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Dominion®

August 25, 2011

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Serial No. NA3-11-047R
Docket No. 52-017
COL/DWL

DOMINION VIRGINIA POWER
NORTH ANNA UNIT 3 COMBINED LICENSE APPLICATION
SRP 19: RESPONSE TO RAI LETTER 79

On August 2, 2011, the NRC requested additional information to support the review of certain portions of the North Anna Unit 3 Combined License Application (COLA). The responses to the following Request for Additional Information (RAI) Questions are provided in Enclosures 1 and 2:

- RAI 5820 Question 19-3 Assessment of Tornado Risks
- RAI 5848 Question 19-4 Impact of Loss of Offsite Power on PRA

This information will be incorporated into a future submission of the North Anna Unit 3 COLA, as described in the enclosures.

Please contact Regina Borsh at (804) 273-2247 (regina.borsh@dom.com) if you have questions.

Very truly yours,

Eugene S. Grecheck

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K1RO

Enclosures:

1. Response to NRC RAI Letter No. 79, RAI 5820 Question 19-3
2. Response to NRC RAI Letter No. 79, RAI 5848 Question 19-4

Commitments made by this letter:

1. Incorporate proposed changes in a future COLA submission.

COMMONWEALTH OF VIRGINIA

COUNTY OF HENRICO

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

Acknowledged before me this 25th day of August, 2011
My registration number is 7173057 and my
Commission expires: August 31, 2012

Wanda K. Marshall

Notary Public

WANDA K. MARSHALL
Notary Public
Commonwealth of Virginia
7173057
My Commission Expires Aug 31, 2012

cc: U. S. Nuclear Regulatory Commission, Region II
C. P. Patel, NRC
T. S. Dozier, NRC
J. T. Reece, NRC

ENCLOSURE 1

Response to NRC RAI Letter 79

RAI 5820, Question 19-3

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

North Anna Unit 3

Dominion

Docket No. 52-017

RAI NO.: 5820 (RAI Letter 79)

SRP SECTION: 19 – PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION

QUESTIONS for PRA and Severe Accidents Branch (SPRA)

DATE OF RAI ISSUE: 08/2/2011

QUESTION NO.: 19-3

The staff reviewed the high winds and tornadoes risk assessment in Chapter 19 of the FSAR and noted that tornadoes were not assessed for shutdown conditions, particularly modes 5 and 6. Please provide an assessment that confirms that tornadoes do not contribute more than ten percent of the total shutdown core damage frequency and total shutdown large release frequency compared to the US-APWR DC PRA. In this assessment, please consider that the containment could be open and that the capability to reclose containment during/following a high wind event could be impacted.

Dominion Response

The risk from tornadoes during modes 5 and 6 does not contribute more than ten percent of the total shutdown core damage frequency (CDF) and total shutdown large release frequency (LRF) compared to the US-APWR PRA for the standard plant design. When a tornado strikes the plant during modes 5 and 6, there is a possibility that a tornado-initiated accident scenario may be induced, with some mitigation functions made inoperable due to damage from the tornado strike. SSCs that provide decay heat removal from the reactor and spent fuel are located in seismic category I structures and will not be affected by a design basis tornado strike. The accident scenarios and the quantification results obtained from the tornado risk analysis for plant shutdown are described below.

1) Accident scenarios

Because the SSCs that provide decay heat removal from the reactor and spent fuel are located in seismic category I structures, loss of offsite power (LOOP) is the only initiating event that could be caused by a design basis tornado strike. A greater than the design basis tornado strike could affect safety-related SSCs that provide decay heat removal.

