems (EP), a sample of 10-15% of electrical cubicles in each Motor Control Center, which contained various types of breakers and are located in several different plant locations and elevations, were internally inspected. Focus areas of the internal inspections were as follows: 1) Wiring pull-out from terminal blocks, 2) Damaged insulators (porcelain, ceramic, or plastic), 3) Wiring pull-out from lugs, 4) Wiring harness spacing issues, 5) Backed out or missing hardware from electrical bus work, 6) Foreign material, 7) Components that have become loose from electrical sockets, 8) Insulator damage to conductors, 9) Signs of electrical flashover, 10) Odd smells or sounds of resonance, and 11) Mechanical and Electrical misalignment. The inspection results were documented in the applicable

inspection logs included in the procedure, and identified discrepancies were entered into the Corrective Action System. No physical or functional damage to safety related or non-safety related SSCs that would have prevented any electrical components from performing their design functions attributable to the earthquake was noted.

A Unit 1 and a Unit 2 list of the surveillance tests to be performed have been developed using guidance from EPRI NP-6695, Appendix B, "Typical Surveillance Tests for PWRs." To ensure a more comprehensive test program is completed prior to restart, additional testing has also been included. Surveillance tests are being performed on the plant electrical systems to demonstrate the availability and operability of those systems.

Dominion letter dated October 3, 2011 (Serial No. 11-566) details further inspections and qualification of the electrical systems, including surveillance testing performed on station batteries. The following testing will be performed prior to unit startup to ensure operability of the electrical systems:

- Functional testing of undervoltage circuitry, including testing which interrupts normal emergency bus power sources, which verifies bus load shed, EDG start and load, and proper load sequencing.
- Engineered Safety Features Actuation System (ESFAS) functional testing which initiates the ESFAS and verifies the required equipment actuates.
- EDG functional testing as delineated in the response to Electrical Question No. 4 above.

Other than repairs to the 2H EDG coolant system, other maintenance or operator action required post-earthquake to restore the integrity of any electrical equipment required for safe shutdown included:

- The Reserve Station Service Transformers (RSST) de-energized during the event due to actuation of their sudden pressure protection relays, resulting in the loss of the normal source of power to the emergency busses. The relay actuations were caused by the seismic event, not by an actual fault condition on the transformers. The transformers were re-energized and normal power restored to the emergency busses several hours into the event after the transformers were checked for damage.

In addition to the above actions, the station switchyard equipment is also being inspected and repairs made, as necessary. A discussion of the switchyard inspection activities is provided in Dominion's letter dated October 10, 2011 (Serial No. 11-577).

Steam Generators

- 1. Describe the evaluations, inspections, and analyses (if any) of the steam generators to ensure the steam generator (SG) supports, tubes, and other internals (tube support structures, steam separation equipment, j-nozzles, wrapper and wrapper supports, blowdown piping, etc) will function as designed. If a sample inspection were performed, please provide justification for limiting the scope of the inspections. Please discuss the results of the inspections highlighting any differences observed since the last inspections.*

Dominion Response

Dominion letter dated October 3, 2011 (Serial No. 11-566) provided the initial response to S/G RAI Question No. 1. In a subsequent phone conversation with the NRC staff, additional clarification of the selected S/G sampling plan and the qualifications of the bobbin probe for S/G inspections was requested. In addition, the results of the Unit 2 "A" and "C" S/Gs inspections were requested. The information below provides the requested clarification and the recently completed Unit 2 S/G inspection results.

The 20 percent sampling plan is consistent with EPRI Steam Generator Management Program Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 7, Section 3, Examination Requirements, Paragraph 3.10 Forced Outage Guidance, sub-paragraph b, Seismic Event greater than DBE, which refers to Paragraph 3.6, Tube Examination Scope and Sampling Plans. Paragraph 3.6 requires that a minimum 20 percent sample of tubes be examined. The full-length bobbin probe inspection was the sample plan identified post-earthquake.

Qualification of Bobbin Probe

EPRI examination technique summary sheet (ETSS) 27091 specifically qualifies the bobbin probe for the detection of freespan foreign object wear in the absence of a foreign object. In addition, although not specifically developed for the detection of foreign objects and foreign object wear, ETSS 96001 demonstrates bobbin probe qualification for the detection of volumetric degradation (thinning) in freespan, sludge pile, and tube support plate locations. ETSS 96004 demonstrates bobbin probe qualification for the detection of wear scars at support structures. Collectively, these ETSSs demonstrate the ability of the bobbin probe to detect volumetric degradation in locations free from interfering signals, and in regions with interfering signals caused by sludge and supports. However, it is known from industry experience that the detection of foreign objects and associated wear confined to a region very close to the top of the tubesheet is difficult for the bobbin probe. It has also been shown through industry experience that bobbin probe detection of foreign objects and foreign object wear in this region is enhanced through the use of a three frequency "Turbo Mix" which eliminates the interfering signals generated by the expansion transition and the top-of-tubesheet.

