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Martin, Robert

From: Rahn, David
Sent: Monday, September 19, 2011 7:17 AM
To: Martin, Robert
Cc: Khanna, Meena; Karwoski, Kenneth; Howe, Allen; McMurtray, Anthony; Bedi, Gurjendra; Lupold, Timothy; Tsao, John; Murphy, Martin; Murphy, Emmett; Mitchell, Matthew; Dennig, Robert; Mathew, Roy; Auluck, Rajender; Harrison, Donnie
Subject: RE: NA RESTART - 2nd set of RAIs
Attachments: EICB Comments on RAI second .docx

Hi Bob:

Please see the attached EICB response to your comments on Question 3 of our I&C-related RAIs.

Dave

From: Martin, Robert
Sent: Sunday, September 18, 2011 2:14 PM
To: McMurtray, Anthony; Bedi, Gurjendra; Lupold, Timothy; Tsao, John; Murphy, Martin; Murphy, Emmett; Mitchell, Matthew; Dennig, Robert; Rahn, David; Mathew, Roy; Auluck, Rajender; Harrison, Donnie
Cc: Khanna, Meena; Karwoski, Kenneth; Howe, Allen
Subject: NA RESTART - 2nd set of RAIs

Attached is the second set of RAIs that you provided in response to our request from last week. Hope I didn't miss anyone.

Please provide the requested acronymns and document titles where noted in the RAIs.

Also, please review the attached report from VEPCO and determine whether you still wish to send these RAIs out. If there is doubt or if it will take awhile to determine this, we should probably go ahead and send them out. If the questions aren't completely answered in the VEPCO report, the value of VEPCO having them asap is considerable. If its an issue where they clearly provide the answer in their Sept 17 report, then so indicate and we can remove the question.

We expect that the review of the VEPCO report to support our safety evaluation will necessitate further requests for information.

D/187

September , 2011

Mr. David A. Heacock
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2, REQUEST FOR INFORMATION REGARDING THE EARTHQUAKE OF AUGUST 23, 2011 (TAC NOS. ME7050 AND ME7051)

Dear Mr. Heacock:

On September 8, 2011, the Nuclear Regulatory Commission staff held a public meeting in Rockville, Maryland with the Virginia Electric and Power Company (VEPCO) to discuss the earthquake of August 23, 2011, and its effect on the North Anna Power Station (NAPS). We reviewed the information provided by VEPCO for the meeting transmitted requests for informaitn on fuels and reactor systems on September 14, 2011. The enclosure includes further requests for informaiton on additional topics. We received your Restart Readiness Determination Plan dated September 17, 2011 and our normal practice would be to await review of that document prior to submitting furhter requests for information. To the extent that the attached requests for information are addressed in your report, please indicate that in your response to this request.

Sincerely,

Robert E. Martin, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339
Enclosure
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OFFICE	NRR/LPL2-1/PM	NRR/LPL2-1/LA		NRR/LPL2-1/BC
NAME	RMartin			GKulesa
DATE	09 //11		09/ /11	09/ /11

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VIRGINIA ELECTRIC AND POWER COMPANY (VEPCO)

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2 (NAPS)

DOCKET NOS. 50-338 AND 50-339

The following requests for information are related to the earthquake of August 23, 2011, that occurred in the vicinity of the NAPS, as discussed in the public meeting held by the Nuclear Regulatory Commission (NRC) staff (staff) on September 8, 2011 (Reference 1). The following questions are grouped according to the format of the NAPS Final Safety Analysis Report (FSAR).

3.9 Mechanical Systems and Components

Snubbers

1. Confirm that a visual examination of all snubbers (small bore and large bore) has been performed to ensure compliance with the design basis acceptance criteria.

2. Confirm that evaluation of snubbers(s) have been performed for snubbers found to be in an unacceptable condition during the visual examination such as a locked snubber, deformation, damaged bearing, missing or broken pin, fluid leak in hydraulic snubbers, etc.

3. 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 and Relief Request N2-I4-CG-001 for North Anna Unit 2 on **Provide date & ADAMS no.**

4. Confirm that an evaluation of snubber(s) has been performed for snubbers located on an unacceptable or damaged piping system discovered during the inspection of the piping system.