There is a possibility that the containment hatch could be open during plant shutdown. In the US-APWR design, access to the containment hatch is provided through the reactor building, which is designed to withstand a design basis tornado. The containment hatch being open does not affect the vulnerabilities of SSCs located in the containment. The containment can communicate with the outdoors through the equipment hatch via the truck bay entrance in the reactor building. However, the truck bay is located on the first floor while the containment hatch is located at an elevation substantially above the truck bay opening. Therefore, there are no direct pathways for tornado induced missiles to reach the containment equipment hatch from the truck bay. Additionally, the truck bay entrance is designed to withstand a design basis tornado when closed and an administrative control will be in place to ensure that the truck bay entrance is closed when a tornado is nearby or forecast for the immediate area. Occurrence of tornadoes intense enough to damage SSCs without a prior severe weather warning are extremely rare.

When a tornado strikes the plant, there is a possibility that a loss of decay heat removal may be induced by the LOOP, with some mitigative functions made inoperable due to damage from the tornado. The shutdown PRA was reviewed to identify possible degradation of mitigative functions that may be caused by a tornado strike. The following mitigation and support systems may be degraded by tornado-induced failures from a design basis tornado strike as discussed in FSAR Section 19.1.5:

- Alternate component cooling water (CCW) utilizing the fire protection water supply system
- Alternate CCW utilizing the non-essential chilled water system
- Non-Class 1E electric power system
- Non-Class 1E Alternate ac power supply system (this is a mitigation system for LOOP events)

Based on the results of the plant vulnerability analysis and the discussion above, tornado-induced accident scenarios were categorized into three scenarios as shown in Table 1.

2) Quantification

The initiating event frequencies F_I [/Y] for each accident scenario were estimated by applying the annual frequency of tornado wind of interest F_T [Y] based on NUREG/CR-4461 Revision 1, duration of shutdown per one refueling outage t_S [hr] , and the refueling outage cycle T_R[Y] using the equation below:

$$F_I = F_T \times (t_s/8760) \times (1/T_R)$$

The conditional core damage probability (CCDP) was calculated based on the PRA model for plant operating state (POS) 8-1, which represents mid-loop operation with cooling by the RHRS after refueling. The CCDP for POS 8-1 was considered to be a representative value for other POSs for the reasons below:

- For LOOP events with no alternate CCW available (Scenario 1 in Table 1), the CCDP is dominated by an accident sequence involving failure to restore the CCW function after recovery of emergency power by the gas turbine generators. This accident sequence is common to all POSs, and considering the impact of loss of CCW function on mitigation systems, this sequence will be the dominating accident scenario for all other POSs.
- For LOOP events with no alternate CCW available and no alternate ac (AAC) power available (Scenario 2 in Table 1), the CCDP is dominated by the accident sequence involving failures of the emergency power source that leads to total loss of AC power. This accident sequence is common to all POSs, and considering the impact of loss of AC power on mitigation systems, this sequence will be the dominating accident scenario for all other POSs.
- POS 8-1 has the least number of CCW trains and Class 1E power available. This condition will lead to higher CCDP values for the above mentioned accident scenarios and is considered to result in bounding CCDP values.

The CDF from tornado strikes during plant shutdown is estimated to be 4.8×10^{-9} per year. This value is less than 10% of the value for low power shutdown LRF from internal events reported in the US-APWR DC PRA. It can be concluded that the CDF and LRF from tornado strikes during plant shutdown will not contribute more than ten percent of the total shutdown CDF and LRF determined in the US-APWR DC PRA.

The dominant core damage scenarios from tornado strike during shutdown are as follows:

- A F-scale 5 tornado, which is beyond design basis, strikes the plant and all safety systems are damaged. This event leads directly to core damage. The frequency of this scenario is 3.7×10^{-09} per year.
- An F-scale 1 or F-scale 2 tornado strikes the plant and the plant switchyard is damaged, resulting in a LOOP event that cannot be recovered within 24 hours.

The fire protection water supply system and the non-essential chilled water system are also damaged by the tornado strike, resulting in unavailability of the alternate CCW function. Emergency power is restored by the Class 1E power sources or the AACs, but the CCW cannot be restored due to failures in either the CCW system, ESW system or the safety-related UHS cooling tower fans. Loss of CCW results in loss of residual heat removal (RHR) function and mitigation functions to inject water to the reactor. Water level in the reactor coolant system (RCS) decreases due to evaporation, and eventually the core will be uncovered. The frequency of this scenario is 6.6×10^{-10} per year.

