Attachment A: North Anna ESP Safety Inquiries

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			Site Characteristics	
1	SER 1-3	SER Section 1.2	What is the expected high water level of Lake Anna and how does it compare with the lakeside property line elevation of land owned by Lake Anna residents?	The Staff's safety evaluations are based on conservative assumptions. Therefore, the Staff estimate of the highest flood water elevation is 270 ft MSL, (NUREG-1835, p. 2-84). This value is not the expected annual maximum high water elevation because under the current operating rules for the Lake Anna Dam, which are regulated by the State of Virginia, releases are generally performed to maintain a water surface elevation of 250 ft MSL (SER p. 2-65). The expected normal high water lake level is 255 feet and Virginia Power and ODEC own the land that forms Lake Anna to a level of 255 ft (Applicant's Environmental Report ("ER"), p. 137). Author: George Wunder SME: Goutam Bagchi Key documents: NUREG-1835, "Safety Evaluation Report for an Early Site Permit (ESP) at the North Anna ESP Site," September 2005 ("NUREG-1835")
2	SER 2-3	SER Section 2.1.1.3	Does the Applicant, Dominion Nuclear North Anna LLC, currently have any right, title, or interest in the proposed ESP site?	No. The ESP site is owned by Virginia Electric and Power Company (Virginia Power) and Old Dominion Electric Cooperative ("ODEC"). Virginia Power and ODEC are direct and indirect wholly-owned subsidiaries, respectively, of Dominion Resources, Inc. ("DRI"). Dominion Nuclear North Anna is an indirect wholly-owned subsidiary of DRI. Author: George Wunder SME: George Wunder Key documents: n/a
3	SER 2-4, 2-5, 2-6	SER Section 2.1.2.1	The Applicant appears to have no authority and control over the exclusion area. The Applicant states that it will "purchase or lease the site from Virginia Power and ODEC" and goes on to predict what the terms of the lease will	In proposed Permit Condition 1, the Staff recommends requiring that approvals called for by state law for agreements providing for shared control of the ESP exclusion area be obtained before construction of any future nuclear plant. Author: George Wunder SME: George Wunder Key documents: NUREG-1835

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			provide. What arrangements or documentation do you have with the current owner of the ESP and NAPS sites that it will agree?	
4	SER 2-8	Application Section 2.1.3.1, ACRS March 2005 Transcript	The ACRS has criticized NRC for failing to incorporate changing knowledge into meteorological calculations, such as considering global warming in the projection of severe storms. Is this general criticism not also appropriate for population predictions where an aging population's desire for a rural environment and a desire to be near a lake could be strongly influencing factors that alter population growth?	The population estimates used in the FSER make use of current indicators of population growth consisting of data such as births and deaths, licensed drivers, tax returns, housing units and permits, auto registrations, increased number of schools and enrollments, and civilian group quarter populations. These estimates are performed every 5 or 10 years, depending upon the indicator, so as to capture the most recent population variations. These estimates potentially include the movement of aging people desiring to move to a rural environment and to be near a water body. The estimate of transient populations would also potentially include the variation in movement of such types of population groups. Therefore no special consideration is warranted. In Section 2.1.3, “Population Distribution,” of the FSER, the Staff evaluated the proposed site against the criterion in Regulatory Position C.4 of Regulatory Guide (“RG”) 4.7, ”General Site Suitability Criteria for Nuclear Power Stations,” Revision 2 (1998), regarding the need to consider alternative sites with lower population densities. This criterion specifies that if the population density in the vicinity of the proposed site projected at the time of initial site approval and within about 5 years thereafter, were to exceed 500 persons per square mile averaged over any radial distance out to 20 miles, then alternative sites should be considered. RG 4.7 states that the information needed to evaluate potential sites at this initial stage of site selection is assumed to be limited to information that is obtainable from published reports, public records, public and private agencies, and individuals knowledgeable about the locality of a potential site. RG 4.7 further states that projected changes in population within about 5 years after initial site approval should be evaluated for the proposed site. Further, population growth in the site vicinity after initial site approval is normal and expected and will be periodically factored into the emergency plan for the site. The Staff determined in the FSER that population densities for the proposed site would be well below 500 persons per square mile averaged over any radial

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				distance out to 20 miles. As part of its response to this question, the Staff re-evaluated population densities for years 2000, 2010, and 2020 assuming 2007 as the time of initial site approval and to about 5 years thereafter (2012). The densities were 117, 153, and 188 persons per square mile for years 2000, 2010, and 2020, respectively, and were still well below (by a factor of more than 2.6) the population density criterion specified in RG 4.7. The Staff is of the view that this margin suffices to bound the uncertainty associated with the future population growth model, method, and prediction used. In its re-evaluation, the Staff applied the county population data to the 20 mile radial grid line. If a county were bisected by the radial grid lines, the population data were proportioned by percent county falling in each radial grid estimated by the Staff. The Staff assumed that population is fairly uniform throughout each bisected county. <u>Author</u>: Seshagiri Tammara <u>SME</u>: Seshagiri Tammara, Jay Lee <u>Key Documents</u>: NUREG-1835, RG 4.7, Application Section 2.1.3.1, and ACRS March 2005 Transcript
5	SER 2-8, Application 2-2-5	Application Section 2.1.3.1	Growth projections 60 years into the future appear primarily based upon year 2000 census numbers and a standard future growth model. It is important to have reasonably accurate numbers for future populations to evaluate population dose calculations and emergency plan evacuation times.	
			A. Given the importance of this information, why	Historical growth rates that consist of school enrollments, automobile registrations, property registrations and others are generally included in the US

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			were not alternate methods for estimating growth explored? Historical growth could be determined from growth rates in school enrollments, growth rates in automobile registrations, or increases in property registrations. Why has little attention generally been paid to historical growth rates?	Census Bureau population estimates every 5 and/or 10 years, as previously addressed in Board Question 4. The constant annual growth rate based on decennial growth is applied to project the population into the future. However, the new annual growth rate based on next decennial growth would be reasonable to apply for the future projections from the next decennial census. The US Census Bureau ("USCB"), based on its 2005 American Community Survey, estimated Virginia total household population in 2005 to be 7,332,608. In addition, the USCB also performed interim projections of the total population for the United States and all States for the years 2005, 2010, 2015, 2020, 2025, and 2030 based on 2000 census data. The USCB projected population for Virginia for the year 2005 (based on 2000 census data) as 7,552,581, compared to the USCB-estimated population for Virginia for the same year 2005 (based on American Community Survey data) of 7,332,608. The projected population is about 3% higher than that of the estimated value. The USCB projected populations for Virginia for the years 2010, 2020, and 2030 are proportioned by the ratio of area within 50-miles of NAPS to the total land area of Virginia state, to estimate equivalent USCB projected population within 50-miles of NAPS. The Staff compared the Applicant's calculated population projected values with the USCB projected population values for 2010, 2020, and 2030 and found that the Applicant's calculated values are conservatively higher by a factor of 16%. The Staff also estimated population projections using each county growth rate from the USCB population data for the years 1990 and 2000 covering the counties within 50 miles of NAPS. The Staff's calculated projected population numbers for the same years of 2010, 2020, and 2030 are about 1% different from the Applicant's estimated projected values. The above methodologies, however, are reasonable in making the population projections. Due to significant uncertainty in projecting the population into the future, no method is viable other than applying enough conservatism in making the projections. The population projections would be periodically reviewed and adjusted, and if appropriate, the dose calculations and evacuation

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				times would be updated accordingly. Author: Seshagiri Tammara SME: Seshagiri Tammara Key Documents: NUREG-1835; Application Section 2.1.3.1
			B. If the proposed modeling method for population growth works, why has the Applicant not demonstrated this by taking census data from around the 1940s and showing that the present population is predicted?	Since each decennial census estimate generally includes the most recent historical growth rates relating to various population types (such as those addressed in Board Question 4 and Question 5A), it is appropriate, for modeling purposes, to use the latest decennial population data in order to utilize recent growth patterns around the proposed ESP site (see the Applicant's response to Board Question 5B) in reasonably projecting population for the next 60 years. Because different population types are reflected in 1940 census data, as compared to 2000 census data, modeling using 1940 data would not validate the model used in the Dominion Application or the Staff's SER. Author: Seshagiri Tammara SME: Seshagiri Tammara Key Documents: NUREG-1835; Application Section 2.1.3.1
			C. Given the long period of extrapolation for population growth, shouldn't some effort be made to establish error bars for future growth predictions?	Using the most recent decennial US Census Bureau population data, the Staff has determined that the calculation of the population projections based on consideration of computed annual growth rate by either each census bloc or county assumption would not vary more than 10-15%. Moreover, this margin of error in the estimation of the population projections has neither health and safety nor regulatory significance. Author: Seshagiri Tammara SME: Seshagiri Tammara Key Documents: NUREG-1835; Application Section 2.1.3.1
6		Application Section 2.1.3.4	The growth predictions in this section seem counter-intuitive in that the percentage growth rates decrease with	The annual growth rate is calculated in both the application and the FSER using USCB decennial census data for the years 1990 to 2000, and this calculated annual growth rate is constantly applied from the base year 2000 to future years covering 2010, 2020, 2030, 2040, and 2065. Therefore, the incremental total population per year from the year 2000 is constant. Estimating the ten-year

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			increasing time. The ten-year growth rate averages 3.5%/yr. from 2000 through 2010 but drops to 1.7%/yr. between 2030 and 2040. This trend continues, dropping to about 1.4%/yr. for the period 2040-2065. Has this behavior been exhibited in any past periods? What explanation can be offered for a decreasing future growth rate?	growth rate from the consecutive tenth base year would result in erroneous population growth trends. Author: Seshagiri Tammara SME: Seshagiri Tammara Key Documents: NUREG-1835; Application Section 2.1.3.4
7	SER 2-17	SER 2.2.3.1	Couldn't the 8,500 gallon gasoline truck or equivalent make delivery closer than 1.5 miles? What about deliveries to the plant?	Based on the guidance provided in RG 1.91, the critical distance calculated that could produce a peak overpressure of 1 psi due to an explosion of a tank truck carrying 8500 gal of gasoline, having an estimated equivalent to 50,700 lb of TNT, would be 1900 ft. The closest point of the nearest highway is 6420 ft (1.5 mi) and therefore, the Staff concluded that no significant damage would occur due to an explosion resulting from a gasoline truck accident on nearest highway. However, for deliveries less than 1.5 mi and greater than the calculated critical distance of 1900 ft, the damage would still be insignificant. The tank truck explosion hazard was evaluated on the basis of potential overpressure as it relates to distance from the plant/site. Specifically, RG 1.91 guidance was used to estimate the minimum distance (i.e., 1900 feet) at which the potential overpressure would not exceed approximately 1 psi. Another approach that may be used to address the specific "local delivery" aspect of tank truck hazards is to consider the likelihood of an accident leading to a significant overpressure. Taking into consideration typical tank truck accident rates and spill frequencies, the likelihood of an accident leading to significant overpressure is judged to be low. For example, the typical truck accident rate for tank trucks carrying flammable liquids is about 4.96×10^{-7} accidents per mile ("Comparative Risks of Hazardous Materials and Non-

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				hazardous Materials Truck Shipment Accidents/Incidents", Final Report, prepared for the Federal Motor Carrier Safety Administration, Battelle, March, 2001, www.fmcsa.dot.gov/documents/hazmatriskfinalreport.pdf) (visited January 25, 2007), and the probability of a release in the event of an accident is 0.086 (American Institute of Chemical Engineers Guidelines for Chemical Transportation Risk Analysis (1995)). Conservatively, the probability of ignition, given a release, may be assumed to be 1. Considering local deliveries within 1900 feet, the delivery rate that would be required in order for the likelihood of an accident exceeding 1 psi pressure to be greater than 10^{-6} per year is estimated to be about 65 deliveries per year. According to the Applicant, gasoline deliveries to the site are expected to be, on average, 3 per year. However, the maximum expected rate of deliveries to the site, including both gasoline and diesel, is approximately 8 per year. Author: Seshagiri Tammara SME: Seshagiri Tammara Key documents: RG 1.91, "Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants," Revision 1 (February 1978)(ADAMS accession number ML003740286)
8	SER 2-14, 2-17	SER Section 2.2.1.1-2.2.3.1	The SER states that there are train tracks 5.5 miles away from the site; a train could create a far larger explosion than a tractor trailer on the interstate. Does the extra half mile beyond the 5-mile radius of interest mean this risk should not be considered at all?	Pursuant to the guidance provided in Section 2.2.1-2.2.2 of Review Standard RS-002, all identified facilities and activities within 8 kilometers (5 mi) of the plant site should be reviewed. Facilities and activities at greater distances should be considered if they otherwise have the potential for affecting safety-related features of a nuclear power plant or plants that might be constructed on the proposed site. The distance from nearby railroad lines is checked to determine if a nuclear power plant or plants that might be constructed on the proposed site is within the range of a "rocketing" tank car assumed to be at a maximum distance of 500 meters (1640 ft). The 5-mile screening criterion is based on the consideration of potential overpressures that may be generated from explosions of rail cars carrying explosive materials. Based on RG 1.91, the critical distance calculated that could produce a peak overpressure of 1 psi due to explosion of a single railroad box car having a maximum probable explosive cargo of 132,000 lb would be about 2295 ft. The railroad distance of 5.5 mi from NAPS is much greater than

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				the critical distance of 2295 ft determined using the guidance in RG 1.91. Therefore, no detailed review of potential railroad accidents was performed. Author: Seshagiri Tammara SME: Seshagiri Tammara Key documents: RS-002, “Processing Applications for Early Site Permits” (May 2004); RG 1.91
9	SER 2-19	SER Section 2.2.3.3	The SER states that the Staff “independently reviewed possible hazards posed by the existing NAPS units.” Please describe what hazards the Staff reviewed and the results of the Staff’s review.	The Staff did not identify any specific hazards with respect to the existing NAPS units that would require technical evaluation. However, there are some potential hazards that may require specific evaluation at the CP or COL phase, for the reasons described below. One type of hazard that the existing NAPS units may present to a new plant is with respect to potential turbine missiles. Low pressure turbine wheel fragments may pose a significant risk to safety-related plant structures, systems and components. However, their trajectories are highly directional, typically confined to within $\pm 15^{\circ}$ of the plane of the wheels. Hence, appropriate placement and/or orientation of plant structures is an effective and relatively easily implemented means of protection. The Staff will perform a review of the turbine missile hazard at the CP or COL stage. Other typical hazards associated with the existing NAPS units are potential toxic or radioactive releases. The Applicant addressed these with respect to the existing units. However, this evaluation requires information regarding control room habitability system features, which are not available at the ESP stage. Hence, these types of hazards will be addressed by the Staff at the CP or COL stage. Author: Seshagiri Tammara SME: Seshagiri Tammara Key documents: n/a
10	SER 2-26	SER Section 2.3.1.1	The potential for freezing in the UHS water storage facility is apparently measured through the	Thickness of ice in a body of water is estimated by using cumulative degree-days. The Staff used Assur's method to independently estimate ice thickness of 17.1 inches (NUREG-1835, p. 2-105). If the future reactor design to be located at the proposed site requires a water-cooled UHS, the Applicant has proposed to

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			number of degree-days below freezing. Why is this a relevant parameter to establish either rate of freezing or a volume of ice?	use an underground storage facility. Using cumulative degree-days, the Staff can calculate the maximum ice thickness, and provide sufficient water volume in the UHS storage tank by adding the volume of ice to the required volume of liquid water for the UHS. This is why the cumulative degree-days has been identified as a site characteristic (NUREG-1835, Page A-18). <u>Author</u>: Goutam Bagchi <u>SME</u>: Goutam Bagchi <u>Key documents</u>: NUREG-1835
11	SER 2-27	SER Table 2.3.1-6	Considering flow requirements and the evaporative losses from cooling towers in UHS systems, and the design requirements of providing cooling water for normal operation, anticipated operational occurrences, safe shutdown, cooldown (first 30 days) and long term cooling for periods in excess of 30 days during adverse natural conditions, please explain why this doesn't [] rule out the use of wet cooling towers for UHS system. Doesn't this look like a situation for dry cooling or the need to qualify Lake Anna for supplying the necessary water?	If the future reactor design to be located at the proposed site requires a water cooled UHS, the Applicant has proposed to use an underground storage facility, and a totally dedicated closed cooling tower for the UHS. There are no interconnections or inter-reliance between normal and emergency cooling systems. (See NUREG 1835, Supplement 1, Page 2-4, penultimate paragraph.) The wet cooling tower proposed for the hybrid cooling tower for Unit 3 is to be used only for normal operation under normal and adverse natural conditions. Therefore, there is no need for qualifying Lake Anna for additional water demand, beyond that accounted for by the Applicant. <u>Author</u>: Goutam Bagchi <u>SME</u>: Goutam Bagchi <u>Key documents</u>: NUREG-1835
12		SSER 1 Section A-1,	This permit condition specifies the use of dry	Permit Condition 3 is intended for the proposed Unit 4. The Applicant chose to use dry cooling for the normal operation of Unit 4. The Staff need not consider,

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		Permit Condition 3.	cooling during normal operation for a fourth proposed unit. Since the ESP specifies an option of partial evaporative cooling for Unit 3 but only dry cooling for Unit 4, is this Permit Condition really intended for Unit 3? If water flow conditions allow the use of evaporative cooling, why wouldn't this be a preferred mode of operation since plant efficiency is improved?	as part of its safety review, the reasons for the Applicant's selection of dry cooling for proposed Unit 4. <u>Author</u>: Goutam Bagchi <u>SME</u>: Goutam Bagchi <u>Key documents</u>: n/a
13	SER 2-27	SER Table 2.3.1-6	Please describe the rationale, criteria, and procedures used in the preparation of Table 2.3.1-6, "Applicant's Proposed Ultimate Heat Sink Meteorological Site Characteristics."	According to the ESP application, the Applicant selected a mechanical draft cooling tower over a buried water storage basin or other passive water storage facility as the ultimate heat sink ("UHS"). In accordance with the guidance provided in RG 1.27, the Applicant selected a set of meteorological design parameters (i.e., maximum wet-bulb temperatures and coincident dry-bulb temperatures) and critical time periods (i.e., 1-day, 5-days, and 30-days) applicable to the design of a mechanical draft cooling tower UHS at the North Anna ESP site to ensure that a 30-day supply of water will be available and that the design-basis temperatures of safety-related equipment will not be exceeded. The Applicant determined the resulting site characteristic values using 1978–2003 meteorological data from Richmond, Virginia. In its FSER with open items, the Staff identified, in Open Item 2.3-3, the need for an additional UHS meteorological site characteristic for use in evaluating the potential for water freezing in the UHS water storage facility, a phenomenon that would reduce the amount of water available for use by the UHS. The Applicant responded to Open Item 2.3-3 by letter dated March 3, 2005 (ADAMS accession number ML050620205) by proposing the use of "maximum cumulative degree-day below freezing" as the relevant site characteristic. The Applicant determined the resulting site characteristic value by reviewing 1949–2001 daily mean dry-

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				bulb temperature data from Piedmont, Virginia. The Staff reviewed the UHS meteorological site characteristic values proposed by the Applicant and found them acceptable based on the discussion presented in FSER Sections 2.3.1.3 and 2.4.7.3. Author: R. Brad Harvey SME: R. Brad Harvey, Goutam Bagchi Key Documents: SSAR Section 2.3.1.3.8 and Table 2.3.1-6; FSAR Sections 2.3.1.3 and 2.4.7.3; RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants (for Comment)," Revision 2 (January 1976) (ADAMS accession number ML003739969); Letter to NRC Staff dated March 3, 2005 (ADAMS accession number ML050620205).
14	Application 2-2-40	Application Section 2.3.1.3.2	The probability of a tornado strike with rotational wind speeds of 260 mph is cited as only 1 x 10^{-7} but the general probability of a tornado strike is considerably higher: 6 x 10^{-5}. Can these higher probability tornados produce consequential damage at a plant site? Please provide evidence to confirm this response.	The site characteristic tornado is based on an exceedance frequency of 10^{-7} per year, which is the same criterion used to define the design basis tornado (DBT) in RG 1.76 and DG-1143 (proposed Revision 1 to RG 1.76). RG 1.117 describes a method for identifying those SSCs of light-water-cooled reactors that should be protected from the effects of the DBT and remain functional. RG 1.117 states that SSCs important to safety that should be protected from the effects of a DBT are: (1) those necessary to ensure the integrity of the reactor coolant pressure boundary; (2) those necessary to ensure the capacity to shut down the reactor and maintain it in a safe shutdown condition; and (3) those whose failure could lead to radioactive releases resulting in offsite exposures greater than 25% of the guideline exposures of 10 CFR Part 100. The higher-probability tornados will have less severe characteristics (e.g., lower maximum wind speeds, pressure drops, and rates of pressure drop) than the DBT. These higher-probability tornados may produce consequential damage to a plant site (e.g., they could damage unprotected radioactive waste systems) but the resulting offsite exposures are expected to remain well below the guideline exposures of 10 CFR Part 100. Author: R. Brad Harvey SME: R. Brad Harvey Key Documents: SSAR Section 2.3.1.3.2; RG 1.76, "Design Basis Tornado for Nuclear Power Plants," April 1974 (ADAMS accession number ML003740273);

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				RG 1.117, “Tornado Design Classification,” Revision 1 (June 1976) (ADAMS accession number ML003739346); DG-1143, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants” (Proposed Revision 1 of Regulatory Guide 1.76) (January 2006) (ADAMS accession number ML053140225)
15	SER 2-31	SER Section 2.3.1.3	Data from NSSL on tornado frequencies is quoted in units of "days per year for a tornado threat within 25 miles.” This would appear to be a reasonably meaningless parameter. Does a value of .05 mean that there is one chance in 20 per year of a tornado with the reference wind speed being within 25 miles of the plant? If these numbers can be considered to be tornado probabilities, then how do these numbers relate to the much lower tornado frequencies referenced above?	A value of 0.05 for the “mean number of days per year with the threat of significant tornados (F2 or greater on the Fujita Scale) occurring within 25 miles of the North Anna ESP site” (Ref. 1) means that there is one chance in 20 per year of having a day where one or more significant tornados occurs within 25 miles in any direction from the North Anna ESP site (Ref. 2). In contrast, an annual strike probability of 6×10^{-5} (Ref. 3) means that there is one chance in 16,667 per year of any given point in the vicinity of the North Anna ESP site being struck by a tornado (Ref. 4). The first number is intended to estimate the probability per year of a tornado occurring within a specified area (i.e., within 25 miles surrounding the North Anna ESP site) whereas the second number is intended to estimate the probability per year that a given location (such as a reactor building) will be struck by a tornado. Author: R. Brad Harvey SME: R. Brad Harvey Key Documents: (1) NUREG-1835, Section 2.3.1.3; (2) National Climatic Data Center (NCDC), "Severe Thunderstorm Climatology, Total Threat," National Severe Storms Laboratory, August 29, 2003, <http://www.nssl.noaa.aov/hazard/totalthreat.html> (November 30, 2004). (ADAMS Accession No. ML043380034); (3) SSAR Section 2.3.1.3.2; (4) Thom, H. C. S. Tornado Probabilities, Monthly Weather Review, 1963, Vol. 91, Nos. 10–12, 730–736.
16	SER 2-34	SER Table 2.3.1-7	What is the effect of including the Staff’s proposed regional climatic site characteristics as ESP site characteristics in Appendix A.3? Don’t these characteristics simply describe the site	10 CFR 52.17(a)(1)(vi) states that the ESP application should describe the meteorological characteristics of the proposed site and that these site characteristics must comply with 10 CFR Part 100. 10 CFR 100(c)(2) states that the factors to be considered when evaluating sites should include the meteorological characteristics of the site that may have an impact upon plant design such as maximum probable wind speed and precipitation. 10 CFR 52.79(a)(1) states that if a COL application references an ESP, it should contain information sufficient to demonstrate that the design of the facility falls within the parameters (e.g., site characteristics) specified in the ESP.

