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

September 14, 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 have reviewed the information provided in the slides provided by VEPCO for the meeting and find that we need additional information as identified in the enclosure.

Sincerely,

Robert Martin

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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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 on September 8, 2011. The licensee's presentation materials from that meeting are available in Reference 1. The following questions are grouped according to the format of the NAPS Final Safety Analysis Report (FSAR).

4.0 Fuel

1. FANP Topical Report, BAW-10239(P)(A), Revision 0 (Reference 2), provides an evaluation of the Advanced Mark-BW fuel assembly in a sample reactor against the criteria defined in the Section 4.2 of the Standard Review Plan (SRP). Section 5.3.4 of the topical report discusses fuel assembly structural damage from external forces, such as the operating basis earthquake (OBE), the safe shutdown earthquake (SSE), as well as SSE + loss-of-coolant-accident (LOCA) loads. The evaluation of faulted conditions also addresses both horizontal (LOCA and seismic) and vertical LOCA effects. Based on the availability of information to date from VEPCO's presentation and the Advanced Mark-BW fuel mechanical design report, the Nuclear Regulatory Commission (NRC) staff is unable to verify the operability condition for the core internals, specifically for the fuel assemblies (grids, fuel rods, guide tubes) and control rods.

    a) Please provide a comprehensive strategy and qualifying criteria for determining the operability of these components.

    b) Provide a comparison of the predicted design basis loads (e.g., local acceleration) on the core internals and fuel assemblies against the predicted loads derived from the measured ground motion data during the seismic event. In addition, compare these predicted loads against the measured yielding load and deflection from the fuel assembly grid crush testing.

    c) Describe all sources of technical information considered in determining the operability and integrity of the fuel, including involvement of the fuel vendors.

2. Describe the extent of fuel assembly inspections which will be performed to confirm the structural integrity of the fuel. Provide specific information on how the inspections will determine that there is no distortion of the fuel lattice array or rod cluster control assembly (RCCA) guide tubes that occurred as a result of the seismic event. Also provide information on guide tube drag and rod drop testing.

3. Describe the extent of fuel assembly inspections and supporting analyses which will be performed to confirm the thermal-hydraulic performance of the fuel. Provide specific information on how the inspections will determine that there is no deflection of any fuel grid mixing vanes or any other component that will alter the thermal-hydraulic performance of the fuel bundle as a result of the seismic event. If any deficiencies are detected, provide information on the impact on the fuel departure from nucleate boiling ratio (DNBR).

4. Describe the extent of inspections and testing which will be performed to demonstrate the operability of the control element drive mechanisms.

5. Provide any nuclear fuel related information that has been gathered considering the Electric Power Research Institute (EPRI) guidance and recommendations found in EPRI report NP-6695 (reference 3). Specific information of interest is control rod drive mechanism operability as related to changes in core instrumentation readouts; changes in primary coolant radiation monitor values; changes in other parameters such as primary coolant flow, temperature, and pressure; loose parts monitoring equipment noise signatures; and primary coolant chemistry sample results.

6. Explain the rationale and extent of the operability determinations for the core components (fuel and control rods) and their support systems.

7. Provide the final root cause analysis report on the cause of the reactor trips.

8. Describe the extent of inspections on the core shroud to investigate possible changes in local flow conditions (e.g., baffle jetting, change in core bypass flow).

9. Third-burned fuel assemblies are generally located along the core periphery in locations where seismic loading may be limiting. These assemblies are at end-of-life and would be discharged to the spent fuel pool (SFP) (no reinsertion). Are there any plans to do detailed investigations and measurements (including rod pulls, dismantling, hot cell examinations) on any of the third-burned fuel assemblies located at the core periphery of Unit 2?

5.0 Reactor systems

1. Describe the evaluations, inspections and analyses of the steam generators (SG) to ensure SG tube integrity?

2. Discuss provisions to ensure that system pressure relief capabilities are maintained.

3. Discuss measures to verify overall reactor coolant system (RCS) pressure boundary integrity.

4. Describe the inspections, examinations and evaluations of the emergency core cooling systems (ECCS) that have been or will be performed to show that the ECCS will continue to perform as designed, especially under simultaneous design basis earthquake loading and ECCS design basis seismic requirements.

5. Were there any complications in residual heat removal (RHR) following the

earthquake? Did all RHR equipment perform as intended?

References:

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

2. Letter, J.F. Mallay, Framatome ANP, Inc., Publication of BAW-10239(P)(A), Revision 0, "Advanced Mark BW Fuel Assembly Mechanical Design Topical Report," October 5, 2004, ADAMS Accession number ML042820190

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

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Robert E. Martin, Senior Project Manager
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