Although not specifically qualified for this purpose, it is considered good practice to use the Turbo Mix to evaluate bobbin probe data in this region.

During the North Anna fall 2011 outages, the tubes in each examined S/G were inspected full length with bobbin coil probes and the data was evaluated using the Turbo Mix. This examination was augmented with a 50% top-of-tubesheet, rotating +Point probe examination comprised of the bundle periphery region extending five tubes into the tube bundle on both legs. The periphery region is the most important with respect to foreign object wear potential at the top-of-tubesheet because it experiences the highest cross flow rate and has generated a large majority of structurally significant foreign object wear identified historically within the industry. The 100% full length bobbin probe inspection combined with the +Point periphery region examination, along with periodic secondary side visual examinations, provide a high degree of confidence that foreign objects and foreign object wear are detected at North Anna prior to becoming structurally significant.

Although outside diameter stress corrosion cracking (ODSCC) is very unlikely to develop in North Anna S/G tubing within the foreseeable future, the Degradation Assessment conservatively identified this degradation mechanism as "potential" within the historical sludge accumulation region at the top of the tubesheet on the hot leg. Consequently, the fall 2011 examinations included a +Point probe examination of 50% of the tubes within this region in each examined S/G. This 50% examination sample is adequate to ensure that the Technical Specification period examination requirements are met. There is no expectation that ODSCC may be initiated by a seismic event; hence, no increased sampling for this potential mechanism was warranted during the fall examinations.

Unit 1 and Unit 2 S/G Inspection Results

The inspection results for the Unit 1 "A" S/G were provided in Dominion's letter dated September 27, 2011 (Serial No. 11-520A).

Prior to the Unit 2 outage, tube support plate (TSP) wear was the only degradation mechanism classified as "existing" in the North Anna Unit 2 S/G tubing. Several other mechanisms were classified as "potential" (i.e., anti-vibration bar (AVB) and foreign object wear, pitting within top of tubesheet sludge region, and hot leg top-of-tubesheet intergranular attack (IGA)/ODSCC within the sludge pile region). It is primarily these damage mechanisms that were targeted by this inspection. In addition, while tube denting is not classified as a degradation mechanism, there was particular interest in the potential for any new tube denting that may have been caused by the August 23, 2011 earthquake.

The S/G tube eddy current inspections were performed per the requirements of the EPRI PWR S/G Examination Guidelines, Revision 7. The tube degradation detection methods were qualified per these guidelines and were confirmed to be appropriate for

use at North Anna. The tables below list the number of tubes inspected for the "A" and "C" S/Gs.

Unit 2 Steam Generator Examinations

Rotating Pancake Coil (RPC) Exams:		S/G "A" Tubes	S/G "C" Tubes
1	Tubesheet-Hot (TSH)±3" (50% of Sludge Region/Critical Area)	149	149
2	TSH±3" (Approximately 50% of 5 tube deep periphery)	570	570
3	TSH±3" (bundle interior outside the Sludge Region/Critical Area)	210	209
4	Row 1 U-bends (100% of the U-bends of row 1 tubes were RPC inspected)	98	98
5	Tubesheet Cold (TSC)±3" Periphery Exams (Approximately 50% of 5 tube deep periphery)	573	572
6	Special Interest Exams (Hot Leg + Cold Leg tubes requiring additional diagnostic testing)	54	59

Bobbin Exams:		S/G "A" Tubes	S/G "C" Tubes
1	Full Length from Hot Leg (Rows >3, these tubes are inspected full length from one tube end to the other as one exam)	3297	3293
2	Hot Leg Candy Canes (Rows 2 and 3, these tubes are inspected from the 7 th tube support plate on the cold leg through the U-bend, continuing through the hot leg tube end)	196	196
3	Hot Leg Straights (Row 1, these tubes are inspected from the 7 th tube support plate on the hot leg through to the hot leg tube end)	98	98
4	Cold Leg Straights (Rows 1 through 3, these tubes are inspected from the 7 th tube support plate on the cold leg through to the cold leg tube end)	294	294

(100% of the tubes were inspected full length with bobbin coil with the exception of the Row 1 U-bends which were 100% RPC inspected.)

During the previous refueling outage (2010), two objects were identified and removed from the top-of-tubesheet (TTS) region; one from S/G "A" and one from S/G "C". Based upon a visual assessment of the region at that time, it was concluded that no tube degradation had been caused by the objects (no eddy current testing (ECT) exams were performed in S/G "A" or "C" in 2010). Follow-up +Point probe exams of these tubes during the current outage (fall 2011) confirmed that the objects had in fact caused no tube degradation.

The depth of sludge accumulation at the TTS in both legs of both S/Gs was measured using the eddy current bobbin probe data. Since sludge lancing was not performed during this outage, the depth measurements convey the undisturbed geometry of the sludge pile. It was confirmed that the TTS critical area bounds the locations of measured hot leg sludge. However, on the cold leg of both S/Gs a region of sludge located outside the critical area region was identified. If this is observed during future outages an adjustment of the critical area may be warranted.