- A tornado with an intensity greater than F-scale 3 or F-scale 4 tornado strikes the plant and the plant switchyard is damaged resulting in a LOOP event that cannot be recovered within 24 hours. The fire protection water supply system and the non-essential chilled water system are also damaged by the tornado strike, resulting in unavailability of the alternate CCW function. The turbine building is also damaged and the AAC is assumed unavailable. In the event of failures of Class 1E power sources, total loss of AC power occur and eventually, the core will be uncovered. If the Class 1E power source is available but the CCW cannot be restored after station power recovery, the core will be damaged as described in the loss of CCW sequence above. The frequency of this scenario is 4.1×10^{-10} per year.

Table 1 - Tornado Accident Scenarios for Plant Shutdown

Scenario	Wind speed	Assumed impact on plant	Initiating Event Frequency [/Y]	CCDP	CDF [/Y]
1	86-135 mph (F1 and F2 scale)	· LOOP (Initiating Event) · Loss of alternate CCW	2.4E-06	2.8E-04	6.6E-10
2	135-200 mph (F3 and F4 scale)	· LOOP (Initiating Event) · Loss of alternate CCW, and · Loss of AAC power supply	2.1E-07	1.9E-03	4.1E-10
3	beyond 200 mph (F5 scale)	Failure of safety related systems Assumed core damage	3.7E-09	1	3.7E-09
Total					4.8E-09

The FSAR will be revised to reflect the assessment described above and clarify key assumptions.

Proposed COLA Revision

FSAR Section 19.1.5 and Table 19.1-206 will be revised as indicated on the attached markups.

Markup of North Anna COLA

The attached markup represents Dominion's good faith effort to show how the COLA will be revised in a future COLA submittal in response to the subject RAI. However, the same COLA content may be impacted by revisions to the DCD, responses to other COLA RAIs, other COLA changes, plant design changes, editorial or typographical corrections, etc. As a result, the final COLA content that appears in a future submittal may be somewhat different than as presented herein.

CCW initiation event to the large release frequency (LRF) for operations at power is considered insignificant. It has been therefore determined that consideration of the site-specific UHS would have no discernible effect on the Level 2 PRA results that are based on the standard US-APWR design. Therefore, the results described below are considered sufficient and applicable.

19.1.5 Safety Insights from the External Events PRA for Operations at Power

NAPS COL 19.3(4)

Replace the second and third paragraphs in DCD Subsection 19.1.5 with the following.

The last three events listed above receive detailed evaluation in the following subsections. The first four events are subject to the screening criteria consistent with the guidance of ASME/ANS RA-Sa-2009 (Reference 19.1.201), taking into consideration the features of advanced light water reactors.

The assessment of the other external events is provided below:

The screenings for other external events are performed using the following steps taking into consideration the features of advanced light water reactors. At first, qualitative screenings are performed because they are easy to obtain lower risk from advanced reactors design features or site characteristics. The qualitative screenings are performed using the analysis reported in Chapter 2 in accordance with the guidelines of ASME/ANS RA-Sa-2009. Section 4.4 of the standard defined the initial preliminary screening criteria as supporting technical requirement EXT-B1. The five qualitative screening criteria are:

1. Lower damage potential than a design basis event
2. Lower event frequency of occurrence than another event
3. Cannot occur close enough to the plant to have an affect
4. Included in the definition of another event
5. Sufficient time to eliminate the source of threat or to provide an adequate response

Following the qualitative screenings, quantitative screenings are performed. The supporting technical requirement EXT-B2 of ASME/ANS

RA-Sa-2009 states that the criteria provided in the 1975 Standard Review Plan can be used as an acceptable basis for the screening criteria of external events. The criteria are:

i. the contribution to core damage frequency (CDF) is less than 10^{-6}/year, or

ii. the design-basis event at annual frequencies of occurrence is between 10^{-7} and 10^{-6}.

