VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

October 25, 2018 10CFR50.90

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Serial No.: 18-069A
NRA/GDM: R0
Docket Nos.: 50-280
50-281
License Nos.: DPR-32
DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
PROPOSED LICENSE AMENDMENT REQUEST
TSTF-490 AND UPDATE OF ALTERNATIVE SOURCE TERM ANALYSES
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION AND REVISED PROPOSED TECHNICAL SPECIFICATIONS PAGES

By letter dated March 2, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML18075A021), Virginia Electric and Power Company (Dominion Energy Virginia) requested changes to the Technical Specifications (TS) for Surry Power Station Units 1 and 2 (SPS). The proposed changes would adopt TS Task Force (TSTF) Traveler, TSTF-490, Revision 0, "Deletion of E Bar Definition and Revision to RCS [reactor coolant system] Specific Activity Tech Spec," dated September 13, 2005 (ADAMS Accession No. ML052630462) and would update the Alternative Source Term (AST) analyses. The proposed changes would replace the current limits on primary coolant gross specific activity, which is based on E-Bar ($\bar{E}$) average disintegration energy, with a new limit for RCS noble gas specific activity. The noble gas specific activity limit would be based on a new dose equivalent Xenon-133 definition that replaces the current $\bar{E}$ average disintegration energy.

By email dated September 25, 2018, the Nuclear Regulatory Commission (NRC) staff provided a request for additional information (RAI) to complete their review. The NRC request and Dominion Energy Virginia's response are provided in Attachment 1 to this letter. In preparing this response, we determined the license amendment request (LAR) as originally proposed requires revision to address NRC RAI questions ARCB-RAI-1 and ARCB-RAI-2. The required revision is discussed in Attachment 1, and the revised, marked-up and typed proposed TS and TS Bases pages are provided in Attachments 2 and 3, respectively. The proposed marked-up and typed TS and TS Bases pages are being provided in their entirety with the revisions made as a result of the RAI responses indicated by double revision bars in the right hand margin of the revised pages. The associated TS Basis changes are included for information.

The information provided in this letter does not affect the conclusions of the significant hazards consideration or the environmental assessment included in the March 2, 2018 LAR.

A001
NRR

Should you have any questions or require additional information, please contact Mr. Gary D. Miller at (804) 273-2771.

Respectfully,

Gerald T. Bischof
Senior Vice President – Nuclear Operations & Fleet Performance

Commitments contained in this letter: None

Attachments:

1. Response to NRC Request for Additional Information, Proposed Changes Associated with TSTF-490 and Updated Alternative Source Term Analyses

2. Revised Marked-up Technical Specifications and Bases Pages

3. Revised Proposed Technical Specifications and Bases Pages

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Mr. Gerald T. Bischof, who is Senior Vice President – Nuclear Operations & Fleet Performance of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 25th day of October, 2018.

My Commission Expires: Jan 31 2022.

Cheryl Ann Kelly
Notary Public

CHERYL ANN KELLY
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My Commission Expires Jan 31, 2022

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NRC Senior Resident Inspector
Surry Power Station

Attachment 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

PROPOSED CHANGES ASSOCIATED WITH TSTF-490 AND UPDATED ALTERNATIVE SOURCE TERM ANALYSES

Virginia Electric and Power Company
(Dominion Energy Virginia)
Surry Power Station Units 1 and 2

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION PROPOSED CHANGES ASSOCIATED WITH TSTF-490 AND UPDATED ALTERNATIVE SOURCE TERM ANALYSES

SURRY POWER STATION UNITS 1 AND 2

NRC Comment:

By letter dated March 2, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML18075A021), Virginia Electric and Power Company (Dominion Energy Virginia) the licensee, requested changes to the Technical Specifications (TS) for Surry Power Station Units 1 and 2 (SPS). The licensee requested to adopt Technical Specifications Task Force (TSTF) Traveler, TSTF-490, Revision 0, "Deletion of E Bar Definition and Revision to RCS [reactor coolant system] Specific Activity Tech Spec," dated September 13, 2005 (ADAMS Accession No. ML052630462) and update the Alternative Source Term (AST) analyses. The proposed changes would replace the current limits on primary coolant gross specific activity which is based on E-Bar ($\bar{E}$) average disintegration energy with a new limit for RCS noble gas specific activity. The noble gas specific activity limit would be based on a new dose equivalent Xenon-133 definition that replaces the current $\bar{E}$ average disintegration energy.

During the Nuclear Regulatory Commission (NRC) staff's review of the license amendment request the NRC staff determined that more information was needed to complete the review.

Regulatory Analysis Basis

1. Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.67, "Accident Source Term," allows licensees seeking to revise their current accident source term in design basis radiological consequence analyses to apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report. Section 50.67(b)(2) requires that the licensee's analysis demonstrates with reasonable assurance that:

    (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

    (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

2. NUREG-0800, standard review plan (SRP) Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms," Revision 0, July 2000.

3. Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792).

ARCB-RAI-1

In the license amendment request, the licensee proposed TS changes to revise TS 3.1.D applicability requirements to specify that TS 3.1.D is applicable whenever average reactor coolant system temperature exceeds 200°F. In addition, the licensee also proposed to revise Note (5) of TS Table 4.1-2B, "Minimum Frequencies for Sampling Tests," to state, "Only required when the unit is in POWER OPERATION," thus not requiring the surveillance requirements to be performed whenever TS 3.1.D is required.

The proposed change revises the conditions for sampling, and may exclude sampling during the plant conditions where TS 3.1.D may be exceeded. After transient conditions (i.e., reactor trip, plant depressurization, shutdown or startup) that end in REACTOR CRITICAL, HOT SHUTDOWN, or INTERMEDIATE SHUTDOWN, the surveillance requirements are not required to be performed. Isotopic spiking and fuel failures are more likely during transient conditions than during steady state plant operations.

Because TS 3.1.D could potentially be exceeded after plant transient or power changes, please justify why sampling is no longer needed in the reactor operations that are proposed to be eliminated and justify how the TS 3.1.D remains consistent with the design bases analysis from which the TS limits are derived (i.e., main steam line break, steam generator tube rupture, etc.). Furthermore, please justify why there is an apparent disparity between the TS applicability (average reactor coolant system temperature exceeds 200°) and the limited mode (POWER OPERATION) under which the surveillance is required.