Tube Plug Examinations

Plugs previously installed in both legs of S/Gs "A" and "C" were visually examined. Each plug was confirmed to be in the expected location and no evidence of leakage or movement was identified.

Secondary Side – S/Gs "A" and "C"

Secondary side inspections included:

- Visual investigation of any accessible locations having eddy current indications potentially related to foreign objects.
- Steam drum visual inspections to evaluate the cleanliness and structural condition of the accessible sub-components such as moisture separators (including J-nozzle overspray locations), drain systems, and interior surfaces.
- Drop down examinations through the primary separators to assess the cleanliness and structural condition of the upper tube bundle and AVB supports.
- Visual inspections of J-nozzle to feeding internal interface for flow assisted corrosion.
- Visual inspections of upper tube support plates via 7th TSP handholes to assess structural condition and cleanliness, including that of TSP wedges and associated welds.

Secondary side structures and material conditions were evaluated in order to assess any potential impact on S/G tube integrity. Any foreign objects or degradation of

internals that could produce foreign objects, as well as any degradation of support structures, are important because tube integrity could be impacted. This was of particular concern due to the August 23rd seismic event. Visual examinations were performed during this outage to develop the information needed for the evaluation. Results of potential significance to tube integrity are discussed below.

J-Nozzles

The internal feeding/J-nozzle interfaces of the J-nozzles in S/Gs "A" and "C" were visually examined. The J-nozzles were confirmed to be present. The videos were reviewed side by side with videos from the previous inspections (2007) in order to identify any locations where flow assisted corrosion (FAC) may have continued to advance. This review revealed no discernible evidence of change in either S/G since the 2007 inspection.

External examinations identified some discoloration of moisture separator riser barrel and feeding OD surfaces adjacent to several J-nozzles due to overspray. No significant material loss was noted at any of these overspray sites.

Feeding UT Inspection Findings

UT thickness measurements were taken in selected regions of the S/Gs "A" and "C" feedings during this outage for the purpose of monitoring FAC related degradation. For each region examined, the minimum and average observed thickness was compared with previous UT examination results. The measurements exceeded the minimum full section design requirement of 0.350 inch by a significant margin.

Foreign Object Wear

As discussed earlier, the S/G work activities performed during this refueling outage included secondary side visual inspections of the steam drum and upper tube bundle in S/Gs "A" and "C." These examinations identified no foreign objects or conditions which could credibly generate foreign objects capable of impacting tube integrity.

Extensive eddy current examinations were performed in S/Gs "A" and "C." This included bobbin probe examinations of each tube, rotating probe examinations of hot leg TTS sludge pile locations, and substantial rotating probe examinations at the TTS in the five tube deep periphery of both legs. No evidence of foreign objects or foreign object wear was identified.

Secondary Side Internals Degradation

No degradation of secondary side internals which could impact tube integrity prior to the next examination was identified during this outage. Significant FAC advancement is not expected in the future. No tube support deficiencies were identified; hence, there are no

known conditions which could impact tube stability. These findings continue to support the planned secondary side inspection intervals incorporated into the long-term secondary management plan.

Conclusions

As indicated by the results of the current outage primary side and secondary side examinations, the North Anna Unit 2 "A" and "C" S/Gs continue to satisfy the structural and leakage integrity requirements delineated in the Dominion S/G Program and North Anna Plant Technical Specifications. Specifically, no degradation exceeding the performance criteria was identified during this or any previous North Anna Unit 2 S/G inspection. This evaluation has demonstrated that there is reasonable assurance that operation of the North Anna Unit 2 S/Gs until the next scheduled examination in each steam generator (spring 2016 or earlier for S/Gs "A" and "C"; fall 2014 or earlier for S/G "B") will not cause the structural or leakage integrity performance criteria to be exceeded. In addition, the absence of conditions which challenge the S/G program performance criteria validates prior outage operational assessment assumptions and conclusions regarding structural and leakage integrity.

No degradation of secondary side internals which could impact tube integrity prior to the next examination was identified during this outage. Significant flow assisted corrosion (FAC) advancement is not expected in the future. No tube support deficiencies were identified; hence, there are no known conditions which could impact tube stability. Again, these findings continue to support the planned secondary side inspection intervals incorporated into the long-term secondary management plan.