For Unit 3, a value of 10^{-7} for the annual frequency of occurrence is used as a more conservative quantitative screening criterion. If an event frequency is greater than 10^{-7}/year, perform bounding analysis or PRA to confirm that the risk is sufficiently low for advanced light water reactors such as less than 1% of total CDF. The remaining external events which do not meet the above screening criteria are assessed using a bounding analysis.

The qualitative and quantitative screenings are performed using the analysis reported in the FSAR Chapter 2 Section 2.2, Section 2.3, and Section 2.4, and Chapter 3 Section 3.5. The summary of the screenings are described in Table 19.1-205. Only Tornadoes is not screened because the probability of expected maximum tornado wind speed on the site is close to 10^{-7}.

High Winds and Tornadoes

For high winds and tornadoes, tornadoes are evaluated using level 1 PRA as a bounding analysis from the discussion in Section 2.3.1.3.2.

The following sections show the results of the tornado PRA elements: 1) tornado hazards, 2) plant vulnerabilities, 3) accident scenario, and 4) quantification.

- Tornado hazard

  A tornado wind speed hazard curve for Unit 3 was developed following NUREG/CR-4461 which also forms the basis for NRC Regulatory Guide 1.76. The tornado hazard methodology developed in NUREG/CR-4461 fully meets the requirements of ASME/ANS RA-Sa-2009 (Reference 19.1-8).

  The Unit 3 is near Lake Anna, Virginia, and is located at 38° 03' latitude and 74° 47' longitude. The tornado hazard curve has been developed based on data reported in NUREG/CR-4461 for the 2° box surrounding the site, which recorded 232 tornado occurrences from 1950 through 2003. The hazard curve produced for the Unit 3 is shown in Figure 19.1-201. Strike and exceedance frequencies for tornadoes categorized in enhanced F-scale intensity are shown in Table 19.1-201.

- Plant vulnerabilities

  Components significant to the internal events PRA were reviewed to identify component vulnerability during tornadoes. Component failures that could cause initiating events were also reviewed.

  All systems and components essential for safe shutdown and for maintaining the integrity of the reactor coolant pressure boundary are located within seismic category I buildings, which are designed to withstand the loading of a design basis tornado. The design basis tornado is described in Section 3.3 and in Table 19.1-202.

  Based on a review of components, the following were identified as potential vulnerabilities during tornadoes with intensities below the design basis tornado.

  - Plant switchyard
  - Piping of the fire protection water supply system
  - CTW for the non-essential chilled water system and associated pipings
  - Selector circuit and breakers of the alternate ac power supply system
  - Permanent buses of the non-safety power system
  - Main steam system downstream of the main steam isolation valves
  - Main feedwater system upstream of the main feedwater isolation valves

  Structure, system, and components (SSCs) will be designed using the site-specific basic wind speed of 96 mph or higher. Within this analysis, plant vulnerabilities located outdoors that are not Seismic Category I or II structures are assumed to be damaged for tornado strikes of intensity enhanced F-scale 1 and greater. In this analysis, the following systems are assumed to be damaged for tornado strikes of intensity enhanced F-scale 1 and greater:

  - Plant switchyard
  - Non-essential chilled water system - Cooling tower only

Alternate component cooling water function, which utilizes the non-essential service water system or the fire protection water supply system, is conservatively assumed to be unavailable for tornado strikes of intensity enhanced F-scale 1 and greater.