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			climate? What is the effect if the list of climate characteristics is incorrect, or needs to be updated at the time of any COL application? If the COL application occurs 20 years after the ESP is issued, is the intervening 20 years of meteorological data to be ignored?	Long-term environmental changes and changes to the region resulting from human or natural causes may introduce changes to the site characteristics that could be relevant to the design basis. If, in the future, the ESP site is no longer in compliance with the terms and conditions of the ESP (e.g., new information shows that the climatic site characteristics no longer represent extreme weather conditions due to climate change), the Staff may seek to modify the ESP or impose requirements on the site in accordance with the provisions of 10 CFR 52.39(a)(1), provided the change is necessary to provide adequate protection of the public health and safety or the common defense and security, or to bring the permit or the site into compliance with the Commission's regulations and orders applicable and in effect at the time the permit was issued. In the alternative, the ESP may be revised as a result of a successful contention or petition raised pursuant to 10 CFR 52.39(a)(2)(i), (ii), or (iii). Author: R. Brad Harvey SME: R. Brad Harvey Key Documents: NUREG-1835, Table 2.3.1-7; NUREG-1835, Supplement 1, "Safety Evaluation for an Early Site Permit (ESP) at the North Anna ESP Site" (November 2006) ("NUREG-1835, Supplement 1"); Appendix A.3; 10 CFR Part 52
17	SER 2-46, Application 2-2-61	SER Section 2.3.4, Application 2.3.4.2	χ/Q values for different accident exposure intervals were calculated by taking a yearly average χ/Q and employing a logarithmic interpolation to obtain values for shorter exposure intervals such as 2 hours, 8 hours, 72 hours, etc. See RG 1.111. While this may be a reasonable approach, it does not necessarily represent the highest	The following represents a combined response to Board Questions 17, 20, and 21. The short-term (accident) χ/Q values presented as site characteristics in FSER Section 2.3.4 have not been presented with error limits (or with a range of possible values with associated exceedance probabilities) because such information is not essential for performing design-basis accident dose assessments for the SSAR. Nonetheless, Table 1 presents a range of EAB and LPZ accident χ/Q values with associated exceedance probabilities in response to the Board's inquiries. The following discussion describes the process used to select the site characteristic EAB and LPZ accident χ/Q values used in the SSAR design-basis accident dose assessments. The accident χ/Q values presented as site characteristics in FSER Tables 2.3.4-1 and 2.3.4-2 were generated by the Applicant in accordance with the guidance

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			values of χ/Q. Why shouldn't error limits be ascribed to χ/Q to confirm that higher values are possible? In lieu of error limits, why not cite probabilities for true values lying below the quoted values?	presented in RG 1.145 as implemented by the PAVAN computer code (NUREG/CR-2858). The PAVAN computer program uses meteorological data in the form of joint frequency distributions of hourly averages of wind direction and wind speed by atmospheric stability class. For each of the 16 downwind direction sectors (i.e., N, NNE, NE, ENE, etc), PAVAN calculates χ/Q values for each combination of wind speed and atmospheric stability at the appropriate downwind distance (i.e., EAB and outer boundary of the LPZ). The χ/Q values calculated for each sector are then ordered from greatest to smallest and an associated cumulative frequency distribution is derived based on the frequency distribution of wind speed and stabilities for that sector. The smallest χ/Q value in the distribution will have a corresponding cumulative frequency equal to the wind direction frequency for that sector. PAVAN then determines for each sector an upper envelope curve based on these data (plotted as χ/Q versus probability of being exceeded) such that no plotted point is above the curve. From this upper envelope, the χ/Q value, which is equaled or exceeded 0.5% of the total time, is obtained. The maximum 0.5% χ/Q value from the 16 sectors becomes the "maximum sector χ/Q value" pursuant to Section C.2.1.1 of RG 1.145. Using the same approach, PAVAN also combines all χ/Q values independent of wind direction into a cumulative frequency distribution for the entire site. An upper envelope curve is then determined and the program selects the χ/Q value which is equaled or exceeded 5.0% of the total time. This is known as the "5% overall site χ/Q value." The larger of the two χ/Q values, either the 0.5% maximum sector value or the 5% overall site value, is used to represent the χ/Q value for a 0–2 hour time interval. Note that this resulting χ/Q value is based on 1-hour averaged data but is conservatively assumed to apply for 2 hours. For the EAB, the 0.5% maximum sector χ/Q value (2.26×10^{-4} sec/m^3) occurred in the SE downwind sector and was higher than the 5% overall site χ/Q value (1.59×10^{-4} sec/m^3). The EAB 0.5% maximum sector χ/Q value is equivalent to an overall site χ/Q value which is equaled or exceeded approximately 2.3% of the total time. For the outer boundary of the LPZ, the 0.5% maximum sector χ/Q value (4.63×10^{-5} sec/m^3) occurred in the ESE downwind sector and was higher than the 5%

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				overall site χ/Q value (2.77×10^{-5} sec/m^3). The LPZ 0.5% maximum sector χ/Q value is equivalent to an overall site χ/Q value which is equaled or exceeded approximately 2.1% of the total time. For determining longer period χ/Q values at the outer boundary of the LPZ, PAVAN performs a logarithmic interpolation between the 2-hour χ/Q value and the annual average (8760-hour) χ/Q value for each of the 16 sectors. In addition, the 5% overall site χ/Q value is used along with the maximum of the 16 annual average χ/Q values to also determine the χ/Q values for the intermediate time periods by logarithmic interpolation. For each time period, the highest among the 16 sector and overall site χ/Q values is identified and becomes the short-term (accident) site characteristic χ/Q value for that time period. Note that the EAB and LPZ χ/Q values used in SSAR and FSER design-basis accident assessments are sometimes referred to as "5%" χ/Q values in comparison to the "50%" EAB and LPZ χ/Q values used in ER and EIS design-basis accident assessments (see the Staff's response to Board Question 107). Author: R. Brad Harvey SME: R. Brad Harvey Key Documents: SSAR Section 2.3.4; NUREG-1835, Section 2.3.4; RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1 (November 1982) (Reissued February 1983 to correct page 1.145-7)(ADAMS accession number ML003740205); NUREG/CR-2858 (PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations); Table 1 (attached hereto).
18	SER 2-48		Table 2.3.4-1 provides the χ/Q values "@ EAB" and "@ LPZ." The former is a specific location - the boundary. The latter is an area - the zone within a 6 mile radius. Please explain	The "χ/Q values @ LPZ" listed in FSER Tables 2.3.4-1 and 2.3.4-2 represent χ/Q values at the outer boundary of the LPZ. Author: R. Brad Harvey SME: R. Brad Harvey Key Documents: NUREG-1835, Table 2.3.4-1

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			whether all LPZ values are the average for the LPZ or are at its outer boundary.	
19	Application 2-2-45	Application 2.3.1.3.6	The frequency of lightning strikes at the plant site appears to have been obtained by determining the annual lightning strikes over a larger area and scaling these numbers to a site area of 0.068 square miles. Isn't this overly simplistic? Doesn't lightning occur between points of appropriate electrical potential which can be influenced by building height and conductivity to ground?	Information provided by the Applicant on the number of lightning flashes per year at the plant site was not considered essential to the Staff's review of the acceptability of the site. Lightning protection system (LPS) design for nuclear power plants does not consider the frequency of lightning strikes at the plant site but rather how big (voltage & current) one strike might be. The LPS is necessary whether there are several strikes or just one. Where multiple strikes become relevant is in the context of maintenance of the LPS. Guidelines for the design and installation of lightning protection systems at nuclear power plants are discussed in RG 1.204. Author: R. Brad Harvey SME: Christina Antonescu, R. Brad Harvey Key Documents: SSAR Section 2.3.1.3.6; NUREG-1835, Section 2.3.1.3; RG 1.204, "Guidelines for Lightning Protection of Nuclear Power Plants" (November 2005) (ADAMS accession number ML052290422)
20	Application 2-2-61	Application 2.3.4.2	Measured wind directions and velocities were combined to generate χ/Q values at specific locations. The bounding case was apparently a wind direction and velocity with a probability of greater than 0.5% and the highest calculated χ/Q. Since the bounding case does not reflect the highest value of χ/Q that	See the Staff's response to Board Question 17. Author: R. Brad Harvey SME: R. Brad Harvey Key Documents: SSAR Section 2.3.4; NUREG-1835, Section 2.3.4; RG 1.145; NUREG/CR-2858 (PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations)

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			is possible for a given site, shouldn't the calculated χ/Q values in the Application and the SER carry error limits that better reflect the true values that are possible?	
21	Application 2-2-61	Application 2.3.4.3	Bounding χ/Q values for different release intervals were apparently obtained by calculating yearly average χ/Q values and using a logarithmic extrapolation to obtain values for shorter release times. As cited above, this approach may be reasonable but it does not represent the highest possible χ/Q values for a given accident exposure duration. The scientific community deals with this type of problem by including error limits for calculated values when higher values are possible. Why shouldn't this also be done in a regulatory environment?	See the Staff's response to Board Question 17. Author: R. Brad Harvey SME: R. Brad Harvey Key Documents: SSAR Section 2.3.4; NUREG-1835, Section 2.3.4; RG 1.145; NUREG/CR-2858
22	SER 2-51	SER Section 1.2.5	Why is it acceptable to exclude the known, normal releases from Units 1 and 2 from a calculation of population	Under 10 CFR Part 52 and 10 CFR Part 100, there are no requirements to consider incremental population doses from co-located operating plants and proposed future units. Doses to individual members of the public associated with effluent releases from existing operating reactors are addressed by NRC regulations and license conditions under 10 CFR Part 50 and 10 CFR Part 20 in

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			doses during normal operation? An answer that simply says this is consistent with regulatory policy is not regarded as acceptable.	controlling and monitoring radioactive effluents. The regulatory requirements include: a. effluent radionuclide concentration limits of Appendix B, Table 2 to 10 CFR Part 20; b. dose limits to members of the public under 10 CFR Sections 20.1301 and 20.1302; c. EPA's environmental radiation standards of 40 CFR Part 190, implemented under 10 CFR Section 20.1301(e); and d. ALARA dose objectives of Appendix I to 10 CFR Part 50. These requirements are implemented through operational programs and procedures mandated by 10 CFR 50.36a, 10 CFR 50.34a, and Section IV of Appendix I to Part 50. The key operational program documents are the Radiological Effluent Technical Specifications (RETS) or Standard Radiological Effluent Controls (SREC), Offsite Dose Calculation Manual (ODCM), and the Radiological Environmental Monitoring Program (REMP). Under 40 CFR Part 190, compliance with dose limits is assessed against the entire site and all sources of radioactivity and external radiation, regardless of the number of power plants. The sources of radioactivity include all liquid and gaseous effluent releases. Compliance is assessed considering the whole site, and not on the basis of individual plants. The implementation of these programs and license conditions is routinely inspected by NRC Regional inspectors. These inspections examine the licensee's radiological effluent monitoring and release programs to ensure that the programs meet all NRC requirements and license conditions. If a plant were to exceed the dose limits of 40 CFR Part 190 or any other requirements of 10 CFR Parts 50 and 20, the inspection would identify the cause and determine whether a proper response and corrective actions had been, or were being, taken by the licensee. Under the provisions of 10 CFR 20.1301(f) and Section IV.C of Appendix I to 10 CFR Part 50, the NRC may impose additional restrictions after evaluating the impacts on members of the public in light of commitments and characterizations contained in the COL application. Such

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				restrictions may address, among other things, limits on effluent releases for the purpose of reducing collective population doses. <u>Author</u>: Jean-Claude Dehmel <u>SME</u>: Jean-Claude Dehmel <u>Key Documents</u>: 10 CFR Part 20 (10 CFR 20.1301 and 10 CFR 20.1302), 10 CFR Part 50, 10 CFR Part 52, 10 CFR Part 100, and 40 CFR Part 190
23	Application 2-2-59	Application 2.3.4.1	Why is there no discussion of the effect of possible inversions that could trap radioactive materials near the ground and increase χ/Q values?	The Applicant generated its short-term (accident) site characteristic χ/Q values in accordance with the guidance presented in RG 1.145 as implemented by the PAVAN computer code (NUREG/CR-2858). PAVAN models the ground-level based inversion conditions that result in the 0–2 hour EAB χ/Q value and the 0–2 hour LPZ χ/Q value that is used to obtain LPZ X/Q values for longer exposure intervals. (The Staff's response to Board Question 17 describes how the 0–2 hour EAB and LPZ accident χ/Q values are selected). The Applicant derived its EAB and LPZ accident χ/Q values by conservatively assuming a ground-level release. By definition, Pasquill stability classes F and G represent ground-level-based inversion conditions (RG 1.23; DG-1164). These moderately and extremely stable atmospheric conditions generally result in the highest predicted χ/Q values for ground-level releases (high χ/Q values indicate poor dispersion conditions which result in higher radiation doses) and occurred approximately 11.7% of the time during the three-year period of onsite meteorological data used as input to PAVAN. The 0–2 hour EAB and LPZ X/Q values (which are exceeded approximately 2.1–2.3% of the time, as discussed in the Staff's response to Board Question 17) typically occur during these worst case ground-level-based inversion conditions. <u>Author</u>: R. Brad Harvey <u>SME</u>: R. Brad Harvey <u>Key Documents</u>: SSAR Section 2.3.4; RG 1.23; DG-1164; RG 1.145; NUREG/CR-2858
24	SER 2-52	SER Section 2.3.5.1	Please provide a regulatory or other authoritative definition of the following terms:	The Applicant generated its long-term (routine release) site characteristics presented in SSAR Section 2.3.5 in accordance with the guidance presented in RG 1.111 as implemented by the XOQDOQ computer code (NUREG/CR-2919). Section C.3 of RG 1.111 states that radioactive decay and dry deposition should

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<tr><td></td><td></td><td></td><td>“undepleted/no decay,” "undepleted/2.26 decay," and "depleted/8[.]00 decay."</td><td>be considered in radiological impact evaluations of potential annual radiation doses to the public resulting from routine releases of radioactive materials in gaseous effluents. Section C.3.a of RG 1.111 states that an overall half-life of 2.26 days is acceptable for evaluating the radioactive decay of short-lived noble gases and an overall half-life of 8 days is acceptable for evaluating the radioactive decay for all iodines released to the atmosphere.

The following definitions for the χ/Q categories listed in the headings of FSER Table 2.3.5-1 are based on the terminology used in the output of the XOQDOQ computer code:

• “Undepleted/No Decay” χ/Q values are atmospheric dispersion factors used to evaluate ground level concentrations of long-lived noble gases, tritium, and carbon 14. The plume is assumed to travel downwind without undergoing dry deposition or radioactive decay.

• “Undepleted/2.26-Day Decay” χ/Q values are atmospheric dispersion factors used to evaluate ground level concentrations of short-lived noble gases. The plume is assumed to travel downwind without undergoing dry deposition but is decayed assuming a half-life of 2.26 days, based on the half-life of Xe-133m.

• “Depleted/8.00-Day Decay” χ/Q values are atmospheric dispersion factors used to evaluate ground level concentrations of radioiodine and particulates. The plume is assumed to travel downwind with dry deposition and is decayed assuming a half-life of 8.00 days, based on the half-life of I-131.

Additional details are provided in the Staff’s response to Board Question 28.

Author: R. Brad Harvey
SME: R. Brad Harvey, Jean-Claude Dehmel
Key Documents: NUREG-1835, Table 2.3.5-1; RG 1.111; NUREG/CR-2919, “XOQDOQ: Computer Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations”</td></tr>
<tr><td>25</td><td>SSER A-18</td><td>SSER Appendix A</td><td>Why is the D/Q for the nearest vegetable</td><td>The value in the FSER is correct. Due to a typographical error, the supplement to the draft EIS had a vegetable garden value of 6.0 x $10^{-8}/m^2$. A comment was</td></tr>
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			garden on A-18 of the [SER] Supplement 6 x 10^{-9} while the comparable value in Table 1-1 on 1-2 of the Draft EIS appears to be a factor of ten different at 6 x 10^{-8}?	received on the supplement to the draft EIS (see Volume 2, p. 4-44 of the final EIS) identifying the correct value, and it was changed in the final EIS to 6.0 x $10^{-9}/m^2$. The calculations for the supplement to the draft EIS used the correct value of 6.0 x $10^{-9}/m^2$. Author: Greg Stoetzel SME: Greg Stoetzel Key Documents: NUREG-1835, Supplement 1, Appendix A.3; NUREG-1811, Supplement 1, "Draft Environmental Impact Statement for an Early Site Permit (ESP) at the North Anna ESP Site" (July 2006), Table 1-1.
26	SER 2-53	SER Table 2.3.5-1	The Applicant states that no milk exposure pathway for isotope ingestion was considered because no cows or goats used for milk consumption were found adjacent to the plant. Given that milk is a high exposure transport path for some isotopes and the fact that the Applicant is trying to look ahead for a period of up to 60 years, shouldn't this exposure pathway be evaluated?	The application states that there is no milk production from cows or goats within a 5-mile radius from the North Anna Site; see ER Tables 2.7-13 and 5.4-9. This conclusion is based on the results of an annual land-use census conducted to identify potential new exposure pathways and dose receptors that would warrant modifications to operational programs required under 10 CFR 50.36a, 10 CFR 50.34a, and Section IV of Appendix I to 10 CFR Part 50. The 5-mile radius is based on NRC guidance, NUREG-1301 (PWR plants), NUREG-1302 (BWR plants), and Radiological Assessment Branch Technical Position (Ref. a - c). The guidance states that the milk exposure pathway must be included in dose calculations when milk is known to be produced within a 5-mile radius. If no milk is produced within that radius, the vegetable exposure pathway must be substituted for it as the pathway with the next most radiological significance for exposure to radioiodines. As such, the vegetable exposure pathway was evaluated in connection with this ESP application. In assessing doses to nearby populations, the applicant has included milk as an exposure pathway in estimating collective doses within a 50-mile radius of the site. An estimate of regional milk production rate is given in ER Table 5.4-3. If new exposure pathways and dose receptors were identified in evaluating the results of the land use census, the scope of the Radiological Environmental Monitoring Program ("REMP") would be modified and similar revisions would be made to the Offsite Dose Calculation Manual ("ODCM") for the purpose of adding new exposure pathways and dose receptors. It should be noted that both of these operational documents are required before fuel loading and reactor startup.

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				The Staff recognizes the possibility that over time, new exposure pathways may be identified and the Staff therefore included COL Action Item 11.1-1, which states: "A COL or CP Applicant should verify that the calculated radiological doses to members of the public from radioactive gaseous and liquid effluents for any facility to be built on the North Anna site are bounded by the radiological doses included in the ESP application and reviewed by the NRC." Appendix A (p. A-8) to the SER (NUREG-1835), Supplement 1, dated November 2006. At the CP or COL stage, the Staff will review updated details on site characteristics, including updated information on offsite exposure pathways and dose receptors, and actual distances from effluent discharge points to offsite dose receptors. Such detailed information will provide the means for the Staff to evaluate whether the Applicant is using information from the most current land-use census, and whether such new information warrants a re-assessment of the radiological impacts associated with effluent releases. The fact that the milk exposure pathway was not considered in the application is addressed in Appendix A (p. A-18) to the FSER (Supplement 1) describing site characteristics. Finally, this issue is also discussed in the FEIS (NUREG-1811), Section 5.9.2.2, p. 5-61. Supporting references a. NUREG-1301, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors," April 1991. The use of this guidance is mandated by NRC Generic Letter 89-01, which is included in this NUREG as Appendix C. b. NUREG-1302, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors," April 1991. The use of this guidance is mandated by NRC Generic Letter 89-01, which is included in this NUREG as Appendix C. c. Radiological Assessment Branch Technical Position, Rev. 1, November 1979. This document is included in NUREG-1301 and NUREG-1302 as Appendix A. Author: Jean-Claude Dehmel

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				SME: Jean-Claude Dehmel Key Documents: 10 CFR 50.36a, 10 CFR 50.34a, Section IV of Appendix I to 10 CFR Part 50, NUREG-1301 (PWR plants), NUREG-1302 (BWR plants), Radiological Assessment Branch Technical Position, NUREG-1835 and NUREG-1811
27	SER 2-55		Legal Question: The SER states that any COL or CP Applicant referencing the SER dispersion calculations for routine releases "should verify that the specific release point characteristics, specific locations of receptors of interest used to generate the ESP routine release atmospheric dispersion site characteristics bound the actual values provided at the COL or CP stage" and makes this COL Action Item 2.3-3. The SER also states that this will be a site characteristic in any ESP. What happens if, at the COL stage, the release point characteristics or locations of receptors are not as specified in the ESP? Would a contention at the COL stage, alleging that the actual values are	See the "NRC Staff Legal Brief in Response to Licensing Board's Safety-Related Questions"

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			different from those used at the ESP stage, be admissible?	
28	SER 2-56	SER Table 2.3.5-2 and Application Table 2.7-14 thru 20	These tables give χ/Q and D/Q values for normal and accident conditions for different locations. The table of isotopes upon which these calculations appear to be based contain fission products only. Why weren't Co-58, Co-60, Mn-54 and the other activation products that exist outside the reactor fuel included in these calculations?	The gaseous effluent source term and its associated doses at the stated receptor locations are based on radionuclides listed in Table 1.3-8 (p.2-1-44) of the SSAR, and Table 5.4-7 (p. 3-5-142) of the ER. This list includes Co-58, Co-60, Mn-54, and other isotopes. The information presents estimates of radioactivity released (curies per year) and corresponding effluent concentration levels (microcuries per ml) by radionuclides. For gaseous effluents, 70 radionuclides are identified as comprising the source term. The total activity comprises about 14,000 curies of noble gases, about 3,500 curies of tritium, and about 24 curies of fission and activation products per plant. The information presented in SER Table 2.3.5.2 and related tabulations in the Application reflect a nomenclature used in defining atmospheric dispersion parameters. As is noted in response to Board Question No. 24, the adjusted atmospheric dispersion parameters refer to groupings by radionuclides half-lives with potentially significant radiological implications in terms of doses to offsite receptors. The 8.0 and 2.26 day decay-corrected atmospheric dispersion parameters are based on the half-lives of I-131 and Xe-133m, respectively. In dose calculations (GASPAR II Code, Ref. a), the 8.0 and 2.26 day half-lives are used to derive an effective transient time of the release from its point of discharge to the location of each stated receptor. Among various input parameters, the GASPAR II code requires that the location, wind sector, distance, and dispersion parameters be specified for each dose receptor. The three dispersion factors are no decay and no plume depletion, decay and no plume depletion, and decay and plume depletion. This information is used to derive an effective transient time of the plume. In turn, the transient time is used to apply a decay correction factor for each radionuclide assumed to be present in the source term, given its radiological decay constant. This approach applies a decay correction for the travel time of each radionuclide from the point of release to the point of exposure. Supporting reference: a. NUREG/CR-4653, "GASPAR II - Technical Reference and User Guide," March 1987. Section 3.1.1, pp. 3.2 - 3.6.