- 2. Provide any new or updated information from the "ongoing investigation" into the source/origin of the loose parts found in the S/G A hot leg channel head. Could the loose parts have broken off from an RCS component (as opposed to having been introduced as a foreign object during a maintenance/inspection activity)? If so, how can it be assured that the subject RCS component can still perform its safety function?*

Dominion Response

Although the cause evaluation is not yet complete, the available data indicates that the parts found in the Unit 1 "A" Steam Generator, see picture below, are not components from within the Reactor Vessel or the Reactor Coolant System. Material analysis has concluded that the material is 300 series stainless steel. The material appears to be some kind of conduit clip; however based on size and shape, it does not match any parts that are stocked at North Anna. It does not appear to come from any components in the RCS, Reactor Vessel, Safety Injection, or RHR systems. Comparison against parts associated with the equipment used for the last 10-Year Vessel ISI Inspection (from inspections performed in 2009) and refueling equipment has not identified a

suspect part. The pieces are too large to have flowed backwards through the S/G tubes to get into the hot leg bowl. Therefore, the only credible source of the foreign materials is that it was introduced through the refueling cavity to the "A" S/G hot leg. Review of the loose parts monitor data indicates that the part may have been introduced during the 10-year Vessel ISI Inspection performed in 2009 refueling outage; however this data is not conclusive. Dominion is continuing to work towards identifying the source of the foreign material and will address any generic issues that may be identified.



Snubbers

2 Please confirm that an evaluation of snubbers has been performed, and how you determined whether a snubber was locked. If a snubber was determined to be unacceptable (e.g., deformation, damaged bearings, missing or broken pin, fluid leak in the hydraulic snubber, etc.) during the visual examination (item 1, above), please discuss the results of these evaluations.

Dominion Response

Dominion letter dated October 3, 2011 (Serial No. 11-566) provided the initial response to RAI Question No. 2. In a subsequent phone call with the NRC staff, additional clarification of the snubber testing and the associated deficiencies indentified was requested. The information below provides the requested clarification.

There are 326 safety related snubbers (TRM program) for Unit 1 and 362 safety related snubbers for Unit 2. Each snubber was visually inspected by qualified engineers (Level II, VT-3). On Unit 1, four (4) snubbers were bench tested to confirm functionality due to low fluid levels being identified. Six (6) additional Unit 1 snubbers were identified as needing cleaning (oil on snubber) or minor repair (one pipe clamp needed to be slightly rotated on the pipe). These visual inspection results are consistent with the previous visual inspection performed in the spring of 2009.

On Unit 2, five (5) snubbers were removed for bench testing due to low fluid levels being identified. Four (4) of the five (5) tested with satisfactory results and one (1) is still awaiting testing. In addition,

- One (1) snubber was replaced for a suspected oil leak, although the snubber had adequate fluid for visual acceptance. The snubber was subsequently functionally tested with satisfactory results.
- One (1) snubber was identified with a rotated pipe clamp. This has been repaired (aligned) and the snubber was functionally tested with satisfactory results.
- One (1) snubber was identified with a bent attachment lug. Inspection by engineering has determined that the damage (bent attachment) appears to be caused by application of a lateral load and not due to the earthquake. The attachment lug will be repaired and the associated snubber will be replaced and functionally tested.

These visual inspection results are consistent with the previous visual inspections performed in spring 2010.

4. *Please confirm that the testing of snubbers (small bore and large bore) as required by Technical Requirement Manual (TRM) Section 3.7.5, has been performed to ensure the operability of all the snubbers. [NRC authorized the use of alternative TRM Section 3.7.5 in lieu of the ASME Code requirements in the safety evaluations for Relief Request, CS-001, for North Anna Unit 1 (ADAMS # ML091350058 dated June 10, 2009) and Relief Request N2-I4-CG-001 for North Anna Unit 2 (ADAMS # ML110260022 dated January 28, 2011)]. Please confirm that snubbers tested during the initial visual examinations (Item 1, 2 and 3) were not included in the sample test performed per the TRM.*

Dominion Response

Dominion letter dated October 3, 2011 (Serial No. 11-566) provided the initial response to RAI Question No. 4. In a subsequent phone call with the NRC staff, additional clarification of the plans for Unit 1 functional snubber testing was requested. The information below provides the requested clarification.

In addition to the TRM functional test group for Unit 2, an additional twelve (12) small bore snubbers have been selected from Unit 1 for functional testing. This sample was developed based on a combination of various buildings/elevations, ease of access (ALARA, scaffold concerns, etc), and snubbers expected to experience high loading during a seismic event. Also, two (2) large bore snubbers on Unit 1 will be tested. The Unit 1 snubber testing will be completed prior to entering Mode 4 on Unit 1.