Seismic Category II structures are designed to withstand a basic wind speed of 155 mph. The Seismic Category II structure that contains PRA related equipment is the turbine building (T/B). Tornado induced failure of the T/B is conservatively assumed to have an effect on the operability of alternate ac power system. In this analysis, the following systems are assumed to be damaged by tornado strikes resulting in failure of the T/B:

- Plant switchyard
- Fire protection water supply system
- Non-essential chilled water system
- Non-safety electric power system
- Alternate ac power supply system

RAI 02.03.01-5 **NAPS ESP VAR 2.3-1**

Site-specific structures and components, e.g., UHS, are damaged by tornadoes exceeding the site-specific tornado maximum wind speed (200 mph). Direct damage to the US-APWR standard design Seismic Category I structures and components within the structure can be caused by tornadoes exceeding the design basis tornado (230 mph). Since safety-related systems are cooled by CCWS, through ESWS sharing with UHS, a tornado strike of greater than 200 mph wind speed can result in functional failures of safety-related systems. In this analysis, safety-related systems are assumed to be damaged for tornado strikes exceeding the site-specific tornado maximum wind speed (wind speed >200 mph).

- Accident scenario

When a tornado strikes the plant, there is a probability that a tornado initiated accident scenario may be induced with some mitigation functions inoperable due to damage from a tornado strike. Based on plant vulnerabilities identified in the previous section, the internal events PRA was reviewed to identify initiating events or degradation of mitigation functions that may be caused by a tornado strike. The following internal events accident initiators may be caused by a below design basis tornado strike:

- Loss of offsite power (LOOP)

- •• Main steam line break downstream of main steam isolation valves
- •• Loss of feedwater flow
- •• Feedwater line break upstream of the main feedwater isolation valves

The following mitigation and support systems may be degraded by tornado-induced failures from a below design basis tornado strike:

- •• Alternate CCW utilizing the fire protection water supply system
- •• Alternate CCW utilizing the non-essential chilled water system
- •• Non-safety electric power system
- •• Alternate ac power supply system (this is a mitigation system for LOOP events, which is initiating event potentially caused by a tornado strike)

Based on the results of the plant vulnerability analysis and the discussion above, tornado induced accident scenarios were categorized into three scenarios as shown in Table 19.1-203. The frequency of each scenario derived from the hazard fragility analysis of the T/B is also shown.

- Quantification

For the tornado induced accident scenarios, the CDF was calculated based on the internal event PRA results. The dominant core damage scenarios were the following:

- •• Failure of all safety systems by a beyond design basis tornado. This event leads directly to core damage. This CDF for this scenario is 1.2E-07/RY.
- •• Tornado strike induced LOOP caused by F-scale 1 or F-scale 2 tornado

  Plant switchyard is damaged by an F-scale 1 or F-scale 2 tornado strike and LOOP that cannot be recovered within 24 hours. The fire protection water supply system and the non-essential chilled water system are also damaged by the tornado strike, resulting in unavailability of the alternate component cooling function. If the gas turbine power generators fail and station blackout occurs, reactor coolant pump (RCP) seal loss-of-coolant accident (LOCA) will occur and eventually the core is damaged. If the CCW pumps or the essential service water (ESW) pumps fail to restart, RCP seal LOCA will occur and eventually the core is damaged. The CDF for this scenario is 1.6E-08/RY.

RAI 19-3

The total CDF caused by a tornado strike during power operation is on the order of magnitude of 1E-07/RY. Tornado induced CDF is one order of magnitude lower than the total CDF for internal events and internal flood and internal fire events.

RAI 19-3

The CDF from tornadoes during low-power and shutdown (LPSD) do not contribute more than ten percent of the total shutdown CDF and total shutdown LRF compared to the design certification PRA. Tornado events during LPSD do not have a significant contribution to risk.

External Flooding

Section 2.4.2 systematically considers the various factors that can contribute to the incident of external flooding. Based on the discussions in this section, the contribution of such events to the total CDF is considered insignificant. These events meet the preliminary screening criteria of ASME/ANS RA-Sa-2009.