Dominion Energy Virginia Response

The proposed revision to Note (5) of TS Table 4.1-2B, "Minimum Frequencies for Sampling Tests," to state, "Only required when the unit is in POWER OPERATION," is no longer being requested. The intent of the proposed Note (5) revision was to permit unit startup to POWER OPERATION prior to performing the associated sampling tests since a main steam line break or steam generator tube rupture would not be expected to occur during startup conditions. However, as noted in the RAI question, after

transient conditions that end in REACTOR CRITICAL, HOT SHUTDOWN, or INTERMEDIATE SHUTDOWN, isotopic spiking and fuel failures are more likely to occur than during steady state plant operations. Consequently, a new Note (4) has been specified for the DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 sampling test frequencies and defined in the table as requiring the sampling test requirements to be performed "Whenever T_{avg} (average RCS temperature) exceeds 200°F," which is consistent with the TS 3.1.D applicability and addresses transient conditions.

ARCB-RAI-2

In the license amendment request, the licensee proposes to revise Note (5) of TS Table 4.1-2B to state, "Only required when the unit is in POWER OPERATION," as discussed above in ARCB-RAI-2. However, Note (5) applies to more than the proposed DEI and DEX surveillance requirements, it also applies to the Radio-Chemical Analysis and the Tritium Activity tests. TSTF-490 only changed the surveillance requirements associated with DEI and DEX, it did not include any other surveillance requirement changes and it did not provide a technical basis for other surveillance changes.

Therefore, provide a technical justification that explains why the surveillance requirements for the Radio-Chemical Analysis and the Tritium Activity tests should be reduced from "When reactor is critical and average primary coolant temperature ≥ 350°F," to "Only required when the unit is in POWER OPERATION."

Dominion Energy Virginia Response

It was not recognized during the development of the License Amendment Request (LAR) that revised Note (5) was also applicable to the Radio-Chemical Analysis and Tritium Activity sampling tests. As noted in the response to ARCB-RAI-1 above, the proposed revision to Note (5) of TS Table 4.1-2B is no longer being requested. As such, Note (5) will remain as currently written; therefore, the Radio-Chemical Analysis and the Tritium Activity sampling tests are no longer affected by the proposed LAR.

ARCB-RAI-3

In the license amendment request, the licensee states in the TS bases for the dose equivalent Xe-133 surveillance requirement that the measurement is the sum of the degassed gamma activities and gaseous gamma activities in the sample taken or equivalent sampling method. Equivalent sampling methods were not discussed in TSTF-490 and were not reviewed or approved by the NRC staff. The addition of this wording is a variation from TSTF-490. Therefore, provide in detail the sampling methods that will be used at SPS and explain how they are equivalent to those discussed in TSTF-490.

Dominion Energy Virginia Response

RCS radioactive gas analysis can be performed using a stripped gas testing rig that physically strips the gas from the liquid phase by pumping air through the sample in a closed loop. However, this method of analyzing stripped gas is cumbersome and time consuming and results in higher dose exposure for the chemistry technicians testing the samples. Consequently, the SPS Chemistry Department uses a simpler yet equally effective method of analyzing radioactive RCS gases. Specifically, SPS uses a pressurized U-tube sample rig for obtaining optimum radioactive gas test results.

The pressurized gas rig consists of thirty centimeters of 3/8" stainless steel tubing with a Swagelok valve installed at each end. The tubing is rated for 6,500 psig and is therefore suitable for withstanding full RCS pressure. The tubing is bent into a U shape for optimal placement on the Canberra multichannel analyzer. The total volume of the U-tube is 11.3 milliliters (ml) including the valves. The samples are delayed by 45 minutes prior to counting to effect lower background radiation. The samples are also counted for 30 minutes, as opposed to the 10-minute counting time used in the gas stripping method, to increase the sensitivity of the U-tube method.

The U-tube rig was evaluated to ensure reliable and accurate results could be obtained. A comparison of the results between the stripped gas method and the pressurized U-tube method of analysis was performed to ensure consistent results were obtained between the two methods.

As stated in the EPRI Fuel Reliability Monitoring and Evaluation Handbook, Revision 2 (2010), page 6-2,

> "A few sites count the pressurized sample directly. This has the advantage of both simplicity and a short decay period for short-lived nuclides. The disadvantages are a complex spectrum and reduced sensitivity for lower energy gamma rays due to gamma ray adsorption in the water and wall of the sample cylinder. This is particularly true for the 81 keV gamma energy of Xe-133. (The 0.093 inch wall of a 1800 psig rated D.O.T cylinder will absorb 67% of the incident 81 keV photons.)"

Consequently, one of the primary concerns noted in the EPRI guidance associated with using a pressurized sample method, such as the U-tube test rig, is the attenuation of Xe-133 due to its low energy gamma (81 KeV). Since Xe-133 has the lowest energy gamma of the gas isotopes of concern, it was used for the primary comparison between the two analysis methods for sensitivity and measurement accuracy. One item that was examined was the minimum detectable amount (MDA) value of the stripped gas analysis method versus the U-tube analysis method. The results of a Unit 2 sample comparison of the MDA of Xe-133 using the stripped gas analysis and the U-tube analysis methods was 4.13E-4 versus 3.46E-4 μCi/ml, respectively. Therefore, the U-tube analysis MDA was lower than the stripped gas analysis MDA, which indicates the U-tube analysis method is more sensitive to Xe-133 than the stripped gas method.

A Unit 1 sample was also taken using each analysis method and the results compared to ensure Xe-133 was being accurately measured when above MDA values. The result using the stripped gas method was 1.86E-3 μ/ml and the U-tube method was 1.31E-3 μ/ml. This difference between the two values was determined to be an acceptable variation in the sample results between the two methods as it was consistent with the variations seen between sample results that were run on a weekly basis using just the stripped gas analysis method.

A comparison of the other remaining radioactive gas isotopes was also performed and, while there were small differences noted in the samples, they were likewise determined to be acceptable and consistent with variations between samples run on a weekly basis using only the stripped gas analysis method.

In summary, the pressurized gas U-tube analysis method is simpler to perform, results in reduced dose to the sampling technicians, and provides sample results that are consistent with those obtained using the stripped gas method and is therefore an acceptable and equivalent sampling method.

ARCB-RAI-4

In the license amendment request, the proposed change deletes the current gross activity, $\bar{E}$ determination, and radio iodine analysis tests and replaces them with a new surveillance to verify DEX is less than the new limit. TSTF-490 deletes surveillance requirement (SR) 3.4.16.1 and SR 3.4.16.3. SR 3.4.16.1 verifies reactor coolant gross specific activity is less than the limit and SR 3.4.16.3 determines $\bar{E}$. SPS gross activity and $\bar{E}$ determination appear to be the equivalent of SR 3.4.16.1 and SR 3.4.16.3. However, the radio iodine analysis test is also proposed for deletion at SPS.