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				Author: Jean-Claude Dehmel SME: Jean-Claude Dehmel Key Documents: SSAR, ER, NUREG-1835
29	SER 2-56	SER Table 2.3.5-2 and Application Table 2.7-14 thru 20	Tritium is an isotope that is both produced external to the fuel and capable of diffusing through the fuel cladding. Test wells at nuclear plants can show relatively high concentrations of tritium. Why is there no mention of this isotope in either the environmental or dose sections of the SER or Application?	Tritium is included as one of several radionuclides in both liquid and gaseous effluents. Tritium is listed in Tables 1.3-7 (p. 2-1-41) and 1.3-8 (p. 2-1-44) of the SSAR, Tables 5.4-6 (p. 3-5-139) and 5.4-7 (p. 3-5-142) of the ER, and Tables H-2 (p. H-4) and H-5 (p. H-7) of the FEIS. In assessing the radiological impacts from liquid and gaseous effluents, the dose results are presented as aggregate, meaning that the reported doses include the dose contribution from all radionuclides comprising the source term for either liquid or gaseous effluents. As such, the dose results presented in the application, SER, and FEIS do not provide any information on the dose contribution due to tritium alone or for any other individual radionuclide. NRC regulations (10 CFR Sections 20.1301 and 20.1302, and 10 CFR Part 50, Appendix I dose objectives) do not specify doses on a per radionuclide basis. It is only in the context of liquid and gaseous effluent concentrations that 10 CFR Part 20 (see Table 2 to Appendix B) states limits by specific radionuclides. The application states that all radioactive liquid effluent discharges will be released to the discharge canal, which empties into the Waste Heat Treatment Facility. Regarding the presence of tritium in test wells, e.g., in groundwater, the Applicant provided information on the scope of the Radiological Environmental Monitoring Program (REMP). The REMP is summarily described in ER Sections 6.2 (p. 3-6-6) and 6.7 (p. 3-6-38). The scope of the REMP includes the collection of groundwater samples, as indicated in ER Table 6.7-2 (p. 3-6-39). The existing REMP will be used for the operation of the proposed plants, and updated using information from the most current land-use census as to whether new information warrants a re-assessment of the radiological impacts associated with liquid effluent releases. Additionally, should plant-derived radioactivity be detected in a groundwater pathway following construction and operation of the proposed plants, the results of groundwater samples would be evaluated as to its source, cause, and extent of the contamination in determining whether the conditions of the license and/or NRC regulations have been violated. Author: Jean-Claude Dehmel

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				<u>SME</u>: Jean-Claude Dehmel <u>Key Documents</u>: 10 CFR Part 20, 10 CFR Part 50
30	SER 2-56	SER Table 2.3.5-2	The section on seismic impacts in the Application and SER presents detailed information on calculations, tests, and measurements—even to the extent of including field notes in the Application. In contrast, almost no information is given relative to the assumptions used for calculation of X/Q or dose. For the case of "normal" plant releases:	
			A. What percentage of failed fuel was assumed for the reactor core?	No specific percentage of failed fuel was assumed for the reactor core. For the ESP review, the Staff was asked to review the Applicant's data using a Plant Parameter Envelope (“PPE”) concept. The PPE does not contain detailed information on fuel characteristics and performance. The presence and concentrations of specific radionuclides in effluents are influenced by many factors, including production mechanisms in fuel pellets, transport from within fuel assemblies to the primary coolant and primary steam, type of reactor coolant chemistry used, leakage rates through various system components, effectiveness of primary coolant purification systems, and effectiveness of waste treatment systems. The estimate of radioactive materials ultimately available in primary coolant or primary steam is characterized as an aggregate without making specific distinctions about the various competing mechanisms that influence the movement and speciation of radioactivity through various primary coolant and process treatment systems. While the Staff recognizes that there are complex processes that have effects on

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				generation and removal of fission and activation products, the methods used to derive radioactivity levels in primary coolant and primary steam are based on relationships that reflect the operational experience of the industry for both BWR and PWR plants. The difficulty lies in major differences in the types of available technologies that may be used to treat effluents, and the fact that no single treatment technology can adequately represent such variety, nor can standard operating practices. On the other hand, all licensed plants must meet the same regulatory criteria, and, consequently, the responsibility rests on COL applicants to describe the type of technology that will be used to meet NRC regulatory requirements and license conditions. This approach is exemplified by the guidance of Regulatory Guide 1.112, and methods described in NUREG-0016 and NUREG-0017, and ANSI/ANS-18.1-1999 (Ref. a - d). The guidance presents acceptable methods for calculating annual average expected releases of radioactive materials in liquid and gaseous effluents released by light-water-cooled nuclear power reactors. Finally, the Staff will have specific details, at the COL stage, on the Applicant's reactor design, radioactive waste processing systems, effectiveness of selected waste processing method, and concentrations of effluents. Such detailed information will provide the means for the Staff to evaluate the proposed effluent source terms and perform independent verifications of radioactivity levels. This issue is subsumed by COL Action Item 11.1-1. Author: Jean-Claude Dehmel SME: Jean-Claude Dehmel Key Documents: RG 1.112, NUREG-0016, NUREG-0017, ANSI/ANS-18.1-1999
			B. What coolant leakage through the steam generator was assumed for PWRs and what condenser leakage for BWRs?	No specific coolant leakage was assumed. See the Staff's response to Board Question 30A. Author: Jean-Claude Dehmel SME: Jean-Claude Dehmel Key Documents: RG 1.112, NUREG-0016, NUREG-0017, ANSI/ANS-18.1-1999
			C. What leakage rates were assumed for pumps and seals?	No specific leakage rates were assumed for pumps and seals. See the Staff's response to Board Question 30A. Author: Jean-Claude Dehmel

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				<u>SME</u>: Jean-Claude Dehmel <u>Key Documents</u>: RG 1.112, NUREG-0016, NUREG-0017, ANSI/ANS-18.1-1999
			D. What concentrations of activation products were assumed?	The assumed amounts of radioactivity discharged in effluents are presented in Tables 1.3-7 (p. 2-1-41) and 1.3-8 (p. 2-1-44) of the SSAR; and Tables 5.4-6 (p. 3-5-139) and 5.4-7 (p. 3-5-142) of the ER. This information is also repeated in Tables H-2 (p. H-4) and H-5 (p. H-7) of the FEIS. The information presents estimates of radioactivity released (curies per year) and corresponding effluent concentration levels (microcuries per ml) by radionuclides. For liquid effluents, 68 radionuclides are identified, and 70 are identified for gaseous effluents. For liquid effluents, the total activity comprises about 850 curies of tritium and about 0.37 curies of fission and activation products per plant. For gaseous effluents, the total activity comprises about 14,000 curies of noble gases, about 3,500 curies of tritium, and about 24 curies of fission and activation products per plant. <u>Author</u>: Jean-Claude Dehmel <u>SME</u>: Jean-Claude Dehmel <u>Key Documents</u>: NUREG-1811
			E. What release rates were assumed from the waste processing facilities at the plant?	In addition to the information presented in response to Board Question 30D. above on activity levels, the assumed discharge rate of liquid effluent is assumed to be about 100 gallons per minute mixed in a site-specific dilution flow rate of about 100,000 gallons per minute. The information is presented in ER Table 5.4-1 (p. 3-5-137). The application does not include any information describing gaseous effluent flow rates, such as volumetric flow rates and types and numbers of plant stacks and vents. The gaseous effluent source term represents an aggregate of all sources of radioactivity. Atmospheric dispersion parameters are used for the purpose of estimating gaseous effluent concentrations at the stated receptor offsite locations. This approach is acceptable to the Staff in the context of the PPE approach used in bounding specific facility features and characteristics. <u>Author</u>: Jean-Claude Dehmel <u>SME</u>: Jean-Claude Dehmel <u>Key Documents</u>: ER

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			Board Questions 30 A - E	Supporting references: a. Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," Rev. 0-R, May 1977. b. NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactor (BWR-Gale Code)" Rev. 1, January 1979. c. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactor (PWR-Gale Code)" Rev. 1, April 1985. d. ANSI/ANS-18.1-1999, " Radioactive Source Term for Normal Operation of Light Water Reactors," September 1999. Author: Jean-Claude Dehmel SME: Jean-Claude Dehmel Key Documents: RG 1.112, NUREG-0016, NUREG-0017, ANSI/ANS-18.1-1999
31	SER 2-56	SER Table 2.3.5-2	If the Applicant chooses a reactor type different from the AP1000 or ABWR reference designs, will they be held to the χ/Q values presented in the Application or will they be allowed to present new values?	The CP or COL Applicant has the option to recalculate the ESP χ/Q site characteristics at the CP or COL stage using actual plant design characteristics. Author: R. Brad Harvey SME: R. Brad Harvey Key Documents: NUREG-1835. Table 2.3.5-2; NUREG-1835, Supplement 1, Appendix A.3; 10 CFR Part 52
32	SER 2-59	SER Section 2.4.1.1	The non-safety related cooling water need for all four units is 121 cubic feet per second [cfs]. Wouldn't this value vary	Yes. The 121 cfs is a time-averaged value which includes both natural evaporation from the lake plus the forced evaporation associated with the heat dissipation systems. It should be noted that the 121 cfs value was determined before the Applicant proposed changes to the cooling water system. The revised value is somewhat lower.

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			significantly with time of year?	Author: Goutam Bagchi SME: Kenneth See Key Documents: North Anna Early Site Permit Application, Response to RAI No. 4, Serial No. 04, dated 5/12/2004, page 30, part e.
33	SER 2-60	SER Section 2.4.1.1	The SER states that "However, if the dry cooling tower system contains a secondary cooling water loop…" The above sentence seems to imply a make-up water need for the dry cooling tower on Unit 4. Do dry cooling towers require any make-up water?	No. Dry tower systems typically have no evaporative water losses and would have no continuous blowdown discharge. In the event that the cooling water loop uses an open sump configuration with a free surface, a small amount of evaporative loss, estimated to be about 1 gpm (0.002 cfs), will be expected to occur. Any make-up water necessary to replenish the small evaporative losses expected for Unit 4 could be obtained from Lake Anna. Author: Goutam Bagchi SME: Kenneth See Key Documents: Application
34	SSER 2-6	SSER Section 2.4.1.3	A natural evaporation rate from the lake was assumed to be 5.6 in./mo. Wouldn't this value vary significantly with season? Why is 5.6 the selected value?	Yes, evaporation rate does vary by season. According to the reference literature used by the Staff in its review (van der Leeden et al.), the mean-monthly reservoir evaporation rate at Richmond, Virginia varies from 1.5 in./mo. (December) up to 5.6 in./mo. (July). The value Staff selected was the largest natural evaporation rate for the month within the year. It should be noted that all components of this calculation were conservative estimates. For example, the Staff also assumed that all waste heat from the existing NAPS units was converted into evaporated water (38 cfs/unit or 76 cfs total). Assuming the lake was at its minimum surface area (elevation 242 ft MSL), the lake elevation drop rate resulting from the NAPS units alone was 5.61 in./mo. Therefore, Staff's computed value of 14.6 in./mo., resulting in a period of 49 days for the lake to drop from elevation 244 ft to 242 ft, is a conservative estimate. Author: Goutam Bagchi SME: Kenneth See Key Documents: van der Leeden, F., F.L. Troise, D.K., "The Water Encyclopedia: Second Edition," Lewis Publishers (1991).

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35	SSER 2-6	SER Section 2.4.1.3	The Staff estimates that lake level would drop only 2 feet in 49 days which reflects a balance between evaporative loss and new flow into the lake (not given for the calculation). Wouldn't this conclusion be strongly dependent on the time of year since water influx can vary significantly?	In order to produce a conservative estimate, any inflows to the lake were assumed to be zero for the entire period. This includes inflows from both tributaries and precipitation. The Staff's value of 49 days is conservative, and was computed to show a lower bound on the minimum number of days it could take the lake to drop from elevation 244 ft to 242 ft. In reality, the number of days would be significantly larger. This calculation was produced to support the statement, "the staff concludes that the water surface elevation in Lake Anna does not fall rapidly and that sufficient time will be available to plant operators before the low water surface elevation shutdown threshold is reached to plan a shutdown of the proposed Unit 3 without endangering its safety, even under severe drought conditions" (NUREG-1835, Supplement 1, p. 2-6). Author: Goutam Bagchi SME: Kenneth See Key Documents: SSER
36	SSER 2-6	SER Section 2.4.1.3	There does not appear to be any discussion of water leakage from the UHS into groundwater. It would seem possible that this could be a route for some transfer of radioactivity into the environment. (For example, tritium in BWR coolant transferring to the UHS through condenser leakage). Should this release path be considered in the SER?	UHS water is completely separate from the normal heat sink ("NHS"). That is to say, during emergency cooling, water from the UHS circulates through the shell side of the heat exchangers and the reactor coolant runs through the tubes inside. As such, the Staff does not envision any significant amount of radioactive contaminants migrating into the UHS reservoir. The Staff does not see the proposed path as likely to transfer radioactivity into the groundwater, and therefore did not consider this pathway in the FSER. Author: Goutam Bagchi SME: Goutam Bagchi Key Documents: n/a
37	SER 2-71	SER Table 2.4.2-1	Please explain exactly what the entry of the value 18.3 under PMP depth (in.) means?	This value refers to the estimated maximum amount of precipitation that could occur over an area of 1 mi^2 during a 1-hour time period at the plant site. Probable maximum precipitation ("PMP") estimates are based on physical limits of the amount of moisture sustainable by the most favorable atmospheric

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				conditions associated with a defined area. Author: Goutam Bagchi SME: Kenneth See Key Documents: NOAA Hydrometeorological Report No. 52, "Application of Probable Maximum Precipitation Estimates – United States East of the 105th Meridian" ("HMR-52").
38	SER 2-77 & 2-78	SER Tables 2.4.3-1	These tables list the PMP values for various size watersheds including North Anna for durations of 6 hour increments. Can the value of 18.2 (no units identified) in Table 2.4.3-2 be interpreted as 18.2 inches water depth accumulation in Lake Anna during the first 6 hours of the storm (an average rate of slightly over 3 inches per hour)? If not, what does it signify?	No. Table 2.4.3-2 gives the maximum amount of precipitation that could occur over an area of 343 mi^2 during the indicated durations at the plant location. Specifically, the value of 18.2 represents 18.2 inches of precipitation over an area of 343 mi^2 during a 6-hour period at the plant location. Author: Goutam Bagchi SME: Kenneth See Key Documents: HMR-52
39	SER 2-74	SER Section 2.4.3.1	Applicant investigated the historical storms used in a 1976 study and three additional storms that occurred in February 1979, March 1996, and June 1995. The additional storms were selected because they produced high water levels in Lake Anna.	According to data supplied by the Applicant the maximum water levels in Lake Anna during the periods in question were as follows: February 1979 251.6 ft msl June 1995 251.4 ft msl March 1996 250.3 ft msl Author: Goutam Bagchi SME: Goutam Bagchi Key Documents: Bechtel Calculation Package 2480G018AT19-S1_MIT

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			What were the high water levels in Lake Anna as a result of those storms?	
40	SER 2-81	SER Figures 2.4.3-2, 2.4.3-3	Comment- The Figures appear to be reversed.	Yes, the figures are reversed. Author: Goutam Bagchi SME: Kenneth See Key Documents: n/a
41	SER 2-85	SER Section 2.4.1.1	The last paragraph in this section states that the Applicant told the Staff that the UHS would consist of a mechanical draft cooling tower over an underground basin **if the selected plant design includes a UHS** (emphasis added). Is the UHS not confirmed for any of the steam generation plants? Under what conditions would a proposed plant need a UHS?	The need for a UHS is based on the design of the plant. Certain reactor types do not require a conventional UHS to provide safety-related cooling during emergency shutdown. For example, some advanced reactor designs use passive cooling and/or use onsite storage tanks, while others use boil-off to achieve cooling. The UHS has not been confirmed for any of the steam generation plants. Author: Goutam Bagchi SME: Kenneth See Key Documents: n/a
42	SER 2-89	SER Section 2.4.5.1	Applicant concluded that given the short fetch length, surges and waves produced from winds or oscillatory waves alone would not produce water heights greater than the still water level resulting from	The probable maximum surge (discussed in SRP 2.4.5, in SSAR Section 2.4.5 and in SER Section 2.4.5) is defined as an increase in still water level due to meteorologically-induced stress (atmospheric pressure reduction) during a probable maximum hurricane, probable maximum windstorm, or a moving squall line (ANSI/ANS-2.8-1992). The ESP site and Lake Anna are located sufficiently far from the nearest body of open water (the Chesapeake Bay) that a storm surge in that body of water would not affect the site. Lake Anna itself is not large enough to develop storm surge under the influence of a probable maximum hurricane. The Staff-estimated wind setup is 0.46 ft, which was added to the

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			the PMF. Water heights produced by PMFs are considerable and in any event, wouldn't the surges and waves produced by wind action be additive to the flood-caused high water level?	water surface elevation resulting from the PMF, and the seiche was estimated to be insignificant. As such, surges and waves produced by wind are additive to the flood-caused high-water level. Author: Goutam Bagchi SME: Kenneth See Key Documents: ANSI/ANS-2.8 – 1992
43	SER 2-115	SER Section 2.4.10.3	The Staff estimated local, intense precipitation for the ESP site be 18.3 inches/hr based on Table 2.4.2-1. This seems high. What is the basis for this number?	The value of 18.3 inches was taken from Figure 24 of HMR-52, p. 79. The value of 6.1 inches was obtained by multiplying the value of 18.3 inches by the ratio (approx 0.33) indicated in Figure 36 of HMR-52, p. 94. Author: Goutam Bagchi SME: Kenneth See Key Documents: HMR-52
44	SER 2-138	Table 2.4.14-1	Staff's values for local intense precipitation are shown as 18.3 in./hr and 6.1 in. in 5 minutes. Please identify the source of these data.	See the Staff's Response to Board Question 43. Author: Goutam Bagchi SME: Kenneth See Key Documents: HMR-52
45	SER 2-117	SER Section 2.4.11.1	Staff mentions that the existing units and the proposed units have different lake water levels for shutdown. Hasn't Applicant modified the intake of the existing units to provide for a 242' MSL threshold elevation for shutdown (the same elevation as proposed for the new units)?	Yes. As stated on page 2-64 of the FSER, the modifications were complete as of March 3, 2005. Author: Goutam Bagchi SME: Kenneth See Key Documents: NUREG-1835

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46	SER 2-121	SER Section 2.4.11.3	Staff reports that since the Applicant's minimum water surface elevation site characteristic is lower than the Staff's estimate, the Applicant's value is acceptable. Does this mean that if Applicant had proposed a level below 242', the Staff would have accepted that? What criteria did the Staff use in arriving at its decision? Was there any consideration of Lake Anna dock owners?	The Staff's assessment was focused on the safety issue of a frequent and sudden loss of cooling water for the NHS resulting in excessive reliance on the UHS. In its water budget calculation based on the historical period of operation of the NAPS/Lake Anna system, the Staff determined that the low water level that would require shutdown of the NHS would happen both rarely and gradually. In the water budget simulation, the level of Lake Anna never dropped below the proposed level operation threshold of 242 feet MSL. This does not suggest that Lake Anna could never reach the 242 level under extreme drought conditions. It does, however, establish that shutdowns of the NHS due to low water in Lake Anna would be rare. Impacts to dock owners were not considered in the safety analysis, since this is not a safety-related issue. <u>Author</u>: Goutam Bagchi <u>SME</u>: Hosung Ahn <u>Key Documents</u>: n/a As stated in Section 5.5.3.4 of the EIS, during times of severe drought, such as the drought of 2001 to 2002, the Staff expects that the usability of stationary boat docks may be impacted when the lake level drops below 76 m (248 feet) MSL. Lakefront houses temporarily could have mud-flat views instead of the preferred water views. However, the drought of 2000 to 2001 was a rare event, and none of these impacts from such an event would be considered permanent. <u>Author</u>: Dr. Michael Scott <u>SME</u>: Jack Cushing <u>Key Documents</u>: NUREG-1811, Volumes 1 and 2
47	SER 2-129	SER Section 2.4.13.1	The SER states that the Applicant provided a conceptual hydrological model of the subsurface environment and pathways for releases of liquid effluent to ground and surface waters from	The conceptual groundwater model of the subsurface flows and pathways at the proposed ESP site is described in the second half of Section 2.4.13.1 in NAPS UFSAR, Pages 2.4-15, as well as in the second half of Section 2.4.13.1 in NUREG-1835. The conceptual model mentioned is a description of conceptualized, simplified processes of groundwater flow and transport phenomena. The Applicant provided the conceptual model to the Staff on August 19, 2004 (ADAMS Accession number ML042440355).