Reactor Vessel Internal (RVI) Components

2 *The Electric Power Research Institute Materials Reliability Program (MRP) developed guidelines for use by the industry in developing aging management programs for Pressurized Water Reactor (PWR) internals. The goal was to ensure the long-term safety, integrity, and reliability of PWR internals using proven and familiar methods for inspection, monitoring, surveillance, and reporting. This guidance, which is contained in MRP-227, "Pressurized Water Reactor Internals Inspections and Evaluation Guidelines," is applicable to all U.S. PWRs and includes visual and nondestructive examination inspection of RVI components. In the submittal dated September 17, 2011, the licensee stated that a visual examination of the RVI components will be performed at North Anna, Unit 2. Considering the potential impact of the recent seismic event, and the fact the North Anna, Units 1 and 2 do not have commitments to MRP-227, address how the visual examination of the North Anna, Unit 2 RVI components will be performed (to ensure that no functional damage occurred). Will the examinations performed achieve the equivalent objectives of MRP-227?*

Dominion Response

The inspections performed on the North Anna Unit 2 reactor internals were consistent with the intent of MRP-227 to ensure the long-term safety, integrity, and reliability of PWR internals. No adverse conditions were observed. MRP-227 is an aging management guideline focused on the long term effects on the reactor internals of certain aging mechanisms, and the inspections that would detect the onset of them and manage the effects of those that are or may become active. It employs several assessment techniques to identify components that are most critical to maintaining the design safety functions considering the potential aging effects and their likelihood of occurrence. A leading indicator approach and sampling techniques are used in its recommended inspection program. Because MRP-227 is an aging management program focused on evolving material properties, and a seismic event is a dynamic loading that may challenge material strength at a point in time but does not substantially alter material properties, many of the inspections required by MRP-227 are not appropriate for post-seismic event inspections. However, its goal of maintaining safety related functions and identifying relevant conditions that are specific to the component and its structural condition is fully appropriate to use as a guide. The inspections that were performed on the North Anna Unit 2 reactor internals are therefore consistent with the intent of MRP-227 but differ in detail from those required by the guideline.

Seismic effects for a Level 0 event expect no damage to seismically designed SSCs and little to no damage to non-seismically designed equipment, so it is reasonable and consistent with MRP-227 to select leading indicators of such effects – the components or portions of components expected to be most structurally sensitive to seismic loads. The selection basis for such components was to focus on areas subject to the following criteria:

- More flexible and therefore potentially more likely to respond to seismic frequencies – these tend to be extended structures such as instrumentation conduit
- Components which span between and connect relatively flexible structures, such that they may be subject to imposed displacements
- Anchorage details of larger components – typically bolting
- Details of interfacing components where large seismic loads are transferred – typically pins and keyways

EPRI NP-6695 does not provide specific selection criteria for reactor internals other than to mention the reactivity control system (which would include control rod guide tubes). The above list however is consistent with the types of design features that are common in other equipment addressed by the guideline.

Although this was not an aging effects inspection as required by MRP-227, the aged condition of the component materials was considered in terms of how it might affect a seismic failure mode. In particular, it is recognized that the irradiated materials would be less ductile than unirradiated materials and that irradiation creep may have reduced bolting preloads. Also, irradiated stainless steel, when subject to very high sustained stress levels, is susceptible to irradiation assisted stress corrosion cracking (IASCC). Baffle-former bolts would be a leading indicator for this mechanism. However, the North Anna reactors uses Type 316 stainless steel bolting material, and cracking of these or the similar Type 304 has not been detected to date among US domestic reactors. This experience includes ultrasonic inspections of the Farley and Crystal River reactors. Also pertinent to North Anna, the Surry reactor baffle-former bolts, with a more susceptible Type 347 material, have been inspected to MRP-227 standards and only three out of 2176 bolts inspected (0.1%) had indications of IASCC. Therefore, IASCC was not considered to be a prevalent preexisting condition for the reactor internals. Also, North Anna Unit 2 has just recently in 2010 performed its ASME XI 10 year ISI B-N-3 inspections, including the upper and lower internals, with no relevant conditions noted.

MRP-227 also addresses fatigue, either due to loss of preload and subsequent alternating loading conditions or anticipated plant operating evolutions and transients. Seismic loading cycles are considered when performing fatigue evaluations. However, since there are very few equivalent full range cycles in a given seismic event, the incremental fatigue induced by a seismic event is negligible unless the loading results in large plastic strains. For example, a seismic event is typically considered to have about 10 cycles of equivalent full range loading. This is bounding for the August 23, 2011 earthquake. For 10 cycles of loading, the allowable stress amplitude permitted by ASME III Appendix I for austenitic stainless steel is 708 ksi, which is equivalent to 2.5% strain. Such high stress or strain levels would clearly plastically deform the component (such as a support) such that the physical results would be readily visible during the

visual inspections. On the other hand if the stresses were more consistent with the observation of negligible damage, for example twice yield stress, then at 78 ksi stress amplitude the Code would permit 5000 cycles of loading. The August 23, 2011 earthquake would use only 0.2% of the permitted number of cycles. Therefore, if there is no observable deformation, any fatigue aging of the materials due to the seismic event is negligible.