Transportation and Nearby Facility Accidents

These events consist of the following:

- Hazards associated with nearby industrial activities, such as manufacturing, processing, or storage facilities
- Hazards associated with nearby military activities, such as military bases, training areas, or aircraft flights
- Hazards associated with nearby transportation routes (aircraft routes, highways, railways, navigable waters, and pipelines)

In Section 2.2.3, design basis events internal and external to the nuclear power plant are defined as those events that have a probability of occurrence on the order of about 10^{-7}/RY or greater and potential consequences serious enough to affect the safety of the plant to the extent that the guidelines in 10 CFR 100 could be exceeded. The following categories are considered for the determination of design basis events: explosions, flammable vapor clouds with a delayed ignition, toxic chemicals, fires, collisions with the intake structure, and liquid spills.

The effects of these events on the safety-related components of the plant are insignificant as discussed in Section 2.2.3. These events meet the preliminary screening criteria of ASME/ANS RA-Sa-2009.

NAPS COL 19.3(4)

Table 19.1-206 Site-specific Key Assumptions *(continued)*

Key Insights and Assumptions	Disposition
Administrative controls are in place to ensure that the truck bay entrance of the reactor building is closed when a tornado is nearby or source of high wind is forecast for the immediate area.	Section 13.5

ENCLOSURE 2

Response to NRC RAI Letter 79

RAI 5848, Question 19-4

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

North Anna Unit 3

Dominion

Docket No. 52-017

RAI NO.: 5848 (RAI Letter 79)

SRP SECTION: 19 – PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION

QUESTIONS for PRA and Severe Accidents Branch (SPRA)

DATE OF RAI ISSUE: 08/2/2011

QUESTION NO.: 19-4

§10 CFR 52.79(a)(46) of the Commission's regulations requires an applicant for a Combined License (COL) to provide a description of the plant-specific PRA and its results. NAPS COL 19.3(4) was added to FSAR subsection 19.1.1.2.1. The text in that section indicates that the only site-specific design issues considered to have the potential to influence the results of the PRA are Essential Service Water System (ESWS) design and ultimate heat sink (UHS) design. The applicant states in this section that: "In cases where it can be shown that assumptions in the certified design PRA (1) bound certain site-specific and plant-specific parameters, and (2) do not have a significant impact on the PRA results and insights, no change to the design certification PRA is necessary. Similarly, certain changes or deviations from the certified design or the certified design PRA need not be reflected in the plant-specific PRA as long as it can be shown that (1) they are not important changes or deviations, and (2) do not have a significant impact on the PRA results and insights."

The frequency of loss of offsite power and probability of recovery are site-specific parameters that may have a significant impact on the results of the PRA. The staff requests that the applicant describe these site-specific parameters for NAPS and discuss the basis for determining that these parameters have no significant impact on the results of the PRA or are bounded by the parameters assumed in the US-APWR design PRA.

Dominion Response

NUREG/CR-6890 provides data to estimate the site-specific loss of offsite power (LOOP) frequency. The site-specific LOOP frequency for the North Anna Power Station (NAPS) location is estimated to be 3.8E-2 /RY when applying the grid-specific LOOP data from the Southeastern Electric Reliability Council for the NAPS site. This value is lower than the generic LOOP frequency of 4.0E-2 /RY used in the standard design PRA. Therefore, the PRA for the standard design is bounding.

Offsite power recovery probability varies with the profile of LOOP frequency, in terms of fraction of LOOP causes (i.e. plant-centered, switchyard-related, grid-related, and weather-related). The site-specific offsite power recovery probability given a LOOP event, estimated using the data of NUREG/CR-6890, is higher than that of the generic data during the early stage of a LOOP. However, the probability of failure to recover offsite power is combined with the LOOP frequency to estimate the frequency of LOOP events exceeding certain durations. The standard design PRA results are conservative because these combined values bound those of the site-specific values. That is the combined value of LOOP frequency and the probability of offsite power recovery failure used in the PRA for the standard design envelopes the NAPS site-specific condition. Therefore, the PRA for the standard design is bounding.

Proposed COLA Revision

None