The surveillance tests at SPS seem to differ from those in the Westinghouse STS as reviewed by the NRC staff in TSTF-490. Therefore, discuss in detail the gross activity, $\bar{E}$ determination, and radio iodine analysis surveillance tests at SPS. Provide a comparison to the SRs in TSTF-490 and explain any differences. In addition, provide the technical basis for the deletion of the radio iodine analysis test at SPS.

Dominion Energy Virginia Response

See the Table below for a comparison of the RCS sample analysis Surveillance Requirements in the NUREG-1431 Westinghouse STS to the SPS custom TS and the basis for elimination of the SPS TS SRs being subsumed by the proposed TS changes.

TABLE: COMPARISON OF ANALYSIS SURVEILLANCE REQUIREMENTS

Sampling Test	NUREG-1431 Westinghouse ISTS	SPS TS	Difference	Basis for Elimination
Gross Activity	As discussed in the SR 3.4.16.1 Bases, the surveillance requirement requires "performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant. While basically a quantitative measure of radionuclides with half-lives longer than 15 minutes, excluding iodines, the measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. The Surveillance provides an indication of any increase in gross specific activity. The surveillance requirement specifically requires verification that the reactor coolant gross specific activity $\leq$ 100/Ē µCi/gm on a 7 Day frequency or in accordance with the SFCP.	TS Table 4.1-2B/SFCP, Item 1, includes: 1. The requirement to perform a Gross Degassed Activity test 5 days per week. Associated Table Note 2 states, "A gross beta gamma degassed activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of µCi/cc". 2. The requirement to perform Ē Determination semiannually which would include calculating Ē and 100/ Ē. Current TS 3.1.D.1 requires the total specific activity of the reactor coolant due to half-lives of more than 15 minutes shall not exceed 100/Ē µCi/cc.	The Westinghouse ISTS require a quantitative measure of degassed and gaseous activities every 7 days. The SPS TS require the quantitative measure of the degassed and gaseous activities semiannually and additionally require a Gross Degassed Activity 5 days per week.	SR measurement of total specific activity (100/ Ē) and gross degassed activity at SPS is being performed for the same purpose and in only a slightly different manner than the Westinghouse ISTS. Furthermore, the measurement of gross activity in both cases is directly tied to Ē determination, and the requirement for performing Ē determination is being removed from the TS for the reasons noted in TSTF-490, which includes the implementation of the DOSE EQUIVALENT (DE) XE-133 SR. Therefore, the Gross Activity sample test is no longer required and can be eliminated consistent with the reasoning provided in TSTF-490.

Sampling Test	NUREG-1431 Westinghouse ISTS	SPS TS	Difference	Basis for Elimination
E-Bar Determination	The Westinghouse ISTS defines the "Ē – AVERAGE DISINTE-GRATION ENERGY" in Section 1.1, Definitions, as follows: "Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than [15] minutes, making up at least 95% of the total non-iodine activity in the coolant." SR 3.4.16.3 requires determination of Ē from a sample taken in Mode 1 after a minimum of 2 effective full power days and 20 days of Mode 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	The same definition for Ē contained in the Westinghouse ISTS is contained in the SPS TS 3.1 Bases, page 3.1-17. TS Table 4.1-2B, Item 1, includes the requirement to perform an Ē determination. Associated Table Note 3 states, "Ē determination will be started when the gross gamma degassed activity of radionuclides with half-lives greater than 15 minutes analysis indicates ≥ 10 µCi/cc. Routine sample(s) for Ē analysis shall only be taken after a minimum of 2 EFPD and 20 days of power operation have elapsed since reactor was last subcritical for 48 hours or longer.	None	The Ē determination surveillance requirement in the SPS TS uses the same definition and is being performed for the same purpose as the Westinghouse ISTS. The requirement for performing Ē determination is being removed from the SPS TS for the reasons noted in TSTF-490, which includes the implementation of the DE XE-133 surveillance requirement. Therefore, the Ē determination surveillance requirement is no longer required and can be eliminated consistent with the reasoning provided in TSTF-490.

Sampling Test	NUREG-1431 Westinghouse ISTS	SPS TS	Difference	Basis for Elimination
Radio-iodine Analysis	SR 3.4.16.2 requires the verification of reactor coolant DEI-131 specific activity ≤ 1.0 µCi/gm every 14 days and between 2 and 6 hours after THERMAL POWER change of >=15% RTP within a 1 hour period.	Table 4.1-2B/SFCP requires the following: 1. The DEI-131 specific activity be performed Once/2 weeks, and 2. Radio-iodine Analysis (including I-131, I-133 & I-135) once every 4 hours whenever the specific activity exceeds 1.0 µCi/cc DEI-131 or 100/E µCi/cc and until the specific activity of the Reactor Coolant System is restored within its limits AND one sample between 2 & 6 hours following a THERMAL POWER change exceeding 15 percent of RATED POWER within a one hour period provided the average primary coolant temperature >350°F.	SPS TS separate DEI-131 and Radio-iodine analysis as two different line items whereas the Westinghouse ISTS combines the requirements under DEI-131.	When performing DEI-131 analysis, I-131 thru I-135 are used in the calculation. This meets the requirements for radio-iodine analysis as currently stated in the SPS TS. The requirements currently in the SPS TS for radio-iodine analysis are incorporated into the proposed DEI-131 surveillance requirement in the proposed SPS TS revision, and can therefore be deleted as a separate test.

ARCB-RAI-5

10 CFR 50.67(b)(2) requires that the licensee's analyses demonstrate with reasonable assurance that adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent for the duration of the accident.

In the license amendment request, the resultant radiation dose associated with occupancy of the control room has been provided. However, the license amendment request is missing a discussion and/or calculation that accounts for the control room personnel radiation exposure received, for the duration of the accidents, upon ingress/egress from the site boundary to the control room. In order to meet 10 CFR 50.67 the radiation dose for accessing the control room must be evaluated from the site boundary to the control room for both ingress and egress for the duration of the accidents.

Provide an analysis of the radiation dose received from ingress and egress to the control room in enough detail that will enable the NRC staff to be able to perform an independent calculation.

Dominion Energy Virginia Response

Control room occupancy is modeled as 100% of the time during the first 24 hours of a LOCA event. Therefore, transit to and from the control room is only expected after the first 24 hours following an accident. During the 30-day event, a total of 30 round trips between the site boundary and the control room were modeled with a conservative one-way transit time of 30 minutes. This leads to a conservative total of 30 hours of ingress/egress transit time per individual during the event.

The containment building returns to a subatmospheric condition in the first 24 hours, preventing any containment airborne release contribution to ingress/egress dose. Additionally, NUREG-0737, Item II.B.2, "Plant Shielding," states that leakage of systems outside containment need not be considered as potential sources. The only remaining contributor to personnel dose during ingress/egress is the direct radiation exposure from the sources within the containment building.