In the historical Section 3.9.1.2.4 the UFSAR includes a description of preoperational tests performed on design-similar plants to North Anna, HB Robinson and Trojan. The objectives of the tests were to detect high cycle fatigue cracking or wear from harmful vibrations. High cycle vibrations are caused by flow turbulence and reactor coolant pump vane pass pulsations. For the prototype testing, 35 locations were examined after 10 days of operation, equivalent to 10^7 cycles. No indications of fatigue cracking or wear were noted. The section also notes that subsequent long term operation of HB Robinson and similar plants have identified no additional high cycle fatigue or wear issues. Consistent with the concept of type-testing, similar detailed preoperational fatigue testing of the North Anna reactor internals was not required.

Of the 35 points of potential fatigue/wear reviewed in the prototype testing, several were inspected in the additional inspections performed for North Anna Unit 2 in response to the seismic event; however, the relevant mechanism and potential effects for a severe seismic event are different from those due to high cycle fatigue loading and wear. For high cycle fatigue, fatigue cracking is expected to be tight with a small crack opening dimension, while for the ten effective strong motion cycles expected during a severe seismic event the damage if any would appear as bending, distortion, tack weld disruption or failure.

Summary of Inspections Performed

Consistent with MRP-227 and EPRI NP-6695, and considering the design approach of redundant components to achieve the design function, the additional inspections were performed on a sample of the total population of each component type. Nevertheless, the functionally critical interface keyways were inspected. Further, to supplement the additional inspections, general overview visual inspections were performed on the reactor internals. The objective of these inspections was to identify any components with observable misalignment or damage. The overview inspections were not limited to the components subject to the additional inspections, but also included the accessible components.

The following table lists the reactor internals components selected for additional inspection, the basis for selection, and the adverse conditions considered indicative of seismic damage.

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Item	Potential Adverse Conditions	Basis for Selection and Remarks
Base of control rod guide tube (CRGT) upper assembly (Extension above Upper Support Plate)	- disrupted weld of lock device on bolt head	The cantilevered guide tube extensions are subject to bending moments due to seismic accelerations. Loads are resisted by the bolts.
Upper core plate keyway inserts, including bolting	- skewed, broken or missing parts - Upper portion of insert bent - denting of insert - deep scratches of insert - disrupted weld of lock device on bolt head	The keyway insert is a bearing surface for the core barrel pins. Although not expected, seismic loading could possibly generate high bearing loads on the insert surfaces, and there may be relative displacement in the radial direction. Any displacement of the keyway would affect the insert bolting.
Upper Support Column Bases (Cast stainless steel)	- distortion of bolt flanges	The support columns are in compressive loading during normal operation but seismic bending moments may be generated at the base, causing high stress in the base bolt flange.
Upper Support Column base bolting	- disrupted weld of lock device on bolt head	The support columns are in compressive loading during normal operation but seismic shear forces may be generated, causing transverse loading of bolt and potential failure of crimped lock device weld.
Core Exit Thermocouple instrumentation conduit supports	- bent or broken support - disrupted support weld - broken support clamp	High loads may have been imposed on these support items due to connection between the mixer column and the upper support column.
Core Exit Thermocouple instrumentation conduit compression fittings	- missing or broken lock device on compression fitting	Flexible small diameter conduit may have responded more strongly to seismic event than other more rigid components, generating bending loads across compression fitting.
Guide Tube lower flange welds	- disruption of weld due to overload	This is a weld of interest because it is required by MRP-227, but does not experience especially high seismic loads and the flange is not loaded in bending.
Core Barrel/Upper Core Plate Guide Pins	- indentations, gouges	These pins are subject to lateral seismic loading from the upper core plate and fuel assemblies.

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Item	Potential Adverse Conditions	Basis for Selection and Remarks
Baffle-former bolts	<ul style="list-style-type: none"> - disrupted or missing weld of bolt head lock bar - missing lock bar - backed out or missing bolt head - if any adverse conditions identified, expand scope to all baffle bolts 	There are a total of 1088 baffle bolts in eight rows of 136 bolts each. A sample is sufficient. The bolts are 316 cold worked (CW) material. The sampled bolts are subject to high differential pressure and irradiation embrittlement, and may be susceptible to IASCC. The sample of 204 bolts examined at former levels 5, 6, 7 also addresses the DC Cook baffle bolting OE.
Baffle edge gaps	<ul style="list-style-type: none"> - unusually wide gaps compared to others - variation in gap width depending on height - if either of above two conditions identified, examine adjacent edge bolts to baffle bolt criteria above 	Increased gaps could allow baffle jetting and increased bypass flow.
Core Plate	<ul style="list-style-type: none"> - displaced or broken fuel pins 	Adverse condition of these would also be detected by failure to properly seat fuel assembly.
Core Support Columns (Below Core Plate)	<ul style="list-style-type: none"> - deformed or broken upper webs that support core plate 	These are cast austenitic stainless steel items that may be embrittled due to thermal and irradiation embrittlement. They are normally under compressive loading, making damage very unlikely. However, these items are also of interest due to being an inspection item of MRP-227.