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			the ESP site. Please provide it.	Author: Goutam Bagchi SME: Hosung Ahn Key Documents: Letter from E.S. Grecheck, Dominion, to U.S. NRC Document Control Desk, "Dominion Nuclear North Anna, LLC, North Anna Early Site Permit Application, Supplemental Response to Request for Additional Information No. 4," dated August 19, 2004 (ADAMS Accession number ML042440355).
48	SER 2-130, SER 2-59, SER 2.136	SER Section 2.4.13.1	The SER states that "no site-specific data are available to determine the chemical characteristics of ground water at the ESP site. The Applicant assumed that the water quality of the crystalline aquifers in the Piedmont Physiographic Province is representative of the water quality at the ESP site." Given that the NAPS industrial facility has been situated on this site for several decades, the assumption that its groundwater is as pure as background seems inappropriate. Please provide any data on the chemical or radiological characteristics of the soil, vadose zone, and groundwater (not just the aquifer) on and below the ESP site and portions of the NAPS site in the vicinity (within 600' of the	The Applicant stated in the NAPS UFSAR, Section 2.4.13.4, that a radiological environmental monitoring program is conducted. The preoperational program is described on the North Anna Environmental Report. While there is no site-specific data available to determine the chemical characteristics of groundwater at the site, the Applicant stated that a detailed numerical model will be developed as part of any COL application. Author: Goutam Bagchi SME: Hosung Ahn Key Documents: NAPS UFSAR Section 2.4.13.4

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			boundary) of the ESP site.	
49	SER 2-132	SER Section 2.4.13.3	Applicant reported that the only observation of piezometric head difference made between the saprolite and the bedrock indicates an upward hydraulic gradient. Please explain what this is and the conditions necessary for it to occur.	An "upward hydraulic gradient" means that the piezometric head at the bedrock is higher than that of upper saprolite layer. This is very common for a well-defined unconfined layer in which water is recharged from upland with a higher piezometric head than that of a surficial aquifer in lowland. Author: Goutam Bagchi SME: Hosung Ahn Key Documents: n/a
50	SER 2-136	SER Section 2.4.13.3	The SER states that "The staff concludes that, because of incomplete knowledge of the subsurface hydrological and chemical properties and the likely composition of the radwaste effluent itself, significant uncertainty exists in the characteristics of radionuclide migration in the subsurface at the ESP site at the time of ESP review. The Staff has determined that after the reactor design is selected and additional details related to radwaste tank design and the location within	

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			the proposed site is known, appropriate subsurface hydrological characterization can be completed." The Board has several questions relating to this passage, as follows:	
			A. What prevents the Applicant and Staff from developing more sufficient knowledge [data] on the "subsurface hydrological and chemical properties" at this time? Isn't this an appropriate part of the ESP assessment?	Without knowing the exact location and elevation of a likely point of accidental release, there is no way to draw liquid pathways above and below ground from the point of release to the accessible environment and thus to make a plan for field investigation to define the subsurface hydrological and chemical properties especially when the hydrogeologic properties are highly varied in space. Author: Goutam Bagchi SME: Hosung Ahn Key Documents: n/a
			B. What prevents the Applicant and Staff from developing a plant parameter envelope for the "likely composition of the radwaste effluent?" PPE assumptions were made for other liquid effluent releases, thus please explain why it was not done here.	At the ESP stage, the information on the quantity, quality, and timing of liquid effluents to be stored in the radwaste tanks is unknown. In any event, however, the Staff has recommended Permit Condition 4, which states, "The NRC Staff proposes to include a condition in any ESP that might be issued in connection with this application requiring that an Applicant referencing such an ESP design any new unit's radwaste systems with features to preclude any and all accidental releases of radionuclides into any potential liquid pathway." Author: Goutam Bagchi SME: Hosung Ahn Key Documents: NUREG-1835, Section 2.4.1.3; Appendix A, Table A-1
			C. Absent a baseline delineating the existing chemical and radiological contamination on the site, what measures will	The onsite radiological environmental program contained in the Offsite Dose Calculation Manual (ODCM) can identify any existing radiological contamination from Virginia Power's Units 1 and 2, if any. This or a similar program will also be applied to the proposed units when they are in operation. In addition, results from the REMP could potentially be used to differentiate contamination based on

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			be taken to distinguish between any existing contamination from Virginia Power's Units 1 and 2 and Dominion's proposed Units 3 and 4?	the operational status of the units. This issue will be clarified at the CP or COL stage. Author: Goutam Bagchi SME: Hosung Ahn, Jean-Claude Dehmel Key Documents: n/a
			D. Legal Question: Absent the foregoing information, should an ESP be granted? How does this comport with the Commission's statement that "where adequate information is not available, early site permits will not be issued?" 54 Fed. Reg. 15372, 15378 (April 18, 1989).	See the "NRC Staff Legal Brief in Response to Licensing Board's Safety-Related Questions"
51	SER 2-136	SER Section 2.4.13.3	The Staff proposed permit condition 4 would require the permit holder to "design any new unit's radwaste systems with such features to preclude any and all accidental releases of radionuclides into any potential liquid pathway." Isn't this impossible? Please explain how you would interpret and implement such a requirement?	Permit Condition 4 requires preclusion of accidental spillage of liquid radwaste effluents by engineered features. This is achievable by design. In order to implement this permit condition, the selected radwaste system design should be robust, and it should incorporate features that would hold any accidental spillage in a seismically-designed building. Author: Goutam Bagchi SME: Hosung Ahn Key Documents: n/a

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52	SER 2-146, SER 2-166	SER Section 2.5.1.1.1, 2.5.1.3.2	Please provide a regulatory or other authoritative definition of "capable tectonic source."	The NRC Staff uses 10 CFR 100.23 to evaluate the suitability of a proposed site from a geologic and seismic standpoint, including the determination of the Safe Shutdown Earthquake ("SSE") ground motion. An acceptable method for meeting the requirements of 10 CFR 100.23 is provided in Regulatory Guide ("RG") 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion." RG 1.165 provides guidance on a number of different procedures that together satisfy the requirements of 10 CFR 100.23 for determining the SSE. Appendix A to RG 1.165 provides a detailed definition of a "capable tectonic source," as follows: Capable Tectonic Source - A capable tectonic source is a tectonic structure that can generate both vibratory ground motion and tectonic surface deformation such as faulting or folding at or near the earth's surface in the present seismotectonic regime. It is described by at least one of the following characteristics: 1. Presence of surface or near-surface deformation of landforms or geologic deposits of a recurring nature within the last approximately 500,000 years or at least once in the last approximately 50,000 years. 2. A reasonable association with one or more moderate to large earthquakes or sustained earthquake activity that are usually accompanied by significant surface deformation. 3. A structural association with a capable tectonic source having characteristics of either items 1 or 2 in this paragraph such that movement on one could be reasonably expected to be accompanied by movement on the other. In some cases, the geological evidence of past activity at or near the ground surface along a potential capable tectonic source may be obscured at a particular site. This might occur, for example, at a site having a deep overburden. For these cases, evidence may exist elsewhere along the structure from which an evaluation of its

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				characteristics in the vicinity of the site can be reasonably based. Such evidence is to be used in determining whether the structure is a capable tectonic source within this definition. Notwithstanding the foregoing paragraphs, the association of a structure with geological structures that are at least pre-Quaternary [older than 1.8 million years], such as many of those found in the Central and Eastern regions of the United States, in the absence of conflicting evidence will demonstrate that the structure is not a capable tectonic source within this definition. Currently operating nuclear power plants are licensed to Appendix A to 10 CFR Part 100, rather than 10 CFR 100.23. The definition of a capable fault in Appendix A is very similar to that provided in RG 1.165, except that 35,000 years is specified in Appendix A, rather than 50,000 years for the first characteristic described above. Author: Clifford Munson SME: Clifford Munson Key Documents: 10 CFR 100.23, RG 1.165
53	SER 2-148 to 2-161	SER Section 2.5.1.1.1	The Applicant and Staff reject a number of geological hypotheses, including Weem's tectonic origin for the seven local fault lines (2-148, 2-164) and Marple and Talwani's research regarding the existence of central and northern segments of the East Coast Fault System (2-161). What are the consequences to the safety of the plant if any or all of these rejected	As described in the SER, the Staff evaluated each of the potential seismic sources in the region surrounding the site. This evaluation focused primarily on the potential Quaternary features identified by the Applicant, including the seven fall lines defined by Weems and the East Coast Fault System described by Marple and Talwani. Although the Staff and the Applicant both concluded that these features are either nontectonic or pre-Quaternary (and therefore not capable), they are covered by the Applicant's probabilistic seismic hazard analysis ("PSHA") for the ESP site. Rather than characterizing the seismic potential of each identified geologic feature, PSHA studies for the Central and Eastern United States ("CEUS") generally define broad areal seismic source zones. This approach is used because there is uncertainty about the underlying causes of earthquakes throughout much of the CEUS. This uncertainty is due to a lack of active surface faulting, a low rate of seismic activity, and a short historical record. In addition to low rates of seismic activity, earthquakes in the CEUS generally occur over diffuse areas, which does not allow for a clear association between seismicity and known geologic features.

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			hypotheses are correct?	To account for this uncertainty related to seismic sources in the CEUS, the Applicant used the Electric Power Research Institute ("EPRI") PSHA methodology, which was published in 1989. To characterize the seismic sources in the CEUS, EPRI formed six independent Earth Science Teams ("ESTs"). Each of the ESTs characterized the seismic sources in the CEUS in terms of magnitude, location, and recurrence. For example, Figure 2.5-19 in the SSAR shows the seismic sources for the Bechtel Group, one of the six ESTs, for the region surrounding the site. The major sources zones defined by this EST are Central Virginia, Southern Appalachian and Atlantic Coastal Region, which covers the entire coast. As stated in RG 1.165, the EPRI CEUS seismic source characterizations have been previously reviewed and accepted by the Staff. The Staff reviewed the Applicant's geologic description in SSAR Sections 2.5.1 and 2.5.2 to determine whether the EPRI CEUS seismic source characterizations needed to be updated. Both the seven fall lines, described by Weems, and the East Coast Fault System, identified by Marple and Talwani, represent potential updates to the EPRI CEUS seismic sources. For the case of the seven fall lines, the Staff concurred with the Applicant's determination that these fall lines are nontectonic features and, therefore, there was no need to update the EPRI source zones. For the East Coast Fault System, the Applicant included this potential source as part of its updating of the source zones for the EPRI PSHA. In summary, due to the uncertainties with characterizing the seismic potential of specific geologic features, PSHAs for the CEUS generally define broad areal source zones in terms of magnitude, recurrence, and location. Although the Staff has previously accepted the EPRI PSHA sources for the CEUS, potential local and regional seismic sources are reviewed in detail to ensure that the previously defined areal source zones adequately characterize the potential of these individual geologic features. <u>Author</u>: Clifford Munson <u>SME</u>: Clifford Munson <u>Key Documents</u>: NUREG-1835, RG 1.165, SSAR Sections 2.5.1 and 2.5.2
54	SER 2-166	SER	The SER states that in	The hazard for faults located near the surface is due to both the seismic waves

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	to 2-167	Section 2.5.1.3.2	1974 the Staff concluded that “unnamed fault ‘a’ is not a capable tectonic source." In the SER the Staff concludes that the Applicant has “adequately investigated the possible extension of fault “a” and that “the ESP site has no fault displacement potential.” What does this mean? Please state and explain the Staff’s current conclusion or opinion as to whether unnamed fault “a” is a “capable tectonic source.”	generated by an earthquake and from the potential for fault surface rupture or displacement. The Staff concluded that no capable faults exist within the site area (including unnamed fault “a”) and, therefore, there is no potential for fault surface deformation. As part of its review of the ESP application, the Staff reviewed the Applicant’s description of the site area structural geology. Structural features at and within a 5-mile radius of the site consist of a series of northeast-striking faults and folds within the metamorphic bedrock. The fault closest to the ESP site is the unnamed fault “a” that traverses the site and was the subject of intensive studies following its exposure within the excavations for abandoned Units 3 and 4. Investigation of this fault by Virginia Power, the NRC Staff, and their respective consultants in the early 1970's revealed that it has a length of about 3000 feet and that the most recent movement on the fault occurred at least one to two million years ago. These investigations included detailed mapping of the excavation fault exposures, three fault trenches, interpretation of aerial photography, and a detailed soil profile analysis. Based on its own investigations and review of the Applicant’s submittals, the NRC Staff concluded that the unnamed fault “a” is not capable, as defined by Appendix A of 10 CFR Part 100. In 1974, the AEC issued Supplement No. 3 to the Safety Evaluation for Units 3 and 4, which concluded that the fault is not capable. During its review of the North Anna ESP application, the Staff confirmed that its earlier conclusions remain valid. This review included an examination of the local area seismic activity and more recent geologic maps. Specifically, the Staff plotted the locations of all earthquakes in the central Virginia area and confirmed that there has not been an increase in seismic activity or any earthquakes near unnamed fault “a”. The Staff also asked the Applicant to clarify the length of unnamed fault “a” since Louis Pavlides of the U.S. Geological Survey (“USGS”) extended this fault farther north and south of the North Anna site. The Applicant stated that Pavlides, who is deceased, did not provide an explanation for extending fault “a” beyond the ESP site. The Applicant also stated that Pavlides did not map any offset stratigraphic contacts in the Lake Anna area to support his mapped location of the fault. The Applicant’s extensive geologic field reconnaissance in support of its ESP application showed no scarps (vertical offsets) or lineaments along the extended trace of fault “a” as mapped by Pavlides. In order to thoroughly evaluate the Applicant’s characterization of fault

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				"a" and the Applicant's other geologic investigations, the Staff sought the assistance of the USGS. The Staff and its USGS advisors visited the ESP site to examine local geologic features and met with the Applicant to discuss the interpretations, assumptions, and conclusions presented by the Applicant concerning potential geologic faulting near the North Anna site. Based on its evaluation, the Staff confirmed that its earlier conclusions concerning the capability of fault "a" are valid. For final confirmation of this conclusion, the Staff proposes to impose a Permit Condition (Permit Condition 7) on the ESP holder, requiring that the Staff be notified no later than 30 days after any site excavations for safety-related structures are opened in order to examine and evaluate all geologic features. <u>Author</u>: Clifford Munson <u>SME</u>: Clifford Munson <u>Key Documents</u>: Appendix A of 10 CFR Part 100
55	SER 2-168	SER Section 2.5.1.4	The SER states that "These results provide an adequate basis to conclude that no capable tectonic faults exist in the plant area (5 mi) that have the potential to cause near-surface displacement." Does the Staff so conclude? Or is this merely a statement that, given these results, such a conclusion is possible?	The conclusion, as quoted in Board Question 55, is the Staff's conclusion based on its review of the geologic and seismologic information submitted by the Applicant in SSAR Section 2.5.1 and its review of the Applicant's responses to Requests for Additional Information ("RAIs") and Open Items ("OIs"). <u>Author</u>: Clifford Munson <u>SME</u>: Clifford Munson <u>Key Documents</u>: SSAR Section 2.5.1
56	SER 2-174+	SER Section 2.5.2.1.6	The March 2005 ACRS testimony notes that the site Safe Shutdown Earthquake exceeds the design SSE at high frequencies for the	The large high-frequency SSE ground motions are acknowledged by the Applicant in SSAR Sections 2.5.2.6.9 and 2.5.2.6.10, in which the Applicant describes potential modifications of the SSE to achieve a more "realistic" design spectrum, which it terms an engineering design spectrum ("EDS"). The Staff's evaluation of these potential modifications is presented in SER Section 2.5.2.3.6 on page 2-201:

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			designs that have been certified to date. No mention of this issue occurs in the subsections that deal with seismic issues. What is the significance and current status of this issue?	In SSAR Sections 2.5.2.6.9 and 2.5.2.6.10, the applicant alluded to future modifications of the site SSE spectrum in order to obtain an engineering design spectrum (EDS) that represents “the proper input into the large nuclear power plant structures.” The applicant stated that the ESP site SSE is not suitable for the design of the SSCs of nuclear power plants because of high spectral accelerations in the high-frequency range (about 15 to 30 Hz). According to the applicant, the EDS would take into account plant-specific structural characterizations and local site conditions, as well as the ESP SSE spectrum. However, the ESP application does not include the EDS because the applicant has not selected a specific reactor design. The applicant proposed to include the EDS as part of a COL application. Because the applicant did not provide any specific recommendations or procedures for developing the EDS, the staff cannot evaluate the merits of the proposed approach. For ESP reviews, the Staff focuses its review on the adequacy of the SSE as determined from the Applicant’s characterization of regional and local seismic sources as well as the local soil and rock properties. The regulations do not require the Staff to compare the site SSE with design response spectra for advanced reactor designs, which may or may not be selected by the Applicant. Rather, the Staff will defer these comparisons to its review of CP or COL applications. As part of its revision and development of regulatory guides and updating of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” the Staff has held several public meetings with nuclear industry representatives to discuss seismic issues, including the treatment of high-frequency ground accelerations. In particular, the Staff and industry have focused on the interface between seismic hazards, described in SRP Section 2.5, and the seismic design of nuclear power plant structures, systems, and components (“SSCs”), described in SRP Section 3.7. The minutes from the latest public meeting (12/14/06) between the Staff and Industry on these issues can be found in ADAMS (Accession number ML0636104870).

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				Author: Clifford Munson SME: Clifford Munson Key Documents: SSAR Section 2.5.2.6, NUREG-0800
57	SER 2-177	SER Section 2.5.2.1.6	The SER states that the determination of a safe shutdown earthquake (SSE) for the site uses a "reference probability." Probability of what?	SER Section 2.5.2.1.6 on page 2-177 describes the reference probability approach for determining the SSE (emphasis added): The method for determining the SSE for a site, as described in RG 1.165, is based on the use of a reference probability. The basis for the procedure in RG 1.165, as well as the determination of the reference probability, is that existing nuclear power plants do not represent an undue risk to the health and safety of the public. As such, using the existing plants as a reference, RG 1.165 recommends a procedure to determine the seismic design basis for future plants. **The reference probability is the average probability of exceeding the SSE ground motion at 5 Hz and 10 Hz, using either the 1993 LLNL PSHA or the 1989 EPRI PSHA.** The NRC staff calculated a reference probability level for 29 nuclear power plant sites in the CEUS; the median reference probability for these 29 sites, using median hazard results, is 10^{-5} per year. A similar value was obtained using both the 1993 LLNL and the 1989 EPRI PSHA level; therefore, RG 1.165 endorses both the LLNL and EPRI PSHA results as suitable for seismic hazard estimation for future siting. Appendix B to RG 1.165 provides a list of the 29 plants used to determine the reference probability as well as a detailed description of the approach. The NRC Staff and industry determined the annual probability of exceedance from either the Lawrence Livermore National Laboratory ("LLNL") or EPRI probabilistic seismic hazard analysis ("PSHA") hazard curves for the 5 and 10 Hz SSE spectral acceleration values for each of the 29 CEUS plants. An example of this is shown in Figure B.1 in Appendix B of RG 1.165. Figure B.2 shows the reference probabilities for each of the sites and that the median value using the LLNL hazard estimates is 1×10^{-5}. The reference probability (1×10^{-5}) is used to determine the controlling earthquakes for the site. The controlling earthquakes are determined in terms of a magnitude and distance for both a low-frequency and distant earthquake and a

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				high-frequency local earthquake. This procedure is described in Appendix C of RG 1.165. The controlling earthquakes are subsequently used to determine the SSE. <u>Author</u>: Clifford Munson <u>SME</u>: Clifford Munson <u>Key Documents</u>: NUREG-1835, Section 2.5.2.1.6, RG 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion " (March 1997) (ADAMS accession number ML003740084)
58	SER 2-177	SER Section 2.5.2.1.6	The SER states that the Staff "calculated a reference probability level for the 29 nuclear power plant sites in the CEUS; the median reference probability for these 29 sites, using median hazard results, is 10^{-5} per year." Please provide the results of these calculations.	The results of the Staff's calculations are provided in Appendix B of Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion." Figure B.2 shows the reference probability values for each of the 29 sites and that the median value using the Lawrence Livermore National Laboratory (LLNL) hazard estimates is 1 x 10^{-5}. <u>Author</u>: Clifford Munson <u>SME</u>: Clifford Munson <u>Key Documents</u>: RG 1.165
59	SER 2-177	SER Section 2.5.2.1.6	Please explain whether there have been any advances in seismic science or data relative to the safety of nuclear power plants since the 29 CEUS reactors were originally sited several decades ago and why it is appropriate to automatically use the median probability from those sites as the	The method for determining the SSE for a site, as described in RG 1.165, is based on the use of a reference probability. The basis for the procedure in RG 1.165, as well as the determination of the reference probability, is that existing nuclear power plants do not represent an undue risk to the health and safety of the public from a seismic standpoint. As such, using the existing plants as a reference, RG 1.165 recommends a procedure to determine the seismic design basis for future plants. The reference probability approach, as described in RG 1.165, was developed in the early 1990s as a method for determining the SSE for future reactor sites. Using the 1993 LLNL and the 1989 EPRI PSHA results, the Staff calculated a median reference probability, as described in Appendix B of RG 1.165, of 1x10^{-5}. Several advances have been made since the early 1990s in the geologic

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			benchmark for safety on an ESP that might be issued in 2007 and apply to reactors built perhaps in 2027 or even 2047.	sciences with regard to the seismic source characterizations for several important CEUS sources such as the New Madrid, MO and Charleston, SC seismic sources. In addition, several new earthquake ground motion models have been developed for the CEUS. Recognizing that such advances and updates would occur, RG 1.165 recommends updating the reference probability: In general, major recomputations of the LLNL and EPRI data base are planned periodically (approximately every ten years), or when there is an important finding or occurrence. The overall revision of the data base will also require a reexamination of the reference probability discussed in Appendix B. However, since a new approach for determining the SSE, described as the performance-based approach, has been developed, EPRI has elected not to re-compute reference probabilities for each of the 29 CEUS sites, nor has the Staff. The Staff's review of the performance-based approach is described in its SER for the Clinton, Illinois ESP site (ADAMS accession number ML061220489) and this approach has been incorporated into a new regulatory guide (RG 1.208 [DG-1146], "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion"). The Staff has recognized that the reference probability approach would require updating the reference probability value as new advances are made in the earth sciences. As mentioned above, the reference probability is computed from the median probabilities of exceeding the SSEs at 29 sites in the CEUS. The selected sites were intended to represent relatively recent designs, which used conservative seismic designs, in order to ensure an adequate level of conservatism in determining the SSE for future sites. However, as the reference probability is based on the probability of exceeding the SSEs at all 29 sites, new models of seismic activity or ground motion in the vicinity of a few sites would necessitate updating the seismic hazard estimates for all of the sites in order to determine a new reference probability. In other words, each prospective siting application would need to potentially justify a current reference probability value, which would require it to update the seismic hazard estimates for all of the 29 CEUS sites. The guidance in RG 1.165 was first used when prospective site owners