Direct radiation exposure from the containment was modeled using the RADTRAD radioisotope inventories in the sprayed and unsprayed regions of the containment building at 24 hours. The MicroShield shielding package was used to determine the resulting dose rate. The containment building was modeled as a right circular cylinder of air with a diameter of 126 feet and a height of 65.5 feet. The height of the cylinder represents the distance from ground level to the top of the cylindrical portion of the containment building. The air cylinder is surrounded by a radial concrete shield with a thickness of 4.5 feet and the concrete shield is surrounded by air. The default

MicroShield densities of 2.35 g/cc for concrete and 0.0122 g/cc for air were used. The radioisotope inventories were uniformly distributed within the cylinder of air representing the containment building.

The dose rate for ingress/egress was determined at a point coplanar with the bottom surface of the cylindrical source and at a radial distance of 183.6 feet from the centerline of the containment building, representing an estimate of the closest point of approach during ingress/egress. The resulting dose rate, with buildup, at the closest point of approach to the containment building was 0.3 mrem/hr, resulting in a conservative accumulated dose of 9 mrem during 30 hours of ingress/egress transit. Given the calculated control room dose of 4.4 rem during the LOCA, the accumulated ingress/egress dose is a negligible contribution.

ARCB-RAI-6

The license amendment request proposes to increase the iodine speciation above the water in the fuel handling accident from 57 percent elemental and 43 percent organic iodide to 70 percent elemental and 30 percent organic iodide to align with draft regulatory guide 1.183 revision 1 (DG-1199).

DG-1199 states that the difference in decontamination factors for elemental and organic iodine species results in the iodine above the water being composed of 70 percent elemental and 30 percent organic species for release pressures that remain less than 1,200 pounds per square inch gauge. However, the release pressure was not discussed in the license amendment request.

Discuss if SPS release pressures remain less than 1,200 pounds per square inch gauge for the fuel handling accident.

Dominion Energy Virginia Response

Refueling can begin 100 hours after shutdown (Technical Specification 3.10.A.9) with the RCS < 140°F (REFUELING SHUTDOWN Technical Specification 1.C.1). Dominion Energy Virginia's fuel vendors have verified that rod internal pressure will be <1200 psig under the conditions of the fuel handling accident (≤100 hours after shutdown, water temperature ≥ 140°F, and fuel rod burnup of 62,000 MWD/MTU).

ARCB-RAI-7

The current licensing basis at SPS assumes that the control room is manually isolated at time 0 based on identification that a fuel handling accident (FHA) has occurred and a 2 minute time critical operator action which accomplishes manual isolation of the control room.

The licensee proposes to extend the time critical operator action to isolate the control room after initiation of the FHA from 2 minutes to 12 minutes. The proposed change also includes 20 seconds for the control room dampers to close after the time critical operator action. The licensee states that in order to accommodate the increase in time critical operator action and damper closure, the control room isolation is modeled as occurring at 10 minutes and 20 seconds or 0.1722 hours and therefore, the control room isolation is changing from 0 hours to 0.1722 hours.

However, there seems to be a disconnect between the modeling which assumes isolation of the control room at 10 minutes and 20 seconds and the operator actions which would isolate the control room at 12 minutes. This appears to be non-conservative in that the radiological dose received by the control room operators would be higher than that calculated by the model. From the information supplied in the application, it appears that the modeling should isolate the control room at 12 minutes and 20 seconds to be consistent with the proposed time critical operator action or the proposed time critical operator action should be limited to 10 minutes to be consistent with the modeling.

Explain how the control room isolation modeling and the time critical operator action are consistent with each other.

Dominion Energy Virginia Response

The current design and licensing basis for the FHA is the control room is isolated prior to the plume reaching the control room normal ventilation intake. This is modeled as isolation at t=0 hours in the RADTRAD code. Isolation timing is currently based on a 2-minute time critical operator action and a calculation that determined that plume travel is 2 minutes from the release location to the control room normal intake. The RADTRAD model does not include the 2 minute plume travel time.

The proposed design and licensing basis for the FHA models isolation at 10 minutes and 20 seconds, as described above, which includes 20 seconds for damper operation after the operator action is completed. However, the sequence of events still includes the 2 minute time for plume travel. Therefore the operator action is accomplished in 12 minutes from the drop of the fuel assembly and 10 minutes after the plume reaches the normal intake. The modeling of control room isolation and the time critical operator action are therefore consistent and correct.

ARCB-RAI-8

The current Steam Generator Tube Rupture (SGTR) licensing basis at SPS assumes that the control room in leakage is either 10 cubic feet per minute (cfm) or 500 cfm. In the license amendment request the licensee proposes to change the SGTR control room in leakage to 0 cfm or 250 cfm and the stated reason for the change is maximum ASTM E741 tracer gas test equals 147 ± 6 cfm. An assumed control room in leakage of

0 cfm does not appear to be consistent with the tracer gas test result.

Explain how the most recent tracer gas test supports a control room in leakage of 0 cfm and explain why two different control room in leakages are used in the SGTR analysis.

Dominion Energy Virginia Response

The limiting SGTR control room dose consequences were determined for the pre-accident iodine spike with offsite power available. Under these conditions the normal ventilation system keeps operating and control room isolation is delayed until the Safety Injection (SI) signal plus 20 seconds for damper operation. As a result of the normal ventilation system high flow rate, delayed isolation, and the early availability for release of the pre-accident spike, the control room dose results increase with decreasing unfiltered inleakage as shown below.

Assumed Unfiltered Inleakage cfm	SGTR Control Room Dose Pre-accident Spike Power Available Rem TEDE
500	0.538
250	0.582
10	0.647
0	0.650

Therefore, as long as measured unfiltered inleakage is ≥ 0 cfm, the pre-accident spike remains bounded. The trend does not occur in the concurrent iodine spike cases or the pre-accident spike case with loss of offsite power. These cases were analyzed with up to 250 cfm of unfiltered inleakage and demonstrated an increase in dose with increasing unfiltered inleakage. Therefore, as long as measured unfiltered inleakage is ≤ 250 cfm the concurrent iodine spike and pre-accident spike with loss of offsite power remain bounded.

Attachment 2

REVISED MARKED-UP TECHNICAL SPECIFICATIONS PAGES

Virginia Electric and Power Company
(Dominion Energy Virginia)
Surry Station Units 1 and 2

K. FIRE SUPPRESSION WATER SYSTEM

A fire suppression water system shall consist of: a water source(s), gravity tank(s) or pump(s), and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include yard hydrant curb valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser.