In the above table, several of the potential adverse conditions are disruption of lock device welds. These are selected as very sensitive indicators of deformation or slippage of the underlying bolt. Since the welds are basically tack welds, they would be expected to visibly tear or shear under transverse displacement and be detectable by normal visual techniques.

The inspection of the guide tube flange weld areas were included even though they were not expected to experience especially high loading during a seismic event and are not considered seismically sensitive. Transverse loads at this lower point in the guide tube are transferred through the flange to the split pins, and via them to the upper core plate. Since no bending loads are present at this pinned connection, structural loads transmitted through the welds are relatively small. However, this weld is a matter of interest since it is listed for inspection by MRP-227 for aging effects due to stress corrosion cracking (SCC) and vibration fatigue. A recent detailed inspection of these welds at Ginna detected no indications of such mechanisms, and North Anna is several years away from the age at which such detailed inspections would be recommended. If there were such preexisting conditions at this location and the seismic loading was

large, then the welds would be expected to visibly tear or disrupt in a visible way. Therefore, the lack of visible damage supports the conclusion that even if present, the hypothesized preexisting conditions did not adversely impact the seismic capability of the joint.

Similarly, inspection of a small sample of the lower support columns below the lower core plate was included as a matter of interest because it is a potential item for inspection required by MRP-227. Some embrittlement is expected in the cast austenitic stainless steel employed in the upper portions of the column, which contacts and supports the lower core support plate. Although likely present to a moderate degree, such embrittlement is not a concern for these columns because they are compressively loaded by the weight of the core and compression force of the fuel bundle springs. Therefore the inspection was supplementary only, to confirm that no damage occurred below the lower core plate.

The inspection did not include items requiring removal of the lower internals. The lower internals were removed during the previous outage and were subject to the normal ASME XI 10 year ISI inspections with no relevant conditions noted. The major components of the lower internals are assembled into a rigid monolithic structure and would cause indications of distress to the bolted baffle assembly if overall structural deformation occurred. The integrity of the bottom mounted instrument supports is assessed by considering the lack of damage of the core exit thermocouple tubing in the upper internals, together with the normal operational checks that are performed upon startup of the reactor. A foreign object search below the core plate, including eight (8) support columns, was performed to detect any loose parts generated by lower internals damage.

The inspections were performed on a best effort basis using remote cameras mounted on hand held poles and submarines. Viewing conditions permitting assessment of surface features were achieved in many cases, but were not sustained for each inspection due to the variable conditions and interference by adjacent components. The components or areas selected for additional examination had adequate viewing conditions for detection of the adverse conditions listed above. A comparison of the results achieved for the additional inspection items listed above and prior ASME XI 10-year ISI B-N-3 VT-3 inspections showed that they were comparable.

Like MRP-227, the inspection plans included expansion of the inspections to additional components if adverse conditions were noted.

The actual inspections of the reactor internals for North Anna Unit 2 during the current forced shutdown were conducted in two phases, one including items located on the upper internals and one for items on the lower internals. The inspections were performed under the direction of NDE Level 3 inspectors and the video results reviewed by an engineer with experience in seismic analysis and a participant in the development of MRP-227. The results of the general overview inspection and the additional

inspection of the seismically sensitive items listed above were that each item was satisfactory and no adverse conditions as listed above were observed. In addition, results of the most recent ASME XI 10-year ISI were reviewed and no differences in the condition of the reactor internals were noted. Finally, the normal foreign object search performed on the lower core plate and in the lower bowl of the reactor vessel found no items related to the reactor vessel internals.

In conclusion, the inspections performed on the North Anna Unit 2 reactor internals were consistent with the intent of MRP-227 to ensure the long-term safety, integrity, and reliability of PWR internals. No adverse conditions were observed. Considered together with the performance checks and validations, as described in Enclosure 3 of Dominion's Sept. 17, 2011 letter (Serial No. 11-520) and Dominion's response to Reactor Vessel Internals RAI Question No. 1 in Dominion's Oct. 3, 2011 letter (Serial No. 11-566), the capability of the internals to perform their safety related functions is ensured.

- 3 *NAPS Updated Final Safety Analysis Report (UFSAR), Section 4.2.2.1 describes the design basis for the NAPS RVI structural support components which includes consideration of seismic loads, either from an operating-basis earthquake (OBE) or a design basis earthquake/safe-shutdown earthquake (DBE/SSE), as described in Table 3.2-1. UFSAR Table 3.7-4 lists maximum seismic stresses, maximum combined stresses (including seismic stresses) and the allowable stresses for safety-related plant components under OBE and DBE/SSE conditions. The NAPS UFSAR contains many citations for specific design criteria related to seismic loads. For example, UFSAR, Section 4.2.2 of the UFSAR states transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell and distributed between the lower radial support to the vessel wall, and to the vessel flange and that the transverse loads of the fuel assemblies are transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by upper core plate alignment pins that are welded into the core barrel. In addition, Table 4.2-1 provides the maximum allowable deflections for the RVI component design at NAPS. Section 4.2.3.1.4 of the NAPS UFSAR states that a dynamic seismic analysis is required for the control rod drive mechanisms when a seismic disturbance has been postulated. The UFSAR specifically states that this analysis is needed to confirm the ability of the mechanisms to meet ASME Code Section III allowable stresses and to confirm its ability to trip when subjected to the seismic disturbance.*

For the UFSAR citations provided above, specify how the current design elements that consider seismic loading have been validated for the beyond design basis earthquake loadings at NAPS, and discuss the findings from the validation. Describe how this validation has occurred for all elements of the NAPS UFSAR that address seismic loading, and discuss the findings from these validations.