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				submitted the ESP applications for CEUS sites in 2003. Because the reference probability recommended in RG 1.165 is based on seismic hazard models from the late 1980s, the Staff recognized that this value was likely to be out of date. Two of the three ESP applicants found that using the reference probability approach recommended by RG 1.165 produced unreasonably high SSE ground motions. Based on the new information from the earth sciences described above, the Staff now estimates that the ground motion recurrence interval is likely to be lower for some CEUS sites. These issues are further described in a Staff memorandum to the Commission dated July 26, 2006 (ADAMS accession number ML052360044). Author: Clifford Munson SME: Clifford Munson Key Documents: RG 1.165
60	SER 2-178	SER Section 2.5.2.1.6	The Applicant has proposed that the seismic reference probability for the SSE for the ESP be relaxed by a factor of at least 5. Does the Staff agree with this relaxation and if so, why?	The Applicant's justification for using a higher reference probability (mean 5 x 10^{-5}) rather than the value recommended by RG 1.165 (median 1 x 10^{-5}) is based on three factors: 1. Recent ground motion models for the CEUS predict higher values for higher frequency ground motions (10 Hz); 2. Recurrence estimates for large earthquakes within the two major Central and Eastern United States (CEUS) seismic sources (New Madrid and Charleston, SC) have decreased from over a thousand years to 500-600 years; 3. Use of a mean rather than a median reference probability value. Since ground motion values for a given magnitude and distance generally follow a lognormal distribution, the Staff determined that the third factor is valid. For example, when the original reference probability value was determined using ground motion models from the late 1980s, the median 1×10^{-5} reference probability value was roughly equivalent to a mean 1×10^{-4} reference probability value. Updated ground motion models have demonstrated that mean and median values are not as far apart; however, mean estimates remain higher than

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				median estimates. To evaluate the overall change to the reference probability value, the Staff used the only current PSHA for the entire CEUS, which is the 2002 USGS PSHA. SER Section 2.5.2.3.6 on page 2-200 describes the Staff's independent analysis: To evaluate the applicant's use of a higher reference probability (5×10^{-5}) and use of mean rather than median PSHA results, the staff performed an independent analysis to reevaluate the reference probabilities for the 29 nuclear power plant sites in the CEUS that were used to determine the original reference probability. For its independent analysis, the staff used the most recent 2002 USGS PSHA mean and median hazard curves to determine the probability of exceeding the SSEs (reference probabilities) for the 29 CEUS sites. The staff also applied the same 5 Hz and 10 Hz site correction factors that were used in the LLNL seismic hazard analysis, published in 1993. Although the staff has not officially endorsed the 2002 USGS PSHA results, the staff was able to verify that the reference probability proposed by the applicant (5×10^{-5}) is sufficiently conservative. This larger reference probability value (5×10^{-5}) implies a lower return period (20,000 yrs) for the design ground motion; however, the staff was able to verify through its analysis that this revised reference probability results in a final SSE of adequate severity that is representative of the seismic hazard for the ESP site. Since the focus of the USGS PSHA is on the development of regional seismic hazard maps for use in building codes and the USGS does not use the expert elicitation methods recommended by RG 1.165, the Staff does not endorse the USGS hazard results. However, the Staff did use the results of the USGS PSHA to determine if there was an overall trend in the seismic hazard values for the CEUS. Using the 2002 USGS PSHA results, the Staff calculated reference probability values closer to 10^{-4} (median 6 to 7×10^{-5} and mean 8 to 9×10^{-5}). The specific median and mean values are not as relevant as the verification that the overall trend in seismic hazards has increased for the CEUS. Based on this result, the Staff concurred with the Applicant's use of a mean 5×10^{-5} reference probability value.

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				An update of either the EPRI or LLNL PSHAs for the entire CEUS in order to determine a new reference probability would require considerable resources and, due to future advances in the earth sciences, would likely be out-of-date within a few years. As such, the Staff did not believe that it was reasonable to expect the Applicant to perform an update of the EPRI PSHA for each of the 29 sites, used to determine the original reference probability value (1 x 10^{-5}). Author: Clifford Munson SME: Clifford Munson Key Documents: RG 1.165
61	SER 2-177	SER Table 2.5.2-1	Table 2.5.2-1 (SSAR Table 2.5-22) compares median and mean values of ground motion acceleration values for the1989 PSHA model and the Updated PSHA model. The ground motion acceleration values for the mean estimates were higher than the median estimated values. Were all of the sites participating in the sample in the CEUS? How many were in the sample? As regards the mean values, what portion of the sample had acceleration values higher than the mean value at the various frequencies?	Table 2.5.2-1 pertains only to North Anna, as it shows a comparison between the EPRI 1989 PSHA results and the updated PSHA, done by the Applicant for the North Anna ESP site. Table 2.5.2-1 compares specific spectral acceleration values for various frequencies at the 10^{-5} mean and median levels. The purpose of the table is to show the cumulative effects of the updates made by the Applicant to the EPRI 1989 PSHA in connection with the ESP application. The specific changes are described in SER Section 2.5.2.1.6 on pages 2-174 thru 2-177 and include updated ground motion models and new seismic sources characterizations. Author: Clifford Munson SME: Clifford Munson Key Documents: NUREG-1835, Section 2.5.2.1.6

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62	SER 2-177 to 2-201	SER Section 2.5.2.1.6	Applicant used two different methods to determine the ground motion response spectra for the final SSE. The first method, referred to as a performance-based method, was studied by the NRC Staff, who raised several questions and indicated to Applicant that it would need more time and resources to review this new method. Applicant then notified Staff that it would revise its submittal and base the selected SSE on the reference probability approach, in accordance with RG 1.165, indicating that it would retain the performance-based approach as an "alternate and further justification for the final SSE."	See the Staff's response to Board Question 63. <u>Author</u>: Clifford Munson <u>SME</u>: Clifford Munson <u>Key Documents</u>: RG 1.165, SSAR Table 2.5-24
63			In using the reference probability approach, Applicant departed from the recommendation clearly stated in RG 1.165 and used a higher reference probability (5x10-5 rather than 1x10-	**Would the use of a higher reference probability generally result in a lower spectral acceleration value?** Yes. PSHA results provide the ground motion hazard for a site in terms of the probability of exceedance for a range of ground motions at a number of frequencies. RG 1.165 prescribes that the reference probability value be used to determine the SSE for a site. Since the PSHA results provide ground motion exceedance probabilities for all of the potential earthquakes surrounding the site,

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			5). In justification of using the higher reference probability, Applicant listed three reasons: (1) higher ground motion estimates from the EPRI ground motion models, (2) shorter recurrence intervals for the New Madrid and Charleston seismic sources, and (3) the use of mean hazard instead of the median hazard. As pointed out in the SER at 2-199, each of these three factors, particularly the first two, increase the overall hazard for the CEUS and specifically for the 29 nuclear power plant sites used to determine the original reference probability. Would the use of a higher reference probability generally result in a lower spectral acceleration value? What would the difference be? Since most of the justification used by the Applicant would tend to increase the overall seismic hazard, how does that justify using a	the reference probability value is used to determine the most likely or controlling earthquakes for the site. For example, for the North Anna site, the two controlling earthquakes at the reference probability level are M5.4 at 12 miles and M7.2 at 191 miles. The first magnitude-distance pair is the high-frequency (i.e., 5 and 10 Hz) controlling earthquake and the second magnitude-distance pair is the low-frequency (1-2.5 Hz) controlling earthquake. These two controlling earthquakes were determined using the higher reference probability value (mean 5×10^{-5}) selected by the Applicant. Staff experience with PSHA results and the determination of controlling earthquakes for many sites has shown that these magnitude-distance values would not be that much different if the RG 1.165 recommended reference probability value (median 1×10^{-5}) were used. Particularly, the low-frequency controlling earthquake (M7.2 at 191 mi) would not change; however, the high-frequency earthquake would probably increase somewhat, but not significantly, in magnitude. To determine the SSE using the controlling earthquakes, ground motion response spectra are calculated for each of the controlling earthquakes. These response spectra are then scaled back to the PSHA hazard curves at the reference probability level. For a lower reference probability value the controlling earthquake response spectra would be scaled to a higher spectral acceleration value and, therefore, the final SSE would be higher. Conversely, for a higher reference probability value, the controlling earthquake response spectra would be scaled to a lower spectral acceleration value, resulting in a lower SSE. **What would the difference be?** As shown in SSAR Table 2.5-24, using the reference probability 5×10^{-5}, the high-frequency controlling earthquake response spectra is scaled to spectral acceleration value of about 0.4g (at 7.5 Hz). Using the 1×10^{-5} median reference probability level would result in a substantially higher spectral acceleration value. However, as described below, the Staff does not consider the 1×10^{-5} median reference probability level to be a representative value with regard to current seismic hazards for the CEUS. **Since most of the justification used by the Applicant would tend to increase the overall seismic hazard, how does that justify using a higher reference probability?**

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			higher reference probability?	The RG 1.165 recommended reference probability (median 1×10^{-5}) implies that new nuclear power plant sites in the CEUS should be designed to remain functional during and after an earthquake ground motion level with a median recurrence interval of 100,000 years. The Applicant's revised reference probability (mean 5×10^{-5}) implies that new nuclear power plant sites in the CEUS should be designed to remain functional during and after an earthquake ground motion level with a mean recurrence interval of 20,000 years. For determination of the SSE, the premise of RG 1.165 is that the seismic designs of currently licensed operating nuclear power plants provide adequate protection of public health and safety. In RG 1.165, the NRC Staff recommended basing an SSE for proposed sites on the median (50th percentile) annual probability of exceeding the SSE ground motion for a group of 29 operating NPP sites in the CEUS. Based on seismic source and ground motion models available at the time it was written, the NRC Staff calculated a median reference probability of 1×10^{-5}. Considering the advances in earth sciences that have occurred since the original reference probability was calculated, the Applicant proposed a higher value (mean 5×10^{-5}). The Staff review of this revised reference probability is described above in response to Board Question 60. In summary, the Staff determined that the Applicant's proposed higher reference probability value is approximately the value that would be determined for the 29 CEUS sites using current ground motion models and seismic source zone characterizations. Author: Clifford Munson SME: Clifford Munson Key Documents: RG 1.165, SSAR Table 2.5-24
64	SER 2-241	SER Section 2.5.4.3.7	According to Applicant, damping ratios for rock are generally between 0.5 and 4.5 percent. Applicant selected 2 percent for the zone III-V rock based on engineering judgment and experience. The	Seismic response spectra and ultimately the SSE represent the peak response of a single degree of freedom oscillator to earthquake excitation. The single degree of freedom oscillator plays a critical role in engineering seismology because it is used as a model for the response of buildings and many other types of structures to earthquake ground motion. To represent the loss of energy in each cycle of motion, damped oscillators are used as the model. In a building, damping is due to the absorption of energy by the elements of the building. The amount of damping is expressed in terms of the damping ratio. Structures are usually lightly damped with damping ratios between 0 and 20

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			Staff agrees. Why do all the seismic spectra curves in the SER and SSAR use 5% critical damping?	percent. By convention, seismic response spectra and SSE are expressed with a 5 percent critical damping ratio. Similar to structures, seismic wave amplitudes are reduced as waves propagate through rock. In geotechnical earthquake engineering, the parameter traditionally used as a measure of energy dissipation during harmonic excitation is the material damping ratio, *D*. Laboratory measurements of rock use a cyclic stress-strain curve to determine *D*. Author: Clifford Munson SME: Clifford Munson Key Documents: n/a
65	SER 1-5, SSER E-1	SER Section 1.3, 7/18/05 ACRS letter, Appendix E.	Applicant and NRC terminology appear to accept the possible existence of more than two new nuclear units at the North Anna site so long as the total thermal power is below 9000 MW. However, the ACRS letter of July 18 to Chairman Diaz states an ACRS conclusion that “the proposed site, subject to the permit conditions recommended by the NRC Staff, can be used for up to two nuclear power units each of up to 4300 MW [4500 MW] without undue risk to the public health and safety." Does the NRC view the ACRS statement as limiting	The Staff’s Safety Evaluation Report states that the Applicant may use the site to construct a unit or units producing up to 9000 MWt of power. The SER makes clear that the PPE is intended to accommodate multiple designs, which could ultimately result in the construction of more than two units on the site. The ACRS gave no indication that it would object to the construction of more than two units. The Staff would not, therefore, view a future submittal for more than two units as necessarily negating ACRS concurrence, provided the 9000 MWt total is not exceeded. In any event, the ACRS is required, pursuant to 10 CFR 52.87, to report on those portions of any future COL application that concern safety. Author: George Wunder SME: George Wunder Key Documents: NUREG-1835, 10 CFR 52.87

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			their concurrence only for the condition of two units?	
66	SER 15-4	SER Section 15.3	Why is there no discussion in the Application or the SER related to the planned measurement of radioactive materials in the air, soil and groundwater?	The application discusses the radiological environmental monitoring program in ER Section 6.2, (pp. 3-6-6 and 3-6-7). The FEIS discusses the radiological environmental monitoring program in Section 5.9.6 (p.5-70). The Applicant has committed to use the current radiological monitoring program to characterize the presence and levels of plant-derived radioactivity at offsite locations. (GS) Compliance with NRC regulations is achieved via the control and monitoring of radioactive effluent releases and by collecting and analyzing environmental media samples. The requirements of 10 CFR 20.1301 and 20.1302 on doses to members of the public, effluent radionuclide concentration limits of Appendix B, Table 2 to 10 CFR Part 20; and ALARA dose criteria of Appendix I to 10 CFR Part 50 are implemented through operational programs and procedures mandated by 10 CFR 50.36a, 10 CFR 50.34a, and Section IV of Appendix I to Part 50. The key operational program documents are the RETS or SREC, ODCM, and the REMP. The operational documents and programs (SREC, ODCM, and REMP) would be reviewed as part of any future CP or COL application. (JCD) The implementation of these programs and license conditions is routinely inspected by NRC Regional Inspectors. These inspections examine the licensee's radiological effluent monitoring and release programs to ensure their programs meet all NRC requirements and license conditions. (JCD) Author: Greg Stoetzel SME: Greg Stoetzel, Jean-Claude Dehmel Key Documents: 10 CFR Part 20, 10 CFR Part 50, NUREG-1835, NUREG-1811
67	SER 2-231	SER Section 2.5.4.1.6	Why is there such a wide range in the measured Factors of Safety (0.91 to 3.61) for soils in near proximity to each other?	The wide range in factors of safety ("FS") cited in the question (0.91 to 3.61) were calculated by Virginia Power for a detailed liquefaction analysis at the North Anna site in December, 1994, for a seismic margin assessment. This analysis is described in SER Section 2.5.4.1.8 on page 2-231. Analysis of the service water reservoir embankment gave FS values ranging from 0.91 to 3.61, with an average of more than 1.5. The Applicant stated that the few values below 1

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				occurred in localized zones, and concluded that overall FS across the embankment are well within acceptable limits. For its review of the ESP application, the Staff examined the Applicant's current liquefaction analysis using the ESP ground motions and did not examine the older results, which used lower ground motions. The Applicant performed a liquefaction analysis of the Zone IIA saprolitic soils and found some zones that may be susceptible to liquefaction. As a result, the Applicant stated that if safety-related structures are founded on the Zone IIA saprolitic soils, these soils would have to be improved to reduce potential settlements and to ensure that the factor of safety against liquefaction is greater than or equal to 1.1. SSAR Section 2.5.4.12 outlines several ground improvement techniques that would be implemented before the Zone IIA saprolitic soils could be used to support safety-related foundations. The Staff proposed Permit Condition 8 to ensure that the ESP holder and/or CP or COL applicant referencing such an ESP improves the Zone IIA saprolitic soils to reduce any liquefaction potential if safety-related structures are to be founded on them. <u>Author</u>: Clifford Munson <u>SME</u>: Clifford Munson <u>Key Documents</u>: NUREG-1835, Section 2.5.4.1.8, SSAR Section 2.5.4.12
68	SER 2-234	SER Section 2.5.4.1.10	Applicant has indicated that zone IIA saprolite is not suitable to support any safety-related structure without ground improvement and has proposed techniques to improve subsurface conditions. Soil borings indicate that IIA saprolite is abundant on the existing plant site and the ESP site. <u>See</u> Tables 2.5-29, 2.5-32, 2.5-38,	The Staff, in its review of SSAR Section 2.5.4.1.10, determined that the Applicant provided an adequate preliminary assessment of the static stability of the ESP site. However, as described in NUREG-0800 (Standard Review Plan) Section 2.5.4.10 and RS-002, for the Staff to perform a complete review of the site static stability, the COL or CP applicant should provide an analysis of the stability of all planned safety-related facilities when the locations of the plant structures are finally specified. As stated in SER Section 2.5.4.3.10 on pages 2–243 and 2-244: This analysis should include bearing capacity, rebound, settlement, and differential settlements, as well as lateral loading conditions for all safety-related facilities. Therefore, the Staff concludes that the applicant's description of the static stability is adequate to provide assurance of the stability of the ESP site, but

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			and 2.5-40. Construction sections of now abandoned former units 3 and 4, such as intake and discharge piping, containment pad, etc., might be salvaged and incorporated into the proposed new facilities. What actions would be taken to assure that settlement problems associated with certain sections of the existing plant do not occur at the new sites?	the Staff needs additional information to support any finding regarding detailed structure-specific stability. The need to provide an analysis of the stability of all planned safety-related facilities, including bearing capacity, rebound, settlement, and differential settlement under dead loads of fills and plant facilities, as well as lateral loading conditions is COL Action Item 2.5-6. Author: Clifford Munson SME: Clifford Munson Key Documents: SSAR Section 2.5.4.1.10, NUREG-0800, NUREG-1835, Section 2.5.4.3.10
69	SER 2-250	SER Section 2.5.6.1	According to Applicant, the North Anna Dam was designed and constructed to meet the requirements for a seismic Class 1 structure in support of the existing NAPS units. Does this mean that Lake Anna could be used for safety-related water use purposes for the existing units? If so, why was this not also considered for similar purposes with the proposed units?	Theoretically, yes. However, Lake Anna may experience low water levels caused by low-flow conditions in the North Anna River. To avoid this, the existing units use a service water reservoir not affected by fluctuating water levels in Lake Anna. Author: Goutam Bagchi SME: Kenneth See Key Documents: NAPS UFSAR Revision 42, dated 09/29/06
70	General		Why isn't it reasonable to use observed ground water flow and settling	Construction activities and engineered backfilling would be expected to significantly alter ground water flow around proposed Units 3 and 4. Therefore, the Staff does not consider the groundwater flow around Units 1 and 2 to be

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			data for Units 1 & 2 as a predictor for behavior of Units 3 & 4?	representative of that around the proposed units for use in estimating transport of radionuclides. Author: Goutam Bagchi SME: Goutam Bagchi Key Documents: n/a
71	3/2/05 ACRS Transcript 154-160	3/2/05 ACRS Transcript	The SSE for the proposed units is much higher than the SSE for the existing 2 units (0.15 g versus possibly 0.5g). Is this an issue for the existing plants or is it simply a different way of looking at seismic information?	Large high-frequency ground accelerations (> 10 Hz) are typical for thin soil and rock sites in the CEUS. The impact of these high frequency ground accelerations on nuclear structures was examined in the late 1990s as part of the Individual Plant Examination for External Events (IPEEE) Program. Based on the evaluations of the IPEEE results, the Staff determined that the large high-frequency ground accelerations have a limited impact on nuclear power plant structures. Although these large high-frequency ground accelerations do not generally impact nuclear structures (a 2g ground motion acceleration value at 15 Hz corresponds to less than 1/10 of an inch of displacement), they must be accounted for in the design analyses. Recognizing that the probability of exceeding the SSE at some of the currently operating reactor sites in the CEUS is higher than previously understood, the Staff recommended that the impact of a higher seismic hazard on operating nuclear power plants be evaluated through the generic issue identification and resolution process as Generic Issue 199. These issues are described in a Staff memorandum to the Commission dated July 26, 2006 (ML052360044). Author: Clifford Munson SME: Clifford Munson Key Documents: n/a
			Radiological Effluent Release Dose Consequences From Normal Operations	
72	SER 11-2	SER Section 11.1.3.1	In this section and throughout the report, it is presumed that fission product inventories scale directly with reactor	In the context of routine plant effluent releases, the source terms are based on information supported by certified reactor designs, assumed radionuclide inventories, or developed using NRC and industry guidance. For the North Anna ESP review, the Staff was asked to review the Applicant's data using a PPE concept. The PPE does not contain detailed information on reactor core physics

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			thermal power. <u>See</u>, for example, SER Section 15.3.4, p 15-6, “Source Terms." For some isotopes, this is not strictly true. As one example, Cs-134, a critical radionuclide, is produced by neutron capture in nonradioactive Cs-133 which is a fission product. Cs-134 thus scales with the square of reactor fluence not with reactor power. Is this effect of sufficient consequence to require modification of any of the radioisotope concentration tables?	and waste processing equipment that will be used to treat radioactive liquid and gaseous process and effluent streams. The estimates of radioactive materials discharged in the environment represent total aggregates without specifying fractional releases by buildings or vents, such as plant stack, spent fuel building, radwaste building, etc. Also, the presence of radioactivity is assumed to scale linearly with the power levels. While the Staff recognizes that there are various mechanisms that yield fission and activation products, such as that illustrated in the question using cesium-134, the methods used to derive radioactivity levels in primary coolant and primary steam are based on relationships that reflect the operational experience of the industry for both BWR and PWR plants. This approach is exemplified by the guidance of Regulatory Guide 1.112, and methods described in NUREG-0016 and NUREG-0017 and ANSI/ANS-18.1-1999 (Ref. a - d). The guidance presents acceptable methods for calculating annual average expected releases of radioactive materials in liquid and gaseous effluents released by light-water-cooled nuclear power reactors. The types and amounts of radioactivity present in effluents, expressed as radionuclide distributions and concentrations, are dependent on methods used to treat effluents before being discharged. For example, the radiological and chemical properties of liquid wastes and types of filtration and adsorbent media used to treat liquid wastes have significant impacts on the selection and effectiveness of treatment methods used, expressed as decontamination factors or removal efficiencies. Accordingly, the presence and concentrations of specific radionuclides in effluents are influenced by many factors, including their production mechanisms in fuel pellets, transport from within fuel assemblies to the primary coolant and primary steam, type of reactor coolant chemistry used, effectiveness of primary coolant purification systems, and effectiveness of waste treatment systems. As a result, these competing factors tend to yield different radiological profiles when comparing radionuclide distributions and concentrations among that expected in core inventories with that observed in primary coolant and steam activity, and that characterizing plant effluent concentrations after treatment. Consequently, the production mechanism of Cs-134 (via the decay of Xe-133 to Cs-133 and its neutron activation to Cs-134), by itself, is not of sufficient consequence given all other competing factors to require a modification of the listing of radionuclides identified in effluents since it is primarily dependent on the migration of noble gases within fuel pellets and the removal effectiveness of cesium by primary

