Talking points for Columbia AST meeting
Accident Dose Branch
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Note: Unless otherwise specified, references to attachments and pages are to be considered from the September 30, 2004 submittal.

1) Attachment 2, page 2, Table 1 states that the fission product inventory is ORIGEN 2-based. Please explain what ORIGEN 2-based means. Was the inventory determined using the ORIGEN 2 code?

2) Regulatory Guide 1.183, Appendix A, Regulatory Position 5.5 states:

If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% ***of the total iodine activity*** *in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.*

The LOCA calculation for ESF leakage appears to exclude the aerosol iodine by using a filter on this pathway that filters 100% of the aerosol. Therefore, the 10% is not of the total iodine, but of the elemental and organic iodine. Please provide justification for this assumption.

3) Columbia Generating Station (CGS) credits the deposition of organic iodine in the steamline piping. Although, the number is small, the Nuclear Regulatory Commission (NRC) staff does not typically credit organic iodine removal. Approved Alternative Source Term (AST) amendments were reviewed and previous credit for organic deposition was not found for any previous amendments. Therefore, this assumption will need to be looked at in more depth including modeling any re-evolution of the iodines in the steamlines, and the impact of any thermal heating of the steamline due to deposition of radionuclides within the steamline. Please provide additional information to justify the use of organic deposition

4) The NRC staff requests additional justification for the time used to model the early-in-vessel release. The CGS model starts