L. OFFSITE DOSE CALCULATION MANUAL (ODCM)

An Offsite Dose Calculation Manual (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.6.B.2 and 6.6.B.3.

M. DOSE EQUIVALENT I-131

Replace with Insert M

~~The dose equivalent I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or in NRC Regulatory Guide 1.109, Revision 1, October 1977.~~

N. GASEOUS RADWASTE TREATMENT SYSTEM

A gaseous radwaste treatment system is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

INSERT M

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11.

W. STAGGERED TEST BASIS

A staggered test basis shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

X. LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank,

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE, or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE, and

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

Insert Y

INSERT Y

Y. DOSE EQUIVALENT XE-133

DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

D. ~~Maximum Reactor Coolant Activity~~

~~Specifications~~

1. ~~The total specific activity of the reactor coolant due to nuclides with half-lives of more than 15 minutes shall not exceed 100/$\bar{E}$ μCi/cc whenever the reactor is critical or the average temperature is greater than 500°F, where $\bar{E}$ is the average sum of the beta and gamma energies, in Mev, per disintegration. If this limit is not satisfied, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection. Should this limit be exceeded by 25%, the reactor shall be made subcritical and cooled to 500°F or less within 2 hours after detection.~~

2. ~~The specific activity of the reactor coolant shall be limited to ≤ 1.0 μCi/cc DOSE EQUIVALENT I-131 whenever the reactor is critical or the average temperature is greater than 500°F.~~

3. ~~The requirements of D.2 above may be modified to allow the specific activity of the reactor coolant > 1.0 μCi/cc DOSE EQUIVALENT I-131 but less than 10.0 μCi/cc DOSE EQUIVALENT I-131. Following shutdown, the unit may be restarted and/or operation may continue for up to 48 hours provided that operation under these circumstances shall not exceed 10 percent of the unit's total yearly operating time. With the specific activity of the reactor coolant > 1.0 μCi/cc DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding 10.0 μCi/cc DOSE EQUIVALENT I-131, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection.~~

4. ~~If the specific activity of the reactor coolant exceeds 1.0 μCi/cc DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/cc, a report shall be prepared and submitted to the Commission pursuant to Specification 6.6.A.2. This report shall contain the results of the specific activity analysis together with the following information:~~

   a. ~~Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,~~

   b. ~~Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded,~~

Replace with Insert 3.1.D

~~c. History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded,~~

~~d. The time duration when the specific activity of the primary coolant exceeded 1.0 μCi/cc DOSE EQUIVALENT I-131,~~

~~e. Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded, and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations, and~~

~~f. Graph of the I-131 concentration and one other radioiodine isotope concentration in μCi/cc as a function of time for the duration of the specific activity above the steady-state level.~~

INSERT 3.1.D

D. RCS Specific Activity

Applicability

The following specifications are applicable whenever T_{avg} (average RCS temperature) exceeds 200°F.

Specification

1. The specific activity of the primary coolant shall be limited to $\leq$1 μCi/gm DOSE EQUIVALENT I-131.

   a. With the specific activity of the primary coolant > 1 μCi/gm DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 $\leq$ 10 μCi/gm once per 4 hours.

   b. With the specific activity of the primary coolant > 1 μCi/gm DOSE EQUIVALENT I-131, but $\leq$ 10 μCi/gm, unit startup or POWER OPERATION may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to $\leq$ 1 μCi/gm limit.

   c. With the specific activity of the primary coolant >1 μCi/gm DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or DOSE EQUIVALENT I-131 is > 10 μCi/gm, place the reactor in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

2. The specific activity of the primary coolant shall be limited to $\leq$ 234 μCi/gm DOSE EQUIVALENT XE-133.

   a. With the specific activity of the primary coolant >234 μCi/gm DOSE EQUIVALENT XE-133, unit startup or POWER OPERATION may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to $\leq$ 234 μCi/gm limit.

   b. With the specific activity of the primary coolant > 234 μCi/gm DOSE EQUIVALENT XE-133 for more than 48 hours during one continuous time interval, place the reactor in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

Basis

~~The specified limit provides protection to the public against the potential release of reactor coolant activity to the atmosphere, as demonstrated by the following analysis of a steam generator tube rupture accident in UFSAR Chapter 14.3.1.~~

~~Rupture of a steam generator tube would allow radionuclides in the reactor coolant to enter the secondary system. The limiting case involves a double-ended tube rupture coincident with loss of the condenser and release of steam from the secondary side to the atmosphere via the main steam safety valves or atmospheric relief valves. This is assumed to continue for 30 minutes in the analysis. The operator will take action to reduce the primary side temperature to a value below that corresponding to the relief or safety valve setpoint. Once this is accomplished the valves can be closed and the release terminated.~~

Insert Basis 3.1.D

~~Permitting startup and/or REACTOR OPERATION to continue for limited time periods with the reactor coolant's specific activity > 1.0 μCi/cc but < 10.0 μCi/cc DOSE EQUIVALENT I-131 accommodates possible iodine spiking phenomena which may occur following changes in THERMAL POWER. Although the analysis of a steam generator tube rupture initiated with primary coolant activity at the 10.0 μCi/cc transient limit shows offsite doses less than or equal to the Regulatory Guide (RG) 1.183 limits, operation at the transient limit is restricted to no more than 10 percent of the unit's yearly operating time to limit the risk of appreciable release following a postulated steam generator tube rupture.~~

~~The basis for the 500°F temperature contained in the Specification is that the saturation pressure corresponding to 500°F, i.e., 680.8 psia, is well below the pressure at which the atmospheric relief valves on the secondary side could be actuated.~~

~~The accident analysis examines two cases of iodine spiking. For the case with a pre-existing iodine spike, the transient coolant activity limit of 10.0 μCi/cc is assumed. For the case of a concurrent spike, the initial activity is assumed to correspond to the steady state limit of 1.0 μCi/cc. The concurrent iodine spike is modeled with a conservative iodine appearance rate. Both cases show doses at the exclusion area and low population zone boundaries which are less than or equal to the RG 1.183 limits and control room doses which are within the General Design Criterion (GDC) 19 guidelines.~~

~~Measurement of~~ $\bar{E}$ ~~will be performed at least twice annually. Calculations required to determine~~ $\bar{E}$ ~~will consist of the following:~~

~~1.~~ $\bar{E}$ ~~shall be the average (weighed in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.~~

~~2.~~ ~~A determination of the beta and gamma decay energy per disintegration of each nuclide determined in (1) above by applying known decay energies and schemes.~~

~~3.~~ ~~A calculation of~~ $\bar{E}$ ~~by appropriate weighing of each nuclide's beta and gamma energy with its concentration as determined in (1) above.~~

~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/cc) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be either: a) those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites", or b) Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Compliance with 10 CFR Part 50, Appendix I."~~

INSERT BASIS 3.1.D

BASES

BACKGROUND – The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR 50.67 (Ref. 1). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

The Reactor Coolant System (RCS) specific activity Limiting Condition for Operation (LCO) limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (Ref. 2).