Dominion Response

The strategy for validating the capability of the reactor vessel internals to perform their design functions is based on Regulatory Guide 1.167, "Restart of a Nuclear Power Plant Shut down by a Seismic Event," which provides NRC-endorsed guidance for licensee's response to seismic events. Regulatory Guide 1.167 references the guidance of EPRI NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake," for required evaluations and inspections based on observed consequences of the seismic event. As stated in Enclosures 1 and 2 of the submittal dated September 17, 2011, evidence of inspections is consistent with Damage Intensity 0 on the EPRI seismic damage scale. EPRI NP-6695 describes how prescribed inspections and tests are keyed to the severity of the earthquake. No specific evaluations or inspections of reactor vessel internals or associated components are specified in EPRI NP-6695 for Intensity 0 earthquakes.

Nevertheless, a margin assessment of the key interface components was provided in Enclosure 3 of Dominion's letter dated September 17, 2011 (Serial No. 11-520) and in the attachment to Dominion's letter dated October 3, 2011 (Serial No. 11-566 – Vessel and Internals RAI Question No. 1). These evaluations of reactor vessel internals design margins were performed based on existing design analyses of the structural integrity of the reactor vessel internals. Only the loads calculated for a seismic event (either the OBE or DBE) were of interest for the evaluation. The calculated seismic-only loads were compared with allowable load limits which correspond to allowed stress limits for Upset conditions (Normal + OBE Loads) for which no deformation is permitted. This provides a more stringent criterion than is typically applied to the DBE loads when assessed in normal design calculations (UFSAR 3.9.3.1.1). This conservative criterion provides additional assurance that reactor vessel dimensions and geometry are maintained. The key interface loads evaluated satisfied this criterion. It is noted that the reactor vessel internals interface loads are those loads between the reactor vessel and core barrel, and between the fuel and core plates, which are used in existing design structural analyses to evaluate the structural integrity of the reactor vessel and its internals.

No specific inspections of reactor vessel internals or associated components are specified in EPRI NP-6695 for Intensity 0 earthquakes; EPRI NP-6695 specifies reactor vessel internal inspections only for Intensity 3 earthquakes. Even so, Dominion in collaboration with Westinghouse identified several inspections of the reactor vessel internals to supplement the above conclusion that the reactor vessel internals remain capable of performing their design bases functions. The details of the reactor vessel internals inspections are provided in the attachment to Dominion's letter dated October 3, 2011 (Serial No. 11-544A – Fuels RAI Question No. 8) and response to RAI Question No. 2 above. The inspection of the Unit 2 reactor vessel internals is complete. The inspections were conducted in two phases: the upper internals and the lower internals. General overview visual inspections were performed on the reactor internals to identify any components with observable misalignment or damage. Twelve (12) specific components of the reactor vessel internals were inspected in greater detail. No adverse

conditions were identified and no Condition Reports were written. Comparison with the 2010 10-year ISI inspections showed no observable changes to the condition of the internals, post-seismic event, compared to the condition in 2010. It is therefore concluded that no seismic-related damage occurred to North Anna Unit 2 as a result of the seismic event.

The reactor vessel internals in North Anna Unit 1 are of the same design as the Unit 2 reactor vessel internals that were inspected with the notable exception that Unit 1 was converted to upflow in the baffle-former region in 1996 due to the part-length baffle bolting, while Unit 2 with full-length baffle bolting remains in the original downflow configuration, and has larger operating loads on the baffle bolts. Because of this difference, Unit 2 is more susceptible to baffle bolt failure and baffle jetting, and it is appropriate to select Unit 2 for baffle region inspections. The effects of the seismic loads on the Unit 1 reactor vessel internals would therefore be similar to or bounded by the effects on the Unit 2 reactor vessel internals. No damage to the Unit 2 reactor vessel internals was identified by the visual inspections that were performed, so it is concluded that the North Anna 1 reactor vessel internals were similarly not subjected to any loads or vibrations that would adversely impact their ability to continue to safely perform their design functions. Therefore, there is a reasonable assurance that there was no damage to reactor vessel internals, and that the reactor vessel internals for North Anna Units 1 and 2 remain functional and capable of performing their design functions as identified in UFSAR 4.2.2.