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				coolant purification and waste treatment systems. It should be noted that Cs-134 is included in the list of radionuclides expected in liquid and gaseous effluents. At the CP or COL stage, the Staff will have specific details on the Applicant's reactor design, radioactive waste processing systems, effectiveness of selected waste processing method, and concentrations of effluents. Such detailed information will provide the means for the Staff to evaluate the proposed effluent source terms and perform independent verifications of radioactivity levels and assess doses to members of the public. The responsibility rests on CP or COL Applicants to describe the type of technology that will be used and maintained to meet NRC regulatory requirements and license conditions. This issue is subsumed by COL Action Item 11.1-1. Supporting references: a. Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," Rev. 0-R, May 1977. b. NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactor, BWR-Gale Code," Rev. 1, January 1979. c. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactor, PWR-Gale Code," Rev. 1, April 1985. d. ANSI/ANS-18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors," September 1999. <u>Author</u>: Jean-Claude Dehmel <u>SME</u>: Greg Stoetzel <u>Key Documents</u>: RG 1.112, NUREG-0016, NUREG-0017, ANSI/ANS-18.1-1999, SER

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73	SER 11-2, SSER viii	SER Section 11.1.3.1, SSER Exec. Sum.	The thermal power limit was increased to 9000 MW (SSER viii) with an appropriate scaling of isotope concentrations. However, the original plant designs were based upon particular temperature and flow conditions and a power increase would appear to produce a shift in one or the other of these numbers. Wouldn't this factor contribute to increased fission product release that is greater than a linear extrapolation?	The response to this question is related to the Staff's response to Board Question 72, above. For routine plant effluent releases, source terms are based on information supported by certified reactor designs, assumed as part of radionuclide inventories, or developed using NRC and industry guidance. For the ESP review, the Staff was asked to review the Applicant's data using a PPE concept. The PPE does not contain detailed information on reactor core physics and waste processing equipment that will be used to treat radioactive liquid and gaseous process and effluent streams. The estimates of radioactive materials discharged in the environment represent total aggregates without specifying fractional releases by buildings or vents, such as plant stack, spent fuel building, radwaste building, etc. As noted in the application (see SSAR Section 1.3.1 and the footnotes to SSAR Tables 1.3-7 and 1.3-8), the source term for the ABWR was scaled up to 4300 MWt from the certified design of 3926 MWt. For the ESBWR design, the expected liquid and gaseous effluent source terms were increased by a margin of 25% without changing the plant power level of 4500 MWt, because this plant design has not yet been certified by the NRC. This approach was used to ensure that if the ESBWR design is approved, the added margin should address possible differences in estimates of radioactive effluent source terms between the draft and final versions of plant design features used to define and control both effluent source terms. The presence and concentrations of specific radionuclides in effluents are influenced by many factors, starting with their production mechanisms in fuel pellets, transport from within fuel assemblies to the primary coolant and primary steam, type of reactor coolant chemistry used, effectiveness of primary coolant purification systems, and effectiveness of waste treatment systems. Moreover, operational specifications, imposed to maintain fuel integrity, core power distribution heat flux rates, linear heating generation rates, maximum peak power density, fuel burn-up rate, and primary coolant chemistry are expected, collectively, to minimize the fraction of failed fuel assemblies and release of fission and activation products into primary coolant. The types and amounts of radioactivity, expressed as radionuclide distributions and concentrations, present in effluents are dependent on methods used to treat effluents before being discharged. For example, the radiological and chemical properties of liquid wastes and types of filtration and adsorbent media used to treat liquid wastes

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				have significant impacts on the selection and effectiveness of treatment methods used, expressed as decontamination factors or removal efficiencies. As a result, these competing factors tend to yield different radiological profiles, when comparing radionuclide distributions and concentrations among that expected in core inventories, observed in primary coolant and steam activity, with that characterizing plant effluent concentrations after treatment. While the Staff recognizes that there are various and complex processes that have effects on generation and removal of fission and activation products, the methods used to derive radioactivity levels in primary coolant and primary steam are based on relationships that reflect the operational experience of the industry for both BWR and PWR plants. This approach is exemplified by the guidance of Regulatory Guide 1.112 and methods described in NUREG-0016 and NUREG-0017 and ANSI/ANS-18.1-1999 (Ref. a - d). The guidance presents acceptable methods for calculating annual average expected releases of radioactive materials in liquid and gaseous effluents released by light-water-cooled nuclear power reactors. Consequently, the assumption that linear scaling of fission and activation products with power levels (in this case from 3926 to 4300 MWt) is not an unreasonable first order estimate for the purpose of characterizing expected amounts of fission and activation products in effluents (Ref. e). Finally, the Staff will have specific details at the CP or COL stage on the Applicant's reactor design, radioactive waste processing systems, effectiveness of selected waste processing method, and concentrations of effluents. Such detailed information will provide the means for the Staff to evaluate the proposed effluent source terms and perform independent verifications of radioactivity levels and assess doses to members of the public. The responsibility rests on the CP or COL Applicant to describe the type of technology that will be used and maintained to meet NRC regulatory requirements and license conditions. This issue is subsumed by COL Action Item No. 11.1-1. Supporting references: a. Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," Rev. 0-R, May 1977.

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				b. NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactor, BWR-Gale Code," Rev. 1, January 1979. c. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactor, PWR-Gale Code," Rev. 1, April 1985. d. ANSI/ANS-18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors," September 1999. e. Karl-Heinz Need, "The Radiochemistry of Nuclear Power Plants with Light Water Reactors," Walter de Gruyter, Berlin, New York, 1997. Section 3.2, p. 66-93. <u>Author</u>: Jean-Claude Dehmel <u>SME</u>: Jean-Claude Dehmel <u>Key Documents</u>: RG 1.112, NUREG-0016, NUREG-0017, ANSI/ANS-18.1-1999, NUREG-1835
74	SER 11-3	SER Section 11.1.3.2	Identical values are quoted for maximum annual dose equivalents during normal operation for both thyroid and total body doses. This would appear to imply that air or water exposure to radioactive iodine is inconsequential. Is this not an unexpected result?	The doses reported in the September 2005 FSER (NUREG-1835) have been superseded by those reported in FSER Supplement 1, dated November 2006. With respect to this question, the revised maximum annual dose results (p.11-5) due to liquid effluents discharged by a single new unit are 0.81 mrem to the total body for the adult, 0.68 mrem for infant thyroid, and 2.5 mrem to the bone for the child. <u>Author</u>: Jean-Claude Dehmel <u>SME</u>: Jean-Claude Dehmel <u>Key Documents</u>: NUREG-1835
75	SSER 11-1	SSER Section 11.1	Is there a regulation or other authoritative definition of "source term?" If so, please	The following presents various definitions of "source term," set forth in NRC regulations, and NRC guidance in calculating releases of radioactive materials in gaseous and liquid effluents from light water reactors. Given the context of the question, the definitions provided here focus on those commonly applied in

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			provide it.	characterizing radioactive materials present in process and effluent streams associated with routine plant operations. Other than the definition contained in 10 CFR Part 50.2, NRC regulations do not define “source term.” As can be noted by the various definitions listed below, the definitions of "source term" vary even within the same context. It should be noted that the definition of "source term" used to characterize radioactive materials associated with reactor accidents and accident releases may be different in consideration of the type of accident and modes of releases of radioactivity into the environment. Such definitions are not included here. a. Regulatory Definition In 10 CFR 50.2, the definition of “source term” is: "Source term refers to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release." b. Definitions Provided in Guidance used by the NRC Regulatory Guide 1.112 (Ref. a) refers to the use of computer codes described in NUREG-0016 (Ref. b) and NUREG-0017 (Ref. c) for calculating releases of radioactive materials in gaseous and liquid effluents from light water reactors. The definition presented in NUREG-0016 is: "Source Term: The calculated annual average quantity of radioactive material released to the environment from a nuclear power reactor during normal operation including anticipated operational occurrences. The source term is the isotopic distribution of radioactive materials used in evaluating the impact of radioactive releases on the environment. Normal operation includes routine outages for maintenance and scheduled refuelings." The definition in NUREG-0017 (Ref. c) is slightly different as it does not include the last sentence about plant outages, i.e., "Normal operation includes routine outages for maintenance and scheduled refuelings."

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				In the context of radiological assessment, NUREG/CR-3332 (Ref. d) provides the following definition in its glossary of terms: "Source term: The quantity of radioactive material released to the biosphere, usually expressed as activity per unit of time. Source terms should be characterized by the identification of specific radionuclides and their physical and chemical forms." In ANSI/ANS-18.1-1999 standard (Ref. e), source term is implicitly defined as follows: "The purpose of this standard is to provide a uniform approach, applicable to light-water-cooled nuclear power plants, for the determination of expected concentrations in fluid streams. Through application of this standard, a common basis for the determination of radioactive source terms for normal operating conditions is established, with the goal of providing a consistent approach for those involved in the design of these facilities. Utilization of this standard is expected to aid the licensing process and the public's understanding of the impact of nuclear power relative to radionuclide concentrations and possible releases to the environment." Regulatory Guide 1.21 (Ref. f) uses "release" or "releases" as a surrogate for source term. It provides the following three definitions: "Abnormal releases - unplanned or uncontrolled release of radioactive material from the site boundary." "Batch releases - discontinuous release of gaseous or liquid effluent which takes place over a finite period of time, usually hours or days." "Continuous release - release of gaseous or liquid effluent which is essentially uninterrupted for extended periods during normal operation of the facility." Standard Review Plan Section 11.1 (NUREG-0800, Ref. g) describes the procedural NRC review and evaluation of applicant source terms.

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				In SRP Section 11.1, the factors reviewed for applicant source terms are: "The SPLB review of radioactive source terms includes consideration of parameters used to determine the concentration of each isotope in the reactor coolant; fraction of fission product activity released to the reactor coolant; concentrations of all nonfission product radioactive isotopes in the reactor coolant; leakage rates and associated fluid activity for all potentially radioactive water and steam systems; and potential sources of radioactive materials in effluents that are not considered in the applicant's safety analysis report (SAR) Section 11.2, "Liquid Waste Management Systems," and SAR Section 11.3, "Gaseous Waste Management Systems."" In SRP Section 11.1, the evaluation for applicant source term is defined as: "For evaluating the source terms, the applicant should provide the relevant information in the SAR as required by 10 CFR Part 50, Section 50.34 and 10 CFR Part 50, and Section 50.34a. This technical information should include all the basic data listed in Appendix A (BWRs) and Appendix B (PWRs) to Regulatory Guide 1.112 required in calculating the releases of radioactive material in liquid and gaseous effluents (the source terms). An acceptable method for satisfying the criteria given in items 1 through 6 consists of using the Gaseous and Liquid Effluent (GALE) Computer Code and the source term parameters given in NUREG-0016 or NUREG-0017 for BWRs and PWRs respectively. Complete listings of the GALE Computer Codes for BWRs and PWRs are given in NUREG-0016 and NUREG-0017 respectively." In NCRP Report No. 92 (Ref. h), "source term" is characterized as: "The source terms obtained include the quantity of each radionuclide released from each release point and characterize each release as short or long duration." Supporting references:

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				a. Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," Rev. 0-R, May 1977. b. NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactor (BWR-Gale Code)" Rev. 1, January 1979. Page 1-4. c. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactor (PWR-Gale Code)" Rev. 1, April 1985. Page 1-6. d. NUREG/CR-3332, "Radiological Assessment - A textbook on Environmental Dose Analysis," September 1983. Page G-6. e. ANSI/ANS-18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors," September 1999. Page 1. f. Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Rev. 1, June 1974. Page 1.21-2. g. NUREG-0800, "Standard Review Plan, Chapter 11.1, Source Terms," draft, rev. 3, April 1996. Note: The SRP is being updated and is expected to be issued as final in mid-Spring 2007. h. NCRP Report No. 92, "Public Radiation Exposure from Nuclear Power Generation in the United States," National Council on Radiation Protection and Measurements, December 1987. Section 9.4.1, p.104. Author: Jean-Claude Dehmel SME: Jean-Claude Dehmel Key Documents: 10 CFR Part 50, RG 1.112, NUREG-0016, NUREG-0017, NUREG/CR-3332, ANSI/ANS-18.1-1999 standard, RG 1.21, NUREG-0800, Section 11.1, NCRP Report No. 92.

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76	SSER 11-1	SSER Section 11.1.1	Based on the PPE for Units 3 and 4, what radiation dose is received (a) immediately outside the reactor containment and (b) at the EAB boundary, from direct transmission of radiation through the reactor shield?	(a) The PPE does not contain values for the radiation dose immediately outside the reactor containment. However, the Staff evaluated the plant radiation zone drawings for three reactor designs (the ABWR, the ESBWR, and the AP1000) which are included in the PPE (the PPE values were based on seven reactor designs). The ABWR, the ESBWR, and the AP1000 are three reactor designs currently being considered by industry for possible COL applications. For all three of these designs, the reactor vessel itself and the surrounding containments provide sufficient shielding to attenuate radiation from the reactor core and radioactive components inside the reactor building to very low levels outside the reactor building. For the ABWR design, radiation levels outside the reactor building, from direct transmission of radiation through the reactor shield, are generally less than 1 mrem/hr, and for the ESBWR design, generally less than 0.6 mrem/hr. For the AP1000 reactor design, radiation levels outside the containment/shield (reactor) building, from direct transmission of radiation through the reactor shield, are generally less than 0.25 mrem/hr. For each of these designs, at grade level, more than half of the perimeter of the reactor building is surrounded by adjoining buildings. Within these adjoining buildings, there are portions of the exterior of the reactor building where radiation levels can exceed the dose rate of 1 mrem/hr for the ABWR design, 0.6 mrem/hr for the ESBWR design, and 0.25 mrem/hr for the AP1000 design. Some of the radiation sources which can contribute to higher dose rates in these areas outside the reactor building include radiation streaming from inside the reactor building through piping or other penetrations in the reactor building secondary containment and contained sources, such as tanks, pumps, and piping, in areas adjacent to the exterior of the reactor building. An example of one such area for the ABWR and ESBWR designs is the steam tunnel between the reactor building and the turbine building which contains the main steam lines connecting the reactor vessel to the turbines. Dose rates in the steam tunnel during operation can exceed 100 mrem/hr (the highest dose rate zone designation used for the GE designs). An example of one such area for the AP1000 design is the area outside the reactor building adjacent in the fuel transfer canal. Dose rates in the fuel transfer canal can exceed 1 rem/hr when a spent fuel assembly is being transferred through the spent fuel transfer canal from the reactor building. The steam tunnel, fuel transfer canal, and other accessible penetrations to the reactor building are all well shielded and located in areas where access is limited

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				by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Supporting references: a. ABWR Design Control Document, SSAR Section 12.3.2.3 and Chapter 12 radiation zone map drawings. b. ESBWR Design Control Document, SSAR Section 12.3.2.2.3 and Chapter 12 radiation zone map drawings. b. AP1000 Design Control Document, SSAR Section 12.3.2.2.1 and Chapter 12 radiation zone map drawings. (b) NUREG-1437, Vol. 1 states, in part, that "...because the primary coolant of an LWR is contained in a heavily shielded area, dose rates in the vicinity of light water reactors are generally undetectable and are less than 1 mrem/year at the site boundary." NUREG-1437 was issued in 1996 and is based on data from the current generation of operating light water reactors. Since the advanced reactor designs under consideration by Dominion are expected to provide shielding that is at least as effective as existing light water reactors, direct dose contribution from the new units at the EAB would be expected to be of the same magnitude. The average annual TLD readings (averaged over a 7-year period) measured at the protected area fence location closest to the proposed site of the new units from the existing Units 1 and 2 is 56 mrem/year. See ER Section 4.5.3.1 and Table 4.5-1. This dose rate includes background radiation. Since the EAB is nearly 5000 feet from this location, the dose from direct transmission of radiation through the reactor shield from the existing two operating units at the EAB would be significantly less than 1 mrem/year. Supporting references: a. NUREG-1437, Vol. 1, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants", May 1996, Section 4.6.1.2.

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				b. North Anna ESP Application – Environmental Report Section 4.5.3.1 and Table 4.5-1. <u>Author</u>: Charles Hinson <u>SME</u>: Charles Hinson <u>Key Documents</u>: ABWR DCD, ESBWR DCD, AP1000 DCD, NUREG-1437
77	SSER 11-2	SSER Table 11.1-1	Legal Question: Table 11.1-1 refers to the Part 50 Appendix I doses as "objectives." Please explain how these objectives are included in the proposed ESP and whether they are legally enforceable. Please explain whether it would it be a violation to exceed these objectives.	See the "NRC Staff Legal Brief in Response to Licensing Board's Safety-Related Questions"
78	SSER 11-2	SSER Table 11.1-1	Legal Question: Table 11.1-1 refers to the Part 50 Appendix I doses on a per unit basis. Please explain whether it is your position that, since the Dominion group of companies would have four reactors on the site, it would be allowed to quadruple the amount of radiation it can release under Appendix I?	See the "NRC Staff Legal Brief in Response to Licensing Board's Safety-Related Questions"
79	SSER 11-2	SSER Table 11.1-1	Legal Question: Table 11.1-1 refers to the 40 CFR Part 190	See the "NRC Staff Legal Brief in Response to Licensing Board's Safety-Related Questions"

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			environmental dose standards. Would it be a violation to exceed these standards? How will they be incorporated into the proposed ESP?	
80	SSER 11-2	SSER Table 11.1-1	Legal Question: Table 11.1-1 specifies that the 40 CFR Part 190 dose limits are for the entire site and apply to all operating units. How will the Part 190 25 mrem/yr total body dose limit be allocated between the two existing reactors (Units 1 and 2) and proposed Units 3 and 4? How will compliance be monitored and measured?	Although Board Question 80 is designated as a legal question, the Staff considered the second portion of the inquiry (concerning how compliance will be monitored and measured) to be better addressed by the technical staff. Therefore, that response is provided below, while the first portion is addressed in the Staff's legal brief. Compliance with the EPA's 40 CFR Part 190 environmental radiation standards is required of NRC reactor licensees by 10 CFR 20.1301(e). This requirement acts in concert with NRC requirements addressing effluent radionuclide concentration limits (Appendix B, Table 2 to 10 CFR Part 20); dose limits to members of the public (10 CFR 20.1301 and 20.1302); and ALARA dose criteria (Appendix I to 10 CFR Part 50). The requirements of 10 CFR 20.1301(e) are implemented through operational programs and procedures mandated by 10 CFR 50.36a, 10 CFR 50.34a, and Section IV of Appendix I to Part 50. The key operational program documents are the Radiological Effluent Technical Specifications (RETS) or Standard Radiological Effluent Controls (SREC), Offsite Dose Calculation Manual (ODCM), and the Radiological Environmental Monitoring Program (REMP). Under 40 CFR Part 190, compliance with dose limits is assessed against the entire site and all sources of radioactivity and external radiation, regardless of the number of power plants. The measured sources of radioactivity include all liquid and gaseous effluent releases, as well as other sources of radiation. Compliance is assessed against the whole site and not on the basis of individual plants. The implementation of these programs and license conditions is routinely inspected by NRC Regional Inspectors. These inspections examine the licensee's radiological effluent monitoring and release programs to ensure that these programs meet all NRC requirements and license conditions. If a plant were to exceed the dose limits of 40 CFR Part 190 or any other requirements of Part 20, the inspection would identify the cause and determine whether a proper

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				response and corrective actions was taken by the licensee. Under the provisions of 10 CFR 20.1301(f), the NRC may impose additional restrictions after evaluating the impacts on members of the public in light of commitments and characterizations contained in the CP or COL application. Author: Jean-Claude Dehmel SME: Jean-Claude Dehmel Key Documents: 10 CFR Part 20, 10 CFR Part 50, 40 CFR Part 190
81	SSER 11-3	SSER Section 11.1.1	The SER states that the Applicant calculated a collective whole body dose for the population within 50 miles of the ESP site. Please provide this collective whole body dose amount and the calculations supporting it. Please confirm whether this dose is based on the PPE maximums and that it segregates the existing units from the proposed units.	Collective whole body doses estimated by the Applicant are presented in Table 5.4-12 of the ER, p. 3-5-148. The doses are transcribed in Table 2, attached hereto, based on the information provided in the application. The results represent total doses for each and both proposed new units by type of radioactive effluents and compares the results against collective doses associated with natural background radioactivity. The dose estimates are based on a total population of about 2.8 million people and exclude the occupational workforce. Other than the information presented in the supporting section of the ER (Section 5.4, Radiological Impacts of Normal Operation, p. 3-5-131), the Applicant did not provide to the Staff any other documentation of the supporting calculations, such as computer input and output files. The doses reflect the radioactive source terms developed using the PPE concept and do not include radioactive releases associated with the operation of the two existing units. Author: Jean-Claude Dehmel SME: Jean-Claude Dehmel Key Documents: ER
82	SSER 11-3	SSER Section 11.1	NRC Reg. Guide 8.29 uses a coefficient of 4 x 10^{-4} fatal cancers per rem for purposes of occupational radiation risk estimates and states that "the scientific community generally	The population dose within a 50-mile radius of proposed Units 3 and 4 was estimated to be 56 person-rem (see ER Table 5.4-12). The Applicant used the population estimate for the year 2040 (~2.8 x 10^6 persons) in calculating the population dose (see ER Table 2.5-8). In response to RAI 2.1.3-1, the Applicant revised its SSAR and provided a population estimate for the year 2065 of ~3.7 x10^6 persons (see SSAR Figures 2.1-8A, 2.1-13A and 2.1-14). Using the coefficient of 4 x 10^{-4} fatal cancers per rem from NRC RG 8.29, and scaling the population dose for the year 2065 population, the estimated fatal cancers for 40