APPLICABLE SAFETY ANALYSES - The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate Standard Review Plan acceptance criteria following a SLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.6.H, Secondary Specific Activity.

The analyses for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a SLB (by a factor of 500), or SGTR (by a factor of 335), respectively. The second case assumes the initial reactor coolant iodine activity is at 10.0 μCi/gm DOSE EQUIVALENT I-131 due to an iodine spike caused by a reactor or RCS transient prior to the accident. In both cases, the noble gas specific activity is assumed to be 234 μCi/gm DOSE EQUIVALENT XE-133.

The SGTR analysis assumes a coincident loss of offsite power. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the Residual Heat Removal (RHR) system is placed in service.

The SLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. Reactor trip occurs after the generation of an SI signal on low steam line pressure. The affected SG blows down completely and steam is vented directly to the Turbine Building to maximize control room dose. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR system is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible if the activity levels do not exceed 10.0 μCi/gm for more than 48 hours.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO - The iodine specific activity in the reactor coolant is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 234 μCi/gm DOSE EQUIVALENT XE-133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate SRP acceptance criteria (Ref. 2).

The SLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).

APPLICABILITY - In REACTOR OPERATION conditions where T_{avg} exceeds 200°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a SLB or SGTR to within the SRP acceptance criteria (Ref. 2).

In COLD SHUTDOWN and REFUELING SHUTDOWN the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

ACTIONS

3.1.D.1.a

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is $\leq$ 10.0 μCi/gm. The completion time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend.

3.1.D.1.b

The DOSE EQUIVALENT I-131 must be restored to within limit within 48 hours. The completion time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A unit startup and/or continued plant operation is permitted relying on required actions 3.1.D.1.a and b while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, POWER OPERATION.

3.1.D.1.c

If the required action of Condition 3.1.D.1.a or 3.1.D.1.b is not met, or if the DOSE EQUIVALENT I-131 is > 10.0 μCi/gm, the reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours. The required completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

3.1.D.2.a

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within the limit within 48 hours. The completion time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A unit startup and/or continued plant operation is permitted relying on required action 3.1.D.2.a while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, POWER OPERATION.

3.1.D.2.b

If the required action or associated Allowed Outage Time of Condition 3.1.D.2.a is not met, or if the DOSE EQUIVALENT XE-133 is > 234 μCi/gm, the reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours. The required action and completion time are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

REFERENCES

1. 10 CFR 50.67

2. Standard Review Plan (SRP) Section 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms"

3. UFSAR, Section 14.3.1 Steam Generator Tube Rupture

4. UFSAR, Section 14.3.2 Steam Line Break

3. With three auxiliary feedwater pumps inoperable, immediately initiate action to restore one inoperable pump to OPERABLE status. Specification 3.0.1 and all other required actions directing mode changes are suspended until one inoperable pump is restored to OPERABLE status.

G. The following actions shall be taken with inoperability of a component or instrumentation other than the flow instrumentation in one or both redundant auxiliary feedwater flowpaths required by Specification 3.6.C.3 on the affected unit: (See Specification 3.7 and TS Table 3.7-6 for auxiliary feedwater flow instrumentation requirements.)

1. With component or instrumentation inoperability in one redundant flowpath, restore the inoperable component or instrumentation to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 6 hours and be less than 350°F and 450 psig within the following 12 hours.

2. With component or instrumentation inoperability affecting both redundant flowpaths, immediately initiate action to restore the inoperable component or instrumentation in one flowpath to OPERABLE status. Specification 3.0.1 and all other required actions directing mode changes are suspended until the inoperable component or instrumentation in one flowpath is restored to OPERABLE status.

H. ~~The specific activity of the secondary coolant system shall be ≤ 0.10 μCi/cc DOSE EQUIVALENT I-131. If the specific activity of the secondary coolant system exceeds 0.10 μCi/cc DOSE EQUIVALENT I-131, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection and in COLD SHUTDOWN within the following 30 hours.~~

Replace with Insert 3.6.H

INSERT 3.6.H

H. The specific activity of the secondary coolant system shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131. If the specific activity of the secondary coolant system exceeds 0.10 μCi/gm DOSE EQUIVALENT I-131, the reactor shall be placed in HOT SHUTDOWN within 6 hours after detection and in COLD SHUTDOWN within the following 30 hours.

RCS Flow

This surveillance requirement in Table 4.1-2A is modified by a note that allows entry into POWER OPERATION, without having performed the surveillance, and placement of the unit in the best condition for performing the surveillance. The note states that the surveillance requirement is not required to be performed until 7 days after reaching a THERMAL POWER of ≥ 90% of RATED POWER (i.e., shall be performed within 7 days after reaching 90% of RATED POWER). [Reference: NRC Safety Evaluation for License Amendments 270/269, issued October 19, 2010] The 7 day period after reaching 90% of RATED POWER is reasonable to establish stable operating conditions, install the test equipment, perform the test, and analyze the results. If reactor power is reduced below 90% of RATED POWER before completion of the RCS flow surveillance, the 7 day period shall be exited, and a separate 7 day period shall be entered when the required condition of reaching 90% of RATED POWER is subsequently achieved. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

INSERT Table 4.1-2B Basis Section

INSERT - BASIS TS Table 4.1-2B

SURVEILLANCE REQUIREMENTS Table 4.1-2B

Item 1 - RCS Coolant Liquid Samples

DOSE EQUIVALENT I-131 - This surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The SFCP 14 day Frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change $\geq$ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

DOSE EQUIVALENT XE-133 - This surveillance requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days per the SFCP. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken or equivalent sampling method. This surveillance provides an indication of any increase in the noble gas specific activity.

Trending the results of this surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The SFCP 7 day Frequency considers the low probability of a gross fuel failure during this time.