5. Confirm that all the snubbers' existing design loads are greater than or equal to the new design loads based on the higher earthquake (August 23, 2011) values.

   Note: The examination of support structures and attachments where snubbers are attached are covered by the inspection of support structures.

   Piping

The information requested below is focused on the scope of the licensee's assessment of piping systems, inspection/evaluation methods, acceptance criteria, results, and corrective actions. The intent of the questions is to determine whether the ASME Class 1, 2, and 3 piping systems and any non-safety related systems that connect to safety related systems satisfy the design basis for the safety related piping so as to demonstrate that their structural integrity is maintained after the recent earthquake.

1. Pipe Stress Analyses.

The staff is interested in the affect of the earthquake on American Society of Mechanical Engineers (ASME) Class 1, 2, and 3 piping systems and any non-safety related systems which connect or could affect ASME Class 1, 2, or 3 piping systems. Will all of these systems for which stress analyses were developed be re-analyzed using the loads experienced from the earthquake on August 23, 2011, including any aftershocks? The staff requests the following specific information:

A. The list of all pipe systems whose stress analyses that have been/will be re-analyzed.
B. The list of any pipe systems whose stress analyses will not be re-evaluated. Provide justification for those systems that will not be re-analyzed.
C. Describe in detail how the pipe stresses will be re-evaluated considering the loading from the recent earthquake. Discuss how the loading from the aftershocks, in addition to the loading from the earthquake on August 23, 2011, will be considered.
D. Discuss how the seismic anchor movements resulting from the recent earthquake are considered in the re-evaluation.
E. Discuss the acceptance criteria for the stress analyses and provide references. Identify the Code of Construction, including the specific edition that was used in the original stress analyses.
F. Discuss the results of the assessment. Identify which piping systems that do not satisfy the acceptance criteria.
G. Discuss any corrective actions that would be taken for those piping systems/components whose stress analysis exceeded the ASME Code, Section III allowable limits as result of the earthquake.

2. The Condition of Piping Systems and Supports.

Much information can be obtained from the walkdown of piping systems and supports. Such walkdowns may provide insights as to damage and where additional inspections may be warranted.

2.1 For ASME Class 1, 2, and 3 piping systems:

A. Identify the piping systems that will and will not be inspected.
B. There are many piping systems that do not require stress analysis (e.g., small-bore piping). Discuss whether they will be inspected for degradation. If not, provide justification for their structural integrity.
C. Discuss whether the buried pipe will be inspected/evaluated.
D. Discuss the inspection technique that will be used and what areas will be inspected. For example, discuss whether the pipe routing (i.e., elevation and location) will be verified to ensure that the pipes have not dislocated from the original design and analyzed position.
E. Discuss how the buried pipes will be inspected.

F. Discuss whether the inspection used will be able to detect flaws inside the pipe wall thickness resulting from the earthquake. If not, discuss how the inspection will ensure the structural integrity of the piping system.
G. Discuss whether pipe insulation will be inspected for damages.
H. Discuss whether the pipes will be inspected after the insulation is removed. If not, discuss how the inspection can be effective to determine the conditions of piping components such as flanges, valves connections, support clamps, and shear lugs that are covered by the insulation.
I. Discuss how the nozzles connecting pipe to the rotary equipment (pumps/compressors/turbines), and vessels (reactor pressure vessels, pressurizers, steam generators, heat exchangers, and tanks) will be inspected.
J. Discuss how the bolted flanges will be inspected for degradation.
K. Discuss how the structural integrity of those pipes or pipe segments that are inaccessible for inspection (e.g., encased in concrete) is assessed.
L. Discuss how the operator inspects those pipes that are located in the higher elevation than the operator (whether scaffolds will be built).
M. Discuss the acceptance criteria of an acceptable pipe and provide the reference of any standards that will be used.
N. Discuss the results of inspection and identify the piping systems that are not acceptable.
O. Discuss the corrective actions for the piping system(s) that is found to be unacceptable.