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			assumes that any exposure to ionizing radiation can cause biological effects." Assuming (a) a linear no threshold application of the Reg Guide coefficient, (b) that proposed Units 3 and 4 operate for 40 years, and (c) using the population estimate provided by the Applicant in response to RAI 2.1.3-1 (SER page 2-8), please calculate and provide the estimated number of additional fatal cancers resulting from routine operation of Units 3 and 4 for the 50 mile radius area assuming the two units operate for 40 years.	years of operation would be: (56 person-rem/yr) (3.7 x 10^6 persons/ 2.8 x 10^6 persons) (4 x 10^{-4} fatal cancer/person-rem) (40 yr) = 1.2 fatal cancers Author: Greg Stoetzel SME: Jack Cushing, Jean-Claude Dehmel Key Documents: SSAR, ER, RG 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure"
83		SSER Section 11.1.1	Have you calculated or estimated the collective whole body dose for the population within 50 miles of the ESP site in the event of a fuel melt DBA? If so, please provide these data and the basis for your estimate or calculation.	The Staff did not calculate or estimate in its SSER the collective whole body dose for the population within 50 miles of the ESP site. There are no regulatory requirements to calculate or estimate the population dose within 50 miles of the ESP site. Author: Jay Lee SME: Jay Lee, Van Ramsdell Key Document: NUREG-1835, Supplement 1, NUREG-1811

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84	SSER 11-4	SSER Section 11.1.3.1	Legal Question: 10 CFR § 20.1301(a) specifies that "each licensee" shall conduct operations so that the TEDE to individual members of the public from the "licensed operation" does not exceed 100 mrem per year, exclusive of background. In the case of multiple reactors at a site, would it ever be possible to multiply the maximum dose allowed by the number of units so that a four unit site could provide an exposure up to 400 mrem per year to an exposed individual? If this is ever possible, under what conditions would it be allowed?	See the "NRC Staff Legal Brief in Response to Licensing Board's Safety-Related Questions"
85	SSER 11-4	SSER Section 11.1.3.1	The SSER refers to Table 5.4-11, which specifies that the total radioactive effluents from the plants will produce a dose of 6.4 mrem/yr and that the total from the "existing units" is 0.32 mrem/yr. Is this correct? Why does the PPE for the two new reactors show them emitting twenty times the amount	Yes, the estimated total body dose for both proposed new units is higher than that associated with the existing operating units. The difference is an artifact of the application of the Plant Parameter Envelope (PPE) concept. Under the PPE concept, the radiological source terms are maximized for the purpose of bounding the upper amounts of radioactivity assumed to be present in plant liquid and gaseous effluents for the two new proposed units. This approach yields dose results that are extremely conservative and contrast significantly with actual dose data from operating plants. This conclusion is supported by historical data on radioactive effluents released from nuclear power plants that show that doses to members of the public in unrestricted areas typically are well below the ALARA dose objectives of Appendix I to 10 CFR Part 50. <u>Author</u>: Jean-Claude Dehmel

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			of radiation as the two existing reactors?	SME: Jean-Claude Dehmel Key Documents: 10 CFR Part 50
86	SSER 11-4, 11-5	SSER Section 11.1.3.2	The SER states that it performed independent evaluations or calculations and obtained "similar" results for the following tables of data provided by the Applicant. Please provide the Staff's independent calculations, evaluations, and similar results for Tables 5.4-6, 5.4-7, 5.4-8, 5.4-9 and 5.4-10 of the ER.	Using the information contained in the application as input data, the Staff performed independent evaluations of compliance with liquid and gaseous effluent concentrations against the limits of Table 2 of Appendix B to 10 CFR Part 20, and doses to the maximally exposed individual against the ALARA dose objectives of Appendix I to 10 CFR Part 50. The results are set forth as Attachment IHPB-1 hereto. As is noted in the SER, the Staff reached conclusions similar to those of the Applicant. Author: Jean-Claude Dehmel SME: Jean-Claude Dehmel Key Documents: 10 CFR Part 20, 10 CFR Part 50, NUREG-1835
87	SSER 11-4, 11-6	SSER Section 11.1.3.1, 11.1.3.2	The SER states that the Applicant's results of 6.4 mrem/yr for the whole body, 27 mrem/yr for the thyroid, and 11 mrem/yr to bone are smaller than the maximum doses specified in 40 CFR § 190.10(a). Did the Staff calculate the results? What were the Staff's results for whole body, thyroid, and bone?	Using the information contained in the application as input data, the Staff performed an independent evaluation of doses to the maximally exposed individual against the ALARA dose objectives of Appendix I to 10 CFR Part 50. The results of this analysis were applied for the purpose of confirming compliance with the EPA's environmental radiation standards of 40 CFR Part 190. The Staff's annual dose results are 6 mrem for the total body, 27 mrem for the thyroid, and 11 mrem for the bone. Author: Jean-Claude Dehmel SME: Jean-Claude Dehmel Key Documents: 10 CFR Part 50, 40 CFR Part 190
			Emergency Planning	
89	SER 13-1	SER Section 13.3	In the event of an emergency, what are the respective responsibilities of Dominion Resources,	The detailed respective responsibilities associated with the proposed new reactor units would need to be addressed in a subsequent COL or CP application referencing the ESP, as part of the required complete and integrated emergency plan. At that time, the planning standards and evaluation criteria in NUREG-0654/FEMA-REP-1 (NUREG-0654) would apply. In evaluation criteria

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			Virginia Electric Power Co., Dominion Nuclear North Anna, North Anna Power Station (the Applicant) and Old Dominion Electric Corporation? Would personnel from these respective organizations have to work in close cooperation on emergency issues?	II.A.1.b of (NUREG-0654) planning standard II.A, "Assignment of Responsibility (Organization Control)," the CP or COL applicant should specify its concept of operations and relationship to the total effort for each organization <u>and sub-organization</u> having an operational role in emergency response. Personnel from those respective organizations that are identified in the CP or COL application as having responsibilities under the emergency plan, would have to work in close cooperation on emergency issues, consistent with the emergency plan. A detailed review of the respective responsibilities of the various corporate entities and sub-organizations is beyond the scope of review for a major features emergency plan. This is consistent with Supplement 2 to NUREG-0654/FEMA-REP-1 (Supplement 2, ADAMS Accession No. ML050130188, April 1996) and NRR Review Standard (RS)-002, "Processing Applications for Early Site Permits" (ADAMS Accession No. ML040700236, May 3, 2004). RS-002 Section 4.5, "Use of Existing Information From Nearby Facilities for ESP Applications," provides (in part) the following, applicable review guidance: *Additional guidance for emergency planning review* – The extent to which emergency planning information for an operating reactor site will be reviewed will be dependent upon the specific ESP application. In general, the existing elements of an established emergency preparedness program and emergency planning information that are relevant to, and provided (or incorporated by reference) in the ESP application will be considered acceptable and adequate; and a detailed review will not be necessary. The Staff used the guidance in Supplement 2 to NUREG-0654/FEMA-REP-1 (Supplement 2), major features II.A, "Assignment of Responsibility (Organization Control)," in its review to evaluate the adequacy of the Applicant's description of corporate responsibilities. Major feature II.A merely asks for the identification of various response organizations and a description of contacts and arrangements. The detailed level of respective responsibilities for the corporate entities and sub-organizations is not within the scope of review for this aspect of a major features emergency plan, and the Staff did not expect or ask for such detailed information. If the CP or COL applicant proceeded with the development of new unit(s) at the ESP site, it would enter into an arrangement with Virginia Power to coordinate and implement an integrated emergency plan, which, in effect, would

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				extend the existing emergency planning and preparedness activities to the new unit(s). Author: Bruce Musico SME: Bruce Musico Key Documents: NUREG-0654
90	SER 13-3	SER Section 13.3.1.1	The SER uses the term ETE (evacuation time estimate) and also refers to "the ETE" as if it is a specific document. Are all SER references to "the ETE" a reference to the "EM/TEC01-220, "Evacuation Time Estimates for the North Anna Power Station and Surrounding Jurisdictions," dated November 2, 2001? Please provide "the ETE."	Yes, there is only one ETE (i.e., "Evacuation Time Estimates for the North Anna Power Station and Surrounding Jurisdictions," IEM/TEC01-220, November 2, 2001, Innovative Emergency Management (IEM), Inc.), which is available under ADAMS Accession No. ML041190476. Additionally, the ETE is provided as an attachment to this response. Author: Bruce Musico SME: Bruce Musico Key Documents: NUREG-1835, SSAR
91	SER 13-13	SER Section 13.3.1.1	Section 13.3.3.3 covers "Onsite Emergency Organizations." For purposes of the ESP application, is the NAPS site (beyond the ESP boundary) considered not "onsite?" If not, please explain how the term onsite and offsite are used with regard to emergency planning. Are NAPS and ESP	Yes, for purposes of emergency preparedness and response, the NAPS site (Units 1 and 2) and ESP site (Units 3 and 4) would be treated as one site (i.e., "onsite"). SER Section 13.3 (page 13-1) states that the proposed ESP site footprint consists of a portion of the existing North Anna Power Station (NAPS) site and is located immediately adjacent to NAPS, such that very little distinction exists between the NAPS site and the ESP site for purposes of emergency planning. Author: Bruce Musico SME: Bruce Musico Key Documents: NUREG-1835, SSAR

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			treated as one site?	
92	SER 13-37	SER Section 13.3.3.10.3	The SER states that "Dominion would use both fixed and portable radiation monitoring equipment to perform dose assessment…" Does the use of the word "Dominion" here also include Virginia Power and Dominion Nuclear North Anna?	As stated in SER Section 13.3 (page 13-1), "Dominion" refers to the ESP Applicant, Dominion Nuclear North Anna, LLC. Virginia Power is a separate corporate subsidiary of Dominion Resources, and it operates the existing NAPS Units 1 and 2. With regard to whom would use the radiation monitoring equipment to perform dose assessment, this activity would be performed in accordance with an agreement between a COL or CP applicant and Virginia Power, in order to coordinate and implement an integrated emergency plan with Virginia Power. As discussed in the Staff's response to Board Question 89, the detailed respective responsibilities associated with the proposed new reactor units, including the dose assessment responsibilities, would need to be addressed in a subsequent COL or CP application that references the ESP. The Staff would evaluate the application against the requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50, using applicable guidance documents (e.g., NUREG-0654/FEMA-REP-1). <u>Author</u>: Bruce Musico <u>SME</u>: Bruce Musico <u>Key Documents</u>: NUREG-1835, 10 CFR Part 50
93	SER 13-39	SER Section 13.3.3.11.1	The SER states that "evacuation decisions would be based on dose projections or offsite monitoring results." Section 5.9.6 "Radiological Monitoring" in the North Anna EIS provides a general description of the offsite monitoring to be carried out at Units 3 and 4. Please explain why this information is not	The discussion in Section 5.9.6, "Radiological Monitoring," of the EIS (NUREG-1811) addresses only the environmental monitoring program associated with the baseline and operational aspects of the site. In contrast, SER Section 13.3.3.10, "Accident Assessment (Supplement 2, Major Feature I)," and SER Section 13.3.3.11, "Protective Response (Supplement 2, Major Feature II.J)," apply only to the applicant's description of monitoring and dose projections associated with a radiological emergency situation. That is, EIS Section 5.9.6 deals with normal operations, while SER Section 13.3.3.11 deals with emergency situations. <u>Author</u>: Bruce Musico <u>SME</u>: Bruce Musico <u>Key Documents</u>: NUREG-1811, NUREG-1835

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			included as a part of the SER.	
94	SER 13-44	SER Section 13.3.3.11.3	The SER states that the Staff "applied current requirements" on Federal guidance relating to protective action recommendations (in the event of an accidental release of radioactivity). The Staff acknowledged that the Federal guidance may change and that "[a] COL or OL applicant should address any such changes, and the Staff will determine compliance with the requirements, in this area during a COL or OL review." The Board has the following questions related to this statement in the SER:	
95			A. Legal Question: Please explain how this statement in the SER comports with 10 CFR § 52.39(a)(1) which states that the "Commission may not impose new requirements, including new emergency planning requirements, on the early site permit or the site for which it was	See the "NRC Staff Legal Brief in Response to Licensing Board's Safety-Related Questions"

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			issued, unless the Commission determines that a modification is necessary either to bring the permit or site into compliance with the Commission's regulations and orders in effect at the time the permit was issued, or to assure adequate protection of the public health and safety or the common defense and security."	
96			B. Legal Question: Contrary to the statement in the SER, does 10 CFR § 50.39(a)(1) mean that the Applicant is immunized (grandfathered) against any more stringent regulatory requirements or guidance for up to 80 years (the term of the ESP (20 years) plus extensions (20 years) plus the term of any COL (40 years)) unless a change can be shown to be "necessary . . . to assure adequate protection of the public health and safety or the common defense and	See the "NRC Staff Legal Brief in Response to Licensing Board's Safety-Related Questions"

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			security?"	
97			C. Legal Question: The SER states, at page 13-49, that "the Staff did not consider the extent to which future radiological protection procedures would address radiological protection and onsite contamination control functions." Would the Applicant be exempt from these future procedures (unless they are shown to be necessary to assure adequate protection of public health and safety)? Please explain.	See the "NRC Staff Legal Brief in Response to Licensing Board's Safety-Related Questions"
			Accident Analysis	
99	SER 15-4	SER Section 15.3.1	What is the basis for the statement that the proposed Design Based Accidents for the ABWR and AP-1000 reactor designs would bound the DNBs for CANDU and gas-cooled reactors?	The basis for the Staff's belief that site acceptability based on the AP1000 and ABWR designs is likely to be valid for the other reactor designs is the Staff's review experience and understanding of the performance of the 700 MWe Advanced CANDU Reactor (ACR-700) design and the Fort St. Vrain gas-cooled reactor, discussed below. In June 2002, Atomic Energy of Canada Limited Technologies (AECL) requested that the NRC conduct a pre-application review of the ACR-700 design for licensing in the United States. During the pre-application review, AECL provided detailed information on the ACR-700 design, expected fission product behavior during and following DBAs, and information on their current experimental database applicable to ACR-700. In October 2004, the Staff issued the "Pre-Application Safety Assessment Report related to the Advanced CANDU Reactor 700 MWe" (PSAR) (ADAMS Accession number ML042110074). Even though the Staff did not review the proposed ACR-700 source term in detail, the Staff identified in the PSAR no issues related to the DBA source term that would

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				preclude certification of the ACR-700 design. With respect to gas-cooled reactors, the Staff issued an operating license for the 330 MWe Fort St. Vrain High Temperature Gas Cooled Reactor (HTGR) design in 1978. (The facility was permanently closed in 1989.) The Staff evaluated the radiological consequences of DBAs at a reactor power level of 879 MWt and its independent confirmatory dose calculation showed the resulting doses were well below the 10 CFR Part 100 dose guidance values. Finally, whether the CANDU and gas-cooled reactors are bounded by these DBA analyses (certified ABWR, proposed AP1000, and proposed ESBWR) would be verified during the Staff's review of any COL or CP application for a specific reactor design at the North Anna ESP site. Author: Jay Lee SME: Jay Lee Key Document: NUREG-1835, Supplement 1; 10 CFR Part 100; ACR-700 PSAR, and "Safety Evaluation" by United States Atomic Energy Commission in the matter of Public Service Company of Colorado, Fort St. Vrain Nuclear Generating Station, Docket No. 50-267 (June 21, 1968).
100	SSER 15-3, 15-9	SSER Section 15.1	The SSER states that the Applicant's response to Supplemental RAI 1 revealed that the highest 2-hour dose at the EAB for certain of the ESBWR DBAs does not occur in the first two hours. How did the Staff handle this fact in developing its proposed site specific χ/Q values in Table 15.3-1?	The 2-hour χ/Q value is calculated the same way whether it is to be used for the first 2 hours of a DBA or for a later period. The timing of the fission product release used for the DBA dose calculation (including the highest 2-hour dose at the EAB) was not used in developing the Staff's site-specific χ/Q values in Table 15.3-1. The DBA dose calculation considers both the fission products released to the environment and the dilution characteristics (or the χ/Q values), but they are determined separately. The site-specific χ/Q values are solely based on the site meteorological characteristics and the EAB and LPZ distances. Author: Jay Lee SME: Jay Lee, R. Brad Harvey, Barry Zalcman Key Document: NUREG-1835, Supplement 1
101	SSER 15-6	SSER Section 15.3.2	Given that the Applicant and Staff have each calculated the site	The "postulated χ/Q values in the certified ABWR DCD" or "the proposed χ/Q values for the AP1000 DCD" were used to obtain the ratios of the site-specific χ/Q values to the "postulated" design χ/Q values. The χ/Q ratios are then used

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			specific χ/Q values for this ESP site, why should the "postulated χ/Q values in the certified ABWR DCD" or the proposed χ/Q values for the AP1000 DCD be used?	along with the calculated doses in the certification documents to assess the suitability of the proposed ESP site for the AP1000 and ABWR design basis accidents. The certified ABWR and the proposed AP1000 designs met the radiological consequence evaluation factors in 10 CFR 50.34(a)(1) with their postulated χ/Q values. The resulting DBA radiological consequences at the proposed site would be lower than those provided in the certified ABWR DCD and the proposed AP1000 DCD, meeting the requirements of 10 CFR 50.34, if the site specific χ/Q values are lower than the proposed χ/Q values in respective DCDs (the χ/Q ratio of less than 1) . The radiological consequences are directly proportional to the χ/Q values (see the Staff's response to Board Question 102 below). Author: Jay Lee SME: Jay Lee Key Document: NUREG-1835, Supplement 1; 10 CFR Part 50
102	SSER 15-6	SSER Section 15.3.3	The SSER states that "Smaller χ/Q values are associated with greater dilution capability, resulting in lower radiological doses. The radiological consequences are thus inversely proportional to the χ/Q values." Don't you mean that they are directly proportional? Please explain.	The referenced statement in the SSER is incorrect. The radiological consequences are directly proportional to the χ/Q values. Author: Jay Lee SME: Jay Lee, R. Brad Harvey Key Document: NUREG-1835, Supplement 1
103	SSER 15-7	SSER Section 15.3.3	The SSER states that "the applicant provided a set of bounding reactor accident source terms as a set of PPE values." Please explain how the Staff knows that, in fact,	The reactor accident source terms are PPE values and they will be specified in any ESP that might be issued for the North Anna ESP site. The Applicant selected these source terms to be bounding values and concluded that the proposed site meets the radiological consequence evaluation factors identified in 10 CFR 50.34(a)(1) with these source terms, based on the guidance in RS-002, "Processing Applications for Early Site Permits." The Staff determined that the proposed PPE values were not unreasonable. The ultimate reactor accident

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			the Applicant's source terms are bounding. In this context, does "bounding" simply mean that, by definition, the ultimate reactor accident source terms in the COL must be within the PPE in order to comply with the ESP?	source terms used in a CP or COL application referencing an ESP must be within these PPE values for this issue to be considered resolved for the CP or COL proceeding. If they are not within these PPE values, the Staff will re-evaluate the radiological consequences at the CP or COL stage to determine whether the proposed facility indeed meets the radiological consequence evaluation factors identified in 10 CFR 50.34(a)(1). Author: Jay Lee SME: Jay Lee Key Document: NUREG-1835, Supplement 1; 10 CFR Part 50
104	SSER 15-8	SSER Section 15.3.5	The SSER states that the Staff "has verified the design specific source terms the applicant has provided." Please describe what the Staff did to verify these source terms.	The Staff verified the design-specific source terms provided by the Applicant by comparing them to those source terms specified in the certified ABWR DCD, the proposed AP1000 DCD, and the proposed ESBWR DCD, Revision 1, Tier 2. Author: Jay Lee SME: Jay Lee Key Documents: NUREG-1835, Supplement 1; ABWR DCD; AP1000 DCD; ESBWR DCD
105	SSER 15-9	SSER Table 15.3-1	The SSER states that the Staff intends to include the site-specific χ/Q values listed as site characteristics in Appendix A in any ES Table 15.3-1 includes a value for "4 to 30 day LPZ." Why is this value not included in Appendix A.3?	The "4–30 χ/Q value @ LPZ" was inadvertently omitted from Appendix A.3 of Supplement 1 to the FSER. This site characteristic will be included in any ESP issued for the North Anna ESP site. Author: R. Brad Harvey SME: R. Brad Harvey, Jay Lee Key Documents: NUREG-1835, Supplement 1, Table 15.3-1 and Appendix A.3
106	SSER 15-9	SSER Section 15.3.5	The SSER states that the Applicant calculated the radiological consequences at the	The Staff verified the radiological consequences at the EAB and LPZ boundary calculated by the Applicant by performing independent confirmatory dose calculations. In performing its dose calculations, the Staff used the ESBWR source term provided by the Applicant, the site-specific χ/Q values listed in Table

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			EAB and LBZ boundary based on the ESBWR source term and X/Qs and that the results obtained by the Applicant are below the TEDE doses specified in 10 CFR § 50.34(a)(1). Please describe what the Staff did to verify the Applicant's calculations.	15.3-1 of the SSER, breathing rates for the "standard man" in International Commission on Radiation Protection Publication II (1959), and dose conversion factors in Federal Guidance Report 11, "Limiting Values of Radionuclide Intake And Air Concentration and Dose Conversion Factors for Inhalation, Submersion, And Ingestion (1988)," U.S. Environmental Protection Agency and Oak Ridge National Laboratory, and in Federal Guidance Report 12, "External Exposure To Radionuclides In Air, Water, And Soil (1993)," U.S. Environmental Protection Agency and Oak Ridge National Laboratory. Author: Jay Lee SME: Jay Lee, Van Ramsdell Key Documents: NUREG-1835, Supplement 1; ESBWR DCD
107	SSER 15-9	SSER Table 15.3.-1	Why are the dispersion factors in Table 5-14 in the Draft EIS different from the dispersion factors in Table 15.3-1 in the SSER? [Board's example not reproduced here.]	The NRC performs dose calculations to meet a variety of different statutory and regulatory objectives. Reactor designs have to meet the safety objectives under the Atomic Energy Act related to NRC's health and safety missions; as result, the DBA accident analysis is performed assuming "adverse" or "conservative" atmospheric dilution conditions (i.e., χ/Q values) of 5^{th} percentile values (see the Staff's response to Board Question 17). These are the values listed in the Staff's safety evaluation report in Table 15.3-1. In addition, to fulfill its responsibilities under the National Environmental Policy Act, the NRC must evaluate the "reasonably foreseeable" impacts on the environment. Consequently, the Staff evaluates accidents assuming that the accident would occur under "typical," "representative," or "reasonable," atmospheric dilution conditions of 50^{th} percentile values. These are the values listed in the Staff's environmental impact statement in Table 5-14. The set of values are different because the statutory and regulatory objectives are different; however, they are based on the same set of meteorological data. Author: Jay Lee SME: Jay Lee, Van Ramsdell Key Documents: NUREG-1835, Supplement 1; NUREG-1811
			General Questions	
108	Responses to RAIs (one of nine).		Some RAIs posed complex questions that did not always appear to be completely addressed	The Staff requires an applicant to provide the information necessary for the Staff to make its findings, as guided by RS-002. In some cases, facts that come to light between the time that the Staff issues a question and the time that the applicant responds may render the Staff's question academic. In such cases,