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in this calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

TABLE 4.1-2B
MINIMUM FREQUENCIES FOR SAMPLING TESTS

	DESCRIPTION	TEST	FREQUENCY	UFSAR SECTION REFERENCE
1.	Reactor Coolant Liquid Samples	Radio-Chemical Analysis (1)	SFCP (5)	
		~~Gross Activity (2)~~	~~SFCP (5)~~	~~9.1~~
		Tritium Activity	SFCP (5)	9.1
		* Chemistry (CL, F & O_2)	SFCP (9)	4
		* Boron Concentration	SFCP	9.1
		~~$\bar{E}$ Determination~~	~~SFCP (3)~~	
		DOSE EQUIVALENT I-131	SFCP ~~(5)~~ (4)(7)	
		DOSE EQUIVALENT XE-133 ~~Radio-iodine Analysis (including I-131, I-133 & I-135)~~	~~Once/4 hours (6) and (7) below~~ SFCP (4)	
2.	Refueling Water Storage	Chemistry (Cl & F)	SFCP	6
3.	Boric Acid Tanks	* Boron Concentration	SFCP	9.1
4.	Chemical Additive Tank	NaOH Concentration	SFCP	6
5.	Spent Fuel Pit	* Boron Concentration	SFCP	9.5
6.	Secondary Coolant	DOSE EQUIVALENT I-131	SFCP	
7.	Stack Gas Iodine and Particulate Samples	* I-131 and particulate radioactive releases	SFCP	

* See Specification 4.1.D

SFCP - Surveillance frequencies are specified in the Surveillance Frequency Control Program.

(1) A radiochemical analysis will be made to evaluate the following corrosion products: Cr-51, Fe-59, Mn-54, Co-58, and Co-60.

(2) ~~A gross beta-gamma degassed activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/cc.~~ Deleted

(3) ~~$\bar{E}$ determination will be started when the gross gamma degassed activity of radionuclides with half-lives greater than 15 minutes analysis indicates ≥ 10 μCi/cc. Routine sample(s) for $\bar{E}$ analyses shall only be taken after a minimum of 2 EFPD and 20 days of power operation have elapsed since reactor was last subcritical for 48 hours or longer.~~ Deleted.

(4) ~~Deleted.~~ Whenever T_{avg} (average RCS temperature) exceeds 200°F.

(5) When reactor is critical and average primary coolant temperature ≥ 350°F.

(6) ~~Whenever the specific activity exceeds 1.0 μCi/cc DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/cc and until the specific activity of the Reactor Coolant System is restored within its limits.~~ Deleted.

(7) One sample between 2 ~~&~~ and 6 hours following a THERMAL POWER change ~~exceeding~~ ≥ 15 percent of RATED POWER within a one hour period ~~provided the average primary coolant temperature ≥ 350°F~~.

(8) Deleted.

(9) Sampling for chloride and fluoride concentrations is not required when fuel is removed from the reactor vessel and the reactor coolant inventory is drained below the reactor vessel flange, whether the upper internals and/or the vessel head are in place or not. Sampling for oxygen concentration is not required when the reactor coolant temperature is below 250 degrees F.

b. ~~The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.1.D.4. In addition, the information itemized in Specification 3.1.D.4 shall be included in this report.~~ Deleted.

3. Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after T_{avg} exceeds 200°F following completion of an inspection performed in accordance with the Specification 6.4.Q, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,
b. Degradation mechanisms found,
c. Nondestructive examination techniques utilized for each degradation mechanism,
d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,
e. Number of tubes plugged during the inspection outage for each degradation mechanism,
f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator.
g. The results of condition monitoring, including the results of tube pulls and in-situ testing,
h. The primary to secondary LEAKAGE rate observed in each SG (if it is not practical to assign the LEAKAGE to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,

Attachment 3

REVISED PROPOSED TECHNICAL SPECIFICATIONS PAGES

Virginia Electric and Power Company
(Dominion Energy Virginia)
Surry Station Units 1 and 2

K. FIRE SUPPRESSION WATER SYSTEM

A fire suppression water system shall consist of: a water source(s), gravity tank(s) or pump(s), and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include yard hydrant curb valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser.

L. OFFSITE DOSE CALCULATION MANUAL (ODCM)

An Offsite Dose Calculation Manual (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.6.B.2 and 6.6.B.3.

M. DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11.

N. GASEOUS RADWASTE TREATMENT SYSTEM

A gaseous radwaste treatment system is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

W. STAGGERED TEST BASIS

A staggered test basis shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

X. LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank,

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE, or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE, and

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

Y. DOSE EQUIVALENT XE-133

DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

D. RCS Specific Activity

Applicability

The following specifications are applicable whenever T_{avg} (average RCS temperature) exceeds 200°F.

Specification

1. The specific activity of the primary coolant shall be limited to ≤ 1 μCi/gm DOSE EQUIVALENT I-131.

   a. With the specific activity of the primary coolant > 1 μCi/gm DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 ≤ 10 μCi/gm once per 4 hours.

   b. With the specific activity of the primary coolant > 1 μCi/gm DOSE EQUIVALENT I-131, but ≤ 10 μCi/gm, unit startup or POWER OPERATION may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to ≤ 1 μCi/gm limit.

   c. With the specific activity of the primary coolant > 1 μCi/gm DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or DOSE EQUIVALENT I-131 is > 10 μCi/gm, place the reactor in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

2. The specific activity of the primary coolant shall be limited to ≤ 234 μCi/gm DOSE EQUIVALENT XE-133.

   a. With the specific activity of the primary coolant > 234 μCi/gm DOSE EQUIVALENT XE-133, unit startup or POWER OPERATION may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to ≤ 234 μCi/gm limit.

   b. With the specific activity of the primary coolant > 234 μCi/gm DOSE EQUIVALENT XE-133 for more than 48 hours during one continuous time interval, place the reactor in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

BASES

BACKGROUND - The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR 50.67 (Ref. 1). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

The Reactor Coolant System (RCS) specific activity Limiting Condition for Operation (LCO) limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (Ref. 2).

APPLICABLE SAFETY ANALYSES - The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate Standard Review Plan acceptance criteria following a SLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.6.H, Secondary Specific Activity.

The analyses for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a SLB (by a factor of 500), or SGTR (by a factor of 335), respectively. The second case assumes the initial reactor coolant iodine activity is at 10.0 μCi/gm DOSE EQUIVALENT I-131 due to an iodine spike caused by a reactor or RCS transient prior to the accident. In both cases, the noble gas specific activity is assumed to be 234 μCi/gm DOSE EQUIVALENT XE-133.

The SGTR analysis assumes a coincident loss of offsite power. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the Residual Heat Removal (RHR) system is placed in service.

The SLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. Reactor trip occurs after the generation of an SI signal on low steam line pressure. The affected SG blows down completely and steam is vented directly to the Turbine Building to maximize control room dose. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR system is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible if the activity levels do not exceed 10.0 μCi/gm for more than 48 hours.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO - The iodine specific activity in the reactor coolant is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 234 μCi/gm DOSE EQUIVALENT XE-133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate SRP acceptance criteria (Ref. 2).

The SLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).

APPLICABILITY - In REACTOR OPERATION conditions where Tavg exceeds 200°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a SLB or SGTR to within the SRP acceptance criteria (Ref. 2).

In COLD SHUTDOWN and REFUELING SHUTDOWN the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

ACTIONS

3.1.D.1.a

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is $\leq$ 10.0 μCi/gm. The completion time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend.

3.1.D.1.b

The DOSE EQUIVALENT I-131 must be restored to within limit within 48 hours. The completion time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A unit startup and/or continued plant operation is permitted relying on required actions 3.1.D.1.a and b while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, POWER OPERATION.

3.1.D.1.c

If the required action of Condition 3.1.D.1.a or 3.1.D.1.b is not met, or if the DOSE EQUIVALENT I-131 is > 10.0 μCi/gm, the reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours. The required completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

3.1.D.2.a

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within the limit within 48 hours. The completion time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A unit startup and/or continued plant operation is permitted relying on required action 3.1.D.2.a while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, POWER OPERATION.

3.1.D.2.b

If the required action or associated Allowed Outage Time of Condition 3.1.D.2.a is not met, or if the DOSE EQUIVALENT XE-133 is > 234 μCi/gm, the reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours. The required action and completion time are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

REFERENCES

1. 10 CFR 50.67
2. Standard Review Plan (SRP) Section 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms"
3. UFSAR, Section 14.3.1 Steam Generator Tube Rupture
4. UFSAR, Section 14.3.2 Steam Line Break

3. With three auxiliary feedwater pumps inoperable, immediately initiate action to restore one inoperable pump to OPERABLE status. Specification 3.0.1 and all other required actions directing mode changes are suspended until one inoperable pump is restored to OPERABLE status.

G. The following actions shall be taken with inoperability of a component or instrumentation other than the flow instrumentation in one or both redundant auxiliary feedwater flowpaths required by Specification 3.6.C.3 on the affected unit: (See Specification 3.7 and TS Table 3.7-6 for auxiliary feedwater flow instrumentation requirements.)

1. With component or instrumentation inoperability in one redundant flowpath, restore the inoperable component or instrumentation to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 6 hours and be less than 350°F and 450 psig within the following 12 hours.

2. With component or instrumentation inoperability affecting both redundant flowpaths, immediately initiate action to restore the inoperable component or instrumentation in one flowpath to OPERABLE status. Specification 3.0.1 and all other required actions directing mode changes are suspended until the inoperable component or instrumentation in one flowpath is restored to OPERABLE status.

H. The specific activity of the secondary coolant system shall be $\leq$ 10 μCi/gm DOSE EQUIVALENT I-131. If the specific activity of the secondary coolant system exceeds 0.10 μCi/gm DOSE EQUIVALENT I-131, the reactor shall be placed in HOT SHUTDOWN within 6 hours after detection and in COLD SHUTDOWN within the following 30 hours.

RCS Flow

This surveillance requirement in Table 4.1-2A is modified by a note that allows entry into POWER OPERATION, without having performed the surveillance, and placement of the unit in the best condition for performing the surveillance. The note states that the surveillance requirement is not required to be performed until 7 days after reaching a THERMAL POWER of ≥ 90% of RATED POWER (i.e., shall be performed within 7 days after reaching 90% of RATED POWER). [Reference: NRC Safety Evaluation for License Amendments 270/269, issued October 19, 2010] The 7 day period after reaching 90% of RATED POWER is reasonable to establish stable operating conditions, install the test equipment, perform the test, and analyze the results. If reactor power is reduced below 90% of RATED POWER before completion of the RCS flow surveillance, the 7 day period shall be exited, and a separate 7 day period shall be entered when the required condition of reaching 90% of RATED POWER is subsequently achieved. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS Table 4.1-2B

Item 1 - RCS Coolant Liquid Samples

DOSE EQUIVALENT I-131 - This surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The SFCP 14 day Frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

DOSE EQUIVALENT XE-133 - This surveillance requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days per the SFCP. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken or equivalent sampling method. This surveillance provides an indication of any increase in the noble gas specific activity.

Trending the results of this surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The SFCP 7 day Frequency considers the low probability of a gross fuel failure during this time.

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in this calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

TABLE 4.1-2B
MINIMUM FREQUENCIES FOR SAMPLING TESTS

	DESCRIPTION	TEST	FREQUENCY	UFSAR SECTION REFERENCE
1.	Reactor Coolant Liquid Samples	Radio-Chemical Analysis (1)	SFCP (5)	
		Tritium Activity	SFCP (5)	9.1
		* Chemistry (CL, F & O_2)	SFCP (9)	4
		* Boron Concentration	SFCP	9.1
		DOSE EQUIVALENT I-131	SFCP (4)(7)	
		DOSE EQUIVALENT XE-133	SFCP (4)	
2.	Refueling Water Storage	Chemistry (Cl & F)	SFCP	6
3.	Boric Acid Tanks	* Boron Concentration	SFCP	9.1
4.	Chemical Additive Tank	NaOH Concentration	SFCP	6
5.	Spent Fuel Pit	* Boron Concentration	SFCP	9.5
6.	Secondary Coolant	DOSE EQUIVALENT I-131	SFCP	
7.	Stack Gas Iodine and Particulate Samples	* I-131 and particulate radioactive releases	SFCP	

* See Specification 4.1.D

SFCP - Surveillance frequencies are specified in the Surveillance Frequency Control Program.

(1) A radiochemical analysis will be made to evaluate the following corrosion products: Cr-51, Fe-59, Mn-54, Co-58, and Co-60.

(2) Deleted

(3) Deleted

(4) Whenever T_{avg} (average RCS temperature) exceeds 200°F.

(5) When reactor is critical and average primary coolant temperature $\geq$ 350°F.

(6) Deleted

(7) One sample between 2 and 6 hours following a THERMAL POWER change $\geq$ 15 percent of RATED POWER within a one hour period.

(8) Deleted.

(9) Sampling for chloride and fluoride concentrations is not required when fuel is removed from the reactor vessel and the reactor coolant inventory is drained below the reactor vessel flange, whether the upper internals and/or the vessel head are in place or not. Sampling for oxygen concentration is not required when the reactor coolant temperature is below 250 degrees F.

b. Deleted

3. Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after T_{avg} exceeds 200°F following completion of an inspection performed in accordance with the Specification 6.4.Q, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,
b. Degradation mechanisms found,
c. Nondestructive examination techniques utilized for each degradation mechanism,
d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,
e. Number of tubes plugged during the inspection outage for each degradation mechanism,
f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator.
g. The results of condition monitoring, including the results of tube pulls and in-situ testing,
h. The primary to secondary LEAKAGE rate observed in each SG (if it is not practical to assign the LEAKAGE to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,