2.2 For pipe support system includes spring and rigid hangers, rigid lateral struts, snubbers, clamps, I-beams, lugs welded to pipe, and base plates that are anchored to the building structures or walls either by bolting and/or welding:

A. Discuss which pipe system's supports will be inspected. Discuss whether all pipe supports in all piping systems will be inspected. If not, discuss the basis for the sample inspection selection.
B. Discuss which components in a pipe support system will and will not be inspected.
C. Discuss inspection technique.
D. Discuss whether the gaps between the pipe and the support structure (e.g., I-beams) will be inspected to verify a sufficient clearance for thermal expansion in accordance with the pipe stress analysis.
E. Discuss whether the snubbers are returned to its original position (i.e., not in the locked position). Discuss whether snubbers are removed from the pipe and tested for operability. If not removed for testing, discuss how a visual examination can determine the operability of the snubbers.
F. Discuss whether the spring hangers have been inspected to ensure proper load carrying capability after the earthquake.
G. Discuss whether the rigid struts have been inspected to ensure that it is not damaged. 4B6. Discuss whether the support base plates that are anchored to the building structures and walls are inspected for the proper attachment.

H. Discuss the acceptance criteria for the pipe support components and reference the bases. Discuss results. Provide a list of damaged pipe supports.

I. Discuss corrective actions for the degraded pipe support. If a pipe support is found to be degraded, discuss whether a stress analysis will be perform for that pipe to ensure that the pipe still satisfies the stress allowable. If not, provide justification.

3. Identify any pipe systems that contain flaws in service prior to the earthquake. Discuss whether these flaws will be inspected by ultrasonic testing (UT) to ensure the flaw(s) has not grown as a result of the earthquake prior to restart. If UT will not be performed, discuss how the flaw(s) can be demonstrated to remain within the acceptance standards of the ASME Code, Section XI, IWB-3000, as a result of the earthquake.

5.0 <u>Steam Generator (SG) and SG tubes</u>

1. Describe the evaluations, inspections and analyses of the SGs to ensure the acceptable condition of the steam generator (SG) supports, SG tubes and other SG internals (tube support structures, steam separation equipment, J-nozzles, wrapper and wrapper supports, blowdown piping, etc.).

2 Are all SGs being inspected? If not, what is the justification for limiting the inspection scope? This justification should include a description of the relative alignment of the tube u-bend planes among the different SGs. Are they parallel to one another, or are they at different angles relative to one another? The staff notes that the SGs are not axi-symmetric. For example, the anti-vibration bars (AVBs) support the u-bends in the direction normal to the plane of the u-bend, but not against in-plane motion. So, depending on the ground motion, the tube bundles of the different SGs may respond differently depending on how each SG is oriented relative to the ground motion. If the plane of the u-bends are not parallel among the SGs at the site, how has this been taken into account in selecting the most limiting SG or SGs for inspection?

3. If differences in the condition of the SG (or SGs) and its components are noted relative to earlier inspections, what criteria will apply in determining whether the inspection should be expanded to any remaining uninspected SGs?

Reactor Internals

Describe VEPCO's plan for the future updating of the NAPS design/licensing basis information related to the consideration of the effects of seismic loading on reactor internals components, to be consistent with the updated seismic hazard assessment.

6.0 <u>Containment</u>

1. Will or has VEPCO perform inservice testing (IST) on containment isolation valves during the shutdown? List the valves tested and provide the results.

2. Will or has VEPCO perform a general visual inspection of the containment consistent with Title 10 of the *Code of Federal Regulations*, Part 50, Appendix J and industry guidance in Nuclear Energy Institute (NEI) 94-01, " need title " and ANSI/ANS 56.8, " need title "? If

performed, provide the results. If not performed, list the containment inspections to be performed and provide the results.

7.0 Instrumentation & Controls

1. The staff understands that VEPCO has been examining all unusual spurious changes of state of instrumentation and control (I&C) and electrical equipment which impacted the sequence of events recorders and other post trip review logs from the August 23rd event, and that the NRC inspection staff members are confirming the licensee's actions in thoroughly investigating the root causes of unexpected equipment performance in this area.

   Confirm that any immediate follow-up actions identified resulting from this effort (e.g., required equipment replacements, enhancements/repairs in equipment mounting configurations, etc.) regarding such unexpected I&C equipment spurious actuation will take place before restart of the units.

2. The licensee's presentation to the NRC staff on September 8, 2011 identified that "comprehensive surveillance testing to validate SSC operability/performance" (448 surveillance tests) will be performed. The staff would like to understand the basis for selection of the particular I&C-related surveillance tests that are scheduled to be performed and whether the licensee has identified any additional acceptance criteria for such testing that may require additional field confirmations or additional test steps to be performed during such surveillance testing.

   For example, some reactor trip system (RTS) and engineered safety feature (ESF) periodic functional testing is performed without including the local transmitter in the loop, and some locally-mounted instrumentation devices have flexible conduit connections. Should these connections be subjected to seismic acceleration in key natural frequencies of the flexible section that are in excess of design basis conditions, the additional stress put on the instrument terminals could weaken the electrical connections at the terminal strips of the devices, which could result in momentary disruption of the signal, but not permanent disruption that would manifest itself under the static conditions normally present during a periodic surveillance test.

   a. Confirm that the possibility that loose electrical and/or mechanical connections were considered as part of the instrumentation walkdowns and was addressed as additional acceptance criteria to be tested when the instructions for the performance of such surveillance testing were developed.

   b. Confirm that the licensee has verified that safety related instrumentation (especially mechanical instrumentation) calibration remains within specification.

c. Confirm that the licensee has verified that safety related instrumentation channel response times remain within specifications. In particular, the settings of mechanically- based instrumentation devices and relays (e.g., Agastat time delay relays, and other devices) that are subject to excessive acceleration can drift, resulting in a total channel response time that could exceed analyzed event response time requirements.

3. The staff requests the licensee to confirm that the plans for start-up testing of each unit include confirmation of proper operation of non-safety, but important to safety control systems, such as would be performed as elements of the pre-operational and power ascension testing described within Appendix A to Revision 3 of Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants" to verify proper operability of the normal (non-safety related) plant control systems (e.g., feedwater control, rod control, pressurizer level and pressure controls, secondary system steam pressure control system, main turbine and feedwater pump turbine control systems, in-core instrumentation, plant annunciator and process computer systems, seismic instrumentation system, plant instrumentation grounding system, etc.).

   a. Please confirm which non-safety but important-to-safety plant systems were identified by the licensee as critical to the safe operation of the plant.
   b. Please confirm what pre-operational testing has been selected to confirm proper operability of these systems prior to start-up.
   c. Please identify what sequence of testing and administrative controls will be utilized during the planned power ascension during restart to ensure that such systems are properly operating before increasing to the next power level.

8.0 Electrical Power Systems

Prior to plant restart

1) Explain how it has been determined that all electrical equipment including electrical equipment that was commercially dedicated by the licensee, including the safety-related batteries required to function during and following a seismic event (OBE/SSE), remain qualified to perform their required safety-functions during all design basis events.

2) Explain how it has been determined that electrical connections (i.e., electrical bus bars (power and control cable and wiring connections at all voltage levels), battery, contactors, etc.) maintained their electrical connection integrity to perform their required safety-functions under both normal and accident conditions and also during and following another seismic event (OBE or SSE or beyond SSE given the magnitude of the recent earthquake).

3) Explain how it has been determined that support features associated with bus bars, battery racks, switchgear, cable raceways, containment electrical penetration assemblies,etc., are adequate to enable electrical equipment to perform their required safety-functions under both

normal and accident conditions and also during and following another seismic event (OBE or SSE or beyond SSE given the magnitude of the recent earthquake).

4) Describe VEPCO's evaluations of the emergency diesel-generators (EDGs) and the support systems (cooling water, starting air and fuel oil) regarding maintainance of required EDG safety-functions during all design basis events.

5) Explain how electrical systems were declared operable. Was any maintenance or operator action required after the seismic event to restore the integrity of any equipment required for plant safe shutdown?

6) Explain how you have determined that the neutron flux instrumentation functioned in accordance with the design requirement and the trip was valid. Covered by SNPB

7) Explain how VEPCO has determined that all pressure boundary welds are intact and will perform their intended safety functions for any postulated design basis events.

Post-restart

Based on NAPS dual unit trips, how did you determine that the offsite power system has adequate capacity and capability to mitigate all design basis events.

Aging Management – License Renewal **DLR spell out acronymns**

For all in-scope license renewal components, respond to the following:

1. For all TLAAs submitted with the License Renewal Application and it's amendments:

   - State whether the recent seismic activity has resulted in a change to the disposition of any TLAA such that the original conclusions do not remain the same.
   - For any dispositions that have changed, state how the TLAA is now dispositioned (i.e., 10 CFR 54.21(c) (1) (i), 10 CFR 54.21(c) (1) (ii), or 10 CFR 54.21(c) (1) (iii).
   - State the basis for the acceptability of the change in disposition. For example, if a disposition changed from 10 CFR 54.21(c) (1) (i) to 10 CFR 54.21(c) (1) (iii), state how the aging effects will be adequately managed throughout the period of extended operation.
   - According to the North Anna UFSAR Table 5.2-4, faulted conditions (Design Basis Earthquake) are not included in the fatigue analysis of the plant components and structures. In addition, OBE earthquakes are also not included in the fatigue analysis. Therefore, for all TLAAs submitted with License Renewal Application (LRA) and its amendments: provide revised fatigue analyses that include the impact of the August 2011 earthquake on the long term operation of the plant (40-60 years). These analyses should also include the impact of earthquake aftershocks, and consider five additional OBE level earthquakes that may occur until the end period of extended operation.

2. While the staff acknowledges that a seismic event is a near singular aging event, given that the recent seismic activity exceeded the current seismic licensing basis with multiple aftershocks, state how:

- It was concluded that no existing flaws or defects sizes were impacted such that augmented license renewal inspections need not be conducted.
- It was concluded that no new flaws or defects occurred such that augmented license renewal inspections need not be conducted.

3. The concrete containment, penetrations, isolation valves, and equipment/personnel hatches were subjected to beyond design basis seismic forces. Please describe the plans and schedule to perform the SIT, ILRT, and ILLRT to demonstrate the ability of the containment to perform its intended function during the period of extended operation.

4. State what augmented license renewal inspections will be conducted at displacement sensitive locations (e.g., tank nozzle connections, piping transitioning between buildings or from a building to the soil, where differential seismic movements occur) to confirm that there was no impact to the pressure boundary function (i.e., PB) or structural and/or function support function (i.e., SNS, SS, SSR), or state the basis for why augmented inspections are not required for programs such as Tank Inspection Activities and Buried Piping and Valve Inspection Activities, or state the basis for why such inspections are not required.

5. State what augmented license renewal inspections will be conducted for structures and piping/component supports to ensure that seismic displacements did not result in significant cracking for concrete and masonry walls, or loss of form for soil, or state the basis for why such inspections are not required.

6. LRA Section B2.2.2, Battery Rack Inspections program states that, "A seismic event would be the limiting condition for battery support rack Integrity." It also states that the program conducts visual inspections. Given that the recent seismic activity exceeded the current seismic licensing basis, state whether augmented surface or volumetric inspections will be conducted to ensure that the battery racks are capable of performing their CLB function. If augmented inspections will not be performed, state the basis why these inspections are not required.

7. During the August 2011 earthquake, the reactor internals were also potentially subjected to beyond design basis loads. Please describe the plans and schedule for inspecting the reactor internals. If the reactor internals are not planned to be inspected, please provide the basis for this decision.

Risk Assessment

Was the functionality of any non-safety related equipment credited in a risk-informed license amendment considered as part of the restart plan? If not, what is Dominion's approach to ensure the continued adequacy of such risk-informed license amendments?

References:

1. VEPCO presentation materials for public meeting of September 8, 2011, (Agencywide Documents Access and Management System (ADAMS) Accession number ML11252A006.

2. Letter, E.S. Grecheck, VEPCO, Summary Report of August 23, 2011 Earthquake Response and Restart Readiness Determination Plan, ADAMS Accession number ML

3. EPRI NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake," December 1989.