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			in the response. Two examples are RAI 3.8-9, which addresses the increase in neutron dose from the Gas Cooled Pebble Bed reactor, and E 3.8-16, which requests in-core differences in LWRs and Advanced Reactors with respect to seven cited features. What actions would NRC typically take to obtain the information requested?	the Staff generally would not pursue an answer to its original question. In all cases, however, the Staff will continue to pursue answers to any questions to which answers are required in order to meet its regulatory obligations. Author: George Wunder SME: George Wunder Key Documents: RS-002
109	General	General	If a plant is built that derives from the current ESP, are there any regulatory repercussions if actual release rates and doses exceed the values approved in the ESP?	The ESP does not approve release rates or dose values. The ESP will contain a source term for accidents, which any future CP or COL applicant referencing the ESP must show to be bounding. Routine radioactive releases and allowable doses from normal operations are governed by the 10 CFR Part 20 and 10 CFR Part 50, Appendix I, and are addressed under the Agency's Reactor Oversight Process for operating reactors. Author: George Wunder SME: George Wunder Key Documents: n/a
110	General	General	Does NRC regularly check actual routine releases from nuclear plants against the claimed releases in applications or licenses?	Yes, the NRC regularly checks actual routine releases from nuclear power plants as part of its Reactor Oversight Process. Requirements in each power reactor's license include compliance with effluent radionuclide concentration limits of Appendix B, Table 2 to 10 CFR Part 20; 10 CFR 20.1301 and 20.1302 dose limits to members of the public; 10 CFR 20.1301(e) with respect to EPA's 40 CFR Part 190 environmental radiation standards; and ALARA dose criteria of Appendix I to 10 CFR Part 50. These requirements are implemented through operational programs and procedures mandated by 10 CFR 50.36a, 10 CFR 50.34a, and Section IV of Appendix I to 10 CFR Part 50. The key operational program documents are RETS or SREC, ODCM, and the REMP. The implementation of these programs and license conditions is routinely inspected

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				by NRC inspectors. These inspections examine the licensee's radiological effluent monitoring and release programs to ensure its programs meet NRC requirements and license conditions. Thus, the data and inspection reports support the conclusion on whether a plant meets NRC's ALARA criteria. If a plant were to exceed the ALARA objectives or any other requirements of Part 20, the inspection would identify the cause and determine whether a proper response and corrective actions were taken by the licensee. Under the provisions of 10 CFR 20.1301(f), the NRC may impose additional restrictions after evaluating the impacts on members of the public in light of commitments and characterizations contained in the CP or COL application. Author: Jean-Claude Dehmel SME: Jean-Claude Dehmel Key Documents: 10 CFR Part 20, 10 CFR Part 50, 40 CFR Part 190
111	SER xiii	SER Exec. Sum.	The Part 52 ESP process is intended to address and resolve site-related issues. The SER serves to identify matters resolved in the Staff's safety review and to identify remaining items to be addressed in a later proceeding (CP, COL, or a design certification proceeding). Many site issues are not resolved because they are related to final design or are simply put off to later licensing actions. Might it be assumed that only those issues resolved in the SER, FEIS, and Commission rulings and decisions will be	The Staff views the ESP as resolving all issues of site suitability. Some issues relating to site suitability, however, can be addressed through placing permit conditions on the design of the proposed facility. To the extent that the Staff has determined that such an issue is appropriately addressed by restrictions on the design of the proposed facility, the Staff has proposed permit conditions to ensure that the issue is addressed by any future CP or COL applicant. Additional design matters are identified in COL Action Items, but these items are information requirements only, and do not defer any determinations necessary for the ESP determination concerning site suitability. Accordingly, no carryover site-related safety issues remain to be flagged. Author: George Wunder SME: George Wunder Key Documents: 10 CFR Part 52

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			considered resolved for purposes of future hearings? If this is not the case, how are carryover site-related issues flagged for handling if they are not listed in an action file such as the COL Action Item list?	
112	SER 2-88	SER Section 2.4.4.3	COL Action Item 2.4-6 requires that an Applicant should demonstrate that the UHS reservoirs are designed to satisfy the NRC's regulations. The NRC Staff says that the detailed design of underground UHS reservoirs is not within the scope of ESP review. Is it true that an ESP permits some site work including the possible construction of cooling towers? If so, could this issue and COL Action Item 2-4-7 fall between the cracks?	An ESP authorizes site preparation and limited construction activities on the site, provided an Applicant submits, and the NRC approves, a site redress plan with its ESP application (see 10 CFR 52.17(c); 50.10(e)(1)). Such activities can then be carried out at the site prior to submission of a CP or COL application. Excavation for a safety-related structure is currently permitted as part of the site preparation and limited construction activities enumerated in 10 CFR 50.10(e)(1). Further, construction of non-safety-related cooling towers would be permitted pursuant to 10 CFR 50.10(e)(1). However, because construction of UHS reservoirs would be considered safety-related construction, such construction would not be permitted as part of the site preparation and limited construction activities. The Applicant has submitted a site redress plan, which the Staff has recommended be approved if an ESP is issued. All proposed COL Action Items, including 2.4-7, would be identified in an Appendix to the ESP, and must be addressed in any CP or COL application referencing that ESP. <u>Author</u>: Goutam Bagchi <u>SME</u>: Goutam Bagchi, George Wunder <u>Key Documents</u>: 10 CFR 52.17(c); 10 CFR 50.10(e)(1).
113	SER 2-89	SER Section	COL Action Item 2.4-7 concerns the adequacy	Adequacy of the water supply for the UHS will be assessed by the Staff in connection with any CP or COL application referencing the ESP (provided the

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		2.4.4.3	of the remaining liquid volume stored in the UHS. How do you determine what is adequate or are you saying that that determination be made and incorporated into a plan of action?	ESP is approved). The Staff's review is governed by SRP Section 9.2.5, "Ultimate Heat Sink." Author: Goutam Bagchi SME: Goutam Bagchi Key Documents: NUREG-0800, Section 9.2.5, "Ultimate Heat Sink"
114	SSER A-7 to A-9	SSER Section A-2	Why aren't the following Action Items identified for a COL application?	
115			A. Radiation exposures to construction personnel should be reevaluated in light of the specific steam supply system chosen. A projected person-rem exposure of 120 person-rem/yr. gives some likelihood of adverse health effects when projected over the entire construction cycle. See Section 4.9.4 of NUREG 1811.	Item 115A has been identified as an issue that must be addressed in a CP or COL application. CP or COL applicants referencing a certified design are required to provide estimated annual doses to construction workers in a new unit construction area as a result of radiation sources from nearby existing operating plants. DG-1145 states, "For multi-unit plants, provide estimated annual doses to construction workers in a new unit construction area, as a result of radiation from onsite radiation sources from the existing operating plant(s)." Section 4.5, Radiation Exposure to Construction Workers, of NUREG-1555 provides guidance to the Staff on the analysis and assessment of potential radiological impacts on the proposed project construction work force. The radiation that the construction personnel would be exposed to during construction of the new units (Units 3 and 4) at North Anna would be the radiation from routine operation of the existing units (Units 1 and 2). Of the three sources of radiation (direct radiation, radiation from gaseous effluents, and radiation from liquid effluents) that construction workers may be exposed to from the existing operating units, the primary dose contributor to construction worker dose at North Anna would be from direct radiation. Since the radiation sources impacting the construction personnel are from the existing operating two units at North Anna, there is no need to reevaluate the radiation exposures to construction personnel in light of the specific steam supply system chosen. In evaluating the radiation exposures to construction personnel, Dominion

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				estimated the direct radiation exposure to construction workers by using TLD measurements of direct radiation levels at the North Anna Unit 1 and 2 protected area fence line. In order to determine the most representative dose rate that construction personnel of Units 3 and 4 would be exposed to, Dominion used the average annual dose rate measurements over the 7-year period from 1996 through 2002 for the TLD location on the west protected area fence closest to the proposed site for Units 3 and 4. The average annual dose rate for all the TLD readings at this location for the 7-year period was 56 mrem/year. Adjusting the TLD exposure time to 2080 hours/year, which is the estimated maximum time per year that a worker would be exposed, Dominion calculated an annual dose to a construction worker of 13 mrem/year from this component of direct radiation. Since the construction workers for Units 3 and 4 would be expected to spend most of their time several hundred feet west of this TLD location, the annual direct dose that they would receive from Units 1 and 2 would be less than this annual dose estimate. The annual dose estimate of 13 mrem/year includes a dose contribution based on the direct dose from the nearby Independent Spent Fuel Storage Installation (ISFSI). However, this dose contribution is based on the ISFSI loading at the time of the TLD measurements. In order to provide a more conservative dose estimate from the ISFSI, Dominion calculated an additional dose component to the construction workers assuming that the ISFSI was fully loaded during the construction period. Correcting for an occupancy rate of 2080 hours/year, the dose contribution to construction workers for Units 3 and 4 from a fully loaded ISFSI would be 9.8 mrem/year. Adding the two direct dose contributions, the estimated direct dose contribution to construction workers for Units 3 and 4 is estimated to be 23 mrem/year. The dose contribution from gaseous and liquid effluents is expected to add an additional 1 mrem/year to the construction worker dose, for a total annual construction worker dose from direct radiation and gaseous and liquid effluents of 24 mrem/year. This annual dose estimate for Unit 3 and 4 construction workers of 24 mrem/year is well below the 10 CFR 20.1301 dose limit for individual members of the public of 100 mrem in a year. This annual dose estimate is also well below the 10 CFR 20.1201 occupational dose limit for adults of 5 rem/year.

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				The Staff has not estimated or calculated accident doses to construction workers at the proposed ESP site resulting from a design basis accident (DBA) at North Anna Units 1 and 2 since construction workers will be evacuated at onset of a DBA in accordance with the North Anna Site Emergency Plan. Supporting references: a. DG-1145, "Combined License Applications for New Power Plants (LWR Edition)," draft issued June 30, 2006, p. C.III.1-125. b. NUREG-1555, "Environmental Standard Review Plan," October 1999, pp. 4.5-1 – 4.5-8. c. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," November 1978, p. 12-6. <u>Author</u>: Charles Hinson <u>SME</u>: Charles Hinson, Jay Lee <u>Key Documents</u>: 10 CFR 20.1301, 10 CFR 20.1201, DG-1145, NUREG-1555, RG 1.70
			B. The impact of localized fogging on transportation accidents should be evaluated.	As discussed below, the Staff addressed the impacts of fogging and icing in Volume 2 of the FEIS. Comments on page 3-12 and the associated response on page 3-13 in Volume 2 of the FEIS address this issue, as follows: **Comment:** [T]he cooling towers that you have cooling the air temperature. What is that going to do from a thermal heat pollution [sic] to the atmosphere? (ST-0001 3) **Comment:** These cooling towers will emit plumes of steam fog formation, which can create fog-icing conditions in the vicinity an average of 70 hours per year (or if three hours per day this equates to 23 extra days of year of fog and/or icing condition on the adjoining roadways)...What type of mitigation can be done to avoid any traffic problems on adjoining roadways?(SW-0005 7)

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				Comment: What type of mitigation can be done to avoid any traffic problems on adjoining roadways as a result of the fog and icing conditions approximately an extra 23 days a year? (SE-0003 7) **Comment:** We're concerned about ...The impact of additional fog and icing from wet cooling towers on local roadways. (ST-0014 17) **Comment:** [To ensure that the proposed construction of a 3rd & 4th reactor will minimize the adverse affect to the quality of life for those that live and use Lake Anna, we also ask that you further evaluate the following concerns prior to your making a final decision on the ESP]…Impact of additional fog and icing from wet cooling towers on local roadways. (SE-0022 29) **Response:** *Operation of the cooling towers in the wet mode would release warm, moist air into the atmosphere creating elevated plumes. As the plumes lose buoyancy and reach ground level, there is a potential for fogging and icing. The staff estimates the maximum hours of fogging to be 70 hours per year beyond naturally occurring fog. No icing is expected to occur. A majority of the estimated fogging would occur within 1000 ft south-southeast from the cooling towers, but could extend as far as 5200 ft. Fogging was estimated to occur in all seasons except summer, but primarily in late fall and winter. During the time the cooling towers operated in the dry mode (which would result in no additional adverse impact and could result in a slight improvement), the atmosphere, rather than Lake Anna, would be the sink for the waste heat. This would not lead to increased ground fog in or around the site area or in the vicinity of the roadways.* As noted above, the Staff's review indicated that additional icing would not occur as result of the additional fogging. A vast majority of the additional hours of fogging will be expected to occur near the cooling tower location inside the plant

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				boundary. The standard onsite mitigation strategy is to restrict the potential impacted area from employee parking and transportation routing. Based on the data from ER table 5.3-26, the Staff estimated that there would be less than an hour of fogging per year on Route 652 with some which would typically occur somewhere between midnight and 3 in the morning. The Staff determined that this fogging is sufficiently infrequent that further mitigation is not warranted. Author: William Sandusky SME: Jack Cushing, R. Brad Harvey Key Documents: NUREG 1811, Volume 2.
			C. The potential release paths of radioactivity into the environment during normal operation should be established and evaluated.	For impacts to members of the public and the environment associated with normal gaseous and liquid effluents, multi-unit plants typically have separate radioactive waste storage tanks and systems, components, and discharge points in order to control, monitor and document the types and amounts of radioactive effluents discharged in the environment from each reactor unit. The standard NRC Technical Specifications for normal radiological gaseous and liquid effluents have controls that are defined on a unit-specific basis. At the ESP stage, applicants do not provide detailed information on the design features of process and effluent treatment systems, and numbers and locations of effluent release points to the environment. Under the PPE concept, the estimates of airborne radioactive materials discharged in the environment represent total aggregates without specifying fractional releases by buildings or vents, such as plant stack, spent fuel building, radwaste building, etc. For liquid effluents, the release point is described as an outlet into the discharge canal, again without describing the specific origins of waste streams from specific subsystems and fractional distributions of radioactivity and radionuclides from various portions of radwaste or liquid waste processing systems. Thus, at the ESP stage, the applicant provides a bounding estimate of the amounts of radioactivity that are expected to be released from all systems under normal operations and anticipated operational occurrences. At the CP or COL stage, however, the Staff will have specific details on the applicant's reactor design, radioactive waste processing systems, number and locations of effluent release points, dilution factors before and after the point of discharge, updated information on offsite exposure pathways and dose receptors, and actual distances from effluent discharge points to offsite dose

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				receptors. Such detailed information will provide the means for the Staff to perform independent verification of radioactive effluent source terms and doses to members of the public. This issue is addressed in COL Action Item No. 11.1-1. Author: Jean-Claude Dehmel SME: Jean-Claude Dehmel Key Documents: NUREG-1835
			D. The procedures and equipment to be used to maintain tritium releases and concentrations below EPA limits should be defined.	For the ESP review, the Staff was asked to review the applicant's data using a PPE concept. The PPE does not contain detailed information on plant systems and components that will be used to treat radioactive liquid and gaseous process and effluent streams sent to radioactive waste reduction systems for the Staff to validate the applicant's assumptions. The applicant commits to design, install, operate, and maintain gaseous, liquid, and solid waste processing systems that will keep effluent releases within 10 CFR Part 20, Appendix B, Table 2 effluent concentration limits; 10 CFR 20.1301 and 20.1302 dose limits to members of the public; 10 CFR 20.1301(e) with respect to EPA's 40 CFR Part 190 environmental radiation standards; and 10 CFR Part 50, Appendix I ALARA dose objectives. These commitments are applicable for all releases and all forms of radioactivity. The types of radionuclides expected typically include tritium, and those characterizing radio-iodines, noble gases, alkali, tellurium, cerium, noble metals, lanthanides, and activation and fission products. For tritium, however, the amounts or concentration levels discharged are primarily dependent on the type of plant (Ref. a - d). For BWRs, tritium concentrations in reactor water and steam are controlled by the loss of water from the main coolant system due to evaporation or leakage and by the production rate of tritium due to activation of deuterium in the coolant and rate of release from fuel assemblies. For PWRs, the concentrations of tritium are primarily a function of the inventory of tritiated liquids, the production rate of tritium due to boron, lithium, and deuterium activation in primary coolant, the rate of release from fuel assemblies, and the extent to which tritiated water is recycled or released from the plant. For either type of plant, there are no waste treatment technologies being used to singly extract tritium from liquid and gaseous effluents. At the CP or COL stage, the Staff will have specific details on the applicant's reactor design, radioactive waste processing systems, effectiveness of radwaste

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				processing methods or equipment (decontamination factor or removal efficiencies, etc.), locations of effluent release points, and concentrations of effluents. Such detailed information will provide the means for the Staff to perform independent verification and calculations of radioactive source terms and assess doses to members of the public. The responsibility rests on the CP or COL applicant to describe the type of technology that will be used and maintained to meet the above regulatory requirements and license conditions. This issue is subsumed by COL Action Item No. 11.1-1. Once a plant is operational, these requirements are implemented through operational programs and procedures mandated by 10 CFR 50.36a, 10 CFR 50.34a, and Section IV of Appendix I to 10 CFR Part 50. The key operational program documents are the RETS or SREC, ODCM, and the REMP. The licensee is required to characterize the presence of radioactivity in liquid and gaseous effluents, assess doses to members of the public, and conduct environmental radiological surveys to confirm that radioactivity levels in environmental media and associated doses comply with NRC and EPA dose limits at dose receptor locations, taking into account site specific exposure pathways identified using the results of annual land-use censuses. The implementation of these programs and license conditions is routinely inspected by NRC Regional Inspectors. Supporting references: a. ANSI/ANS-18.1-1999, “Radioactive Source Term for Normal Operation of Light Water Reactors, 1999. Tables 8 and 9 footnotes on tritium. b. NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactor, BWR-Gale Code," Rev. 1, January 1979. Section 2.2.15, p. 2-43. c. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactor, PWR-Gale Code," Rev. 1, April 1985. Section 2.2.17, p. 2-68. d. Karl-Heinz Need, “The Radiochemistry of Nuclear Power Plants with Light Water Reactors,” Walter de Gruyter, Berlin, New York, 1997.

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				Section 4.2.3. Author: Jean-Claude Dehmel SME: Jean-Claude Dehmel Key Documents: 10 CFR Part 20, 10 CFR Part 50, 40 CFR Part 190, SER, ANSI/ANS-18.1-1999, NUREG-0016, NUREG-0017
			E. Specified allowed soil settling rates should be readdressed in light in of subsoil compositions identified for the COL.	As described above in the Staff's response to Board Question 68, COL Action Item 2.5-6 states that an applicant should provide an analysis of the stability of all planned safety-related facilities, including bearing capacity, rebound, settlement, and differential settlement under dead loads of fills and plant facilities, as well as lateral loading conditions. In addition, the relationship between building foundations and underlying materials is also addressed under COL Action Item 2.5-2. Section 2.5.4.3 of NUREG-0800 and RS-002 direct the Staff to compare the applicant's plot plans and the profiles of safety-related facilities with the subsurface profile and material profiles. Based on this comparison, the Staff can determine if (1) the applicant performed sufficient exploration of the subsurface and (2) the applicant's foundation design assumptions contain adequate margins of safety. The Applicant decided to defer providing this information until a CP or COL application is submitted. Author: Clifford Munson SME: Clifford Munson Key Documents: NUREG-0800, RS-002
116	SSER viii	SER Exec. Sum.	Appendix A is described as "certain site-related items that an applicant will need to address at the combined license or construction stage" and that "these items . . . are more appropriately addressed at later stages."	

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			A. Legal Question: Does Appendix A run afoul of 10 CFR § 52.39(a)(1), which states that an ESP is final and that thereafter "the Commission may not impose new requirements . . . on the site?" Please provide legal support and analysis.	See the "NRC Staff Legal Brief in Response to Licensing Board's Safety-Related Questions"
			B. Legal Question: How does the quoted provision comport with the Commission's refusal, when it promulgated the ESP regulations, to condone the issuance of "partial" ESP permits. See 54 Fed Reg. 15372, 15378 n.3 (April 18, 1989) ("the Commission declines to follow the suggestion . . . that partial early site permits be issued."). By incorporating so many items to be determined later, isn't the Staff proposing a "partial ESP?"	See the "NRC Staff Legal Brief in Response to Licensing Board's Safety-Related Questions"
			C. Legal Question: How does this provision comport with the Commission's statement	See the "NRC Staff Legal Brief in Response to Licensing Board's Safety-Related Questions"

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			that "[w]here adequate information is not available, early site permits will not be issued?" 54 Fed Reg. at 15378 n.3.	
			D. Legal Question: Are all of these matters unresolved within the meaning of 10 CFR § 52.39(a)(2). If not, why not?	See the "NRC Staff Legal Brief in Response to Licensing Board's Safety-Related Questions"
			E. Legal Question: Will a petition alleging that the site or Applicant is not in compliance with a permit conditions, COL action item, site characteristic, or bounding parameter specified in Appendix A be within the scope and litigable (provided it meets the other criteria of 10 CFR § 2.309(f)(2)) at the COL stage?	See the "NRC Staff Legal Brief in Response to Licensing Board's Safety-Related Questions"

Attachment A: North Anna ESP Safety Inquiries

Table 1 (Board Questions 17, 20, and 21)

North Anna ESP EAB and LPZ Accident X/Q Values					
Exceedance Probability	**EAB**	**LPZ**			
	0-2 hours	**0-8 hours**	**8-24 hours**	**1-4 days**	**4-30 days**
Maximum Calculated Value	9.21E-04	4.63E-05	2.83E-05	9.76E-06	2.11E-06
1%	2.94E-04	2.50E-05	1.63E-05	6.40E-06	1.67E-06
Chosen Site Characteristic Value	**2.26E-04**	**2.05E-05**	**1.36E-05**	**5.57E-06**	**1.55E-06**
3%	1.99E-04	1.73E-05	1.17E-05	4.97E-06	1.46E-06
5%	1.59E-04	1.33E-05	9.21E-06	4.15E-06	1.32E-06

Table 2 (Board Question 81)

Type of Radioactive Effluents	Dose (person-rem per year)	
	Each new unit	Both new units
Liquid	8.6	17
Noble gases (Gaseous)	3.5	7
Iodines and particulates (Gaseous)	1.4	2.8
H-3 and C-14 (Gaseous)	14	29
Total	28	56
Natural background	920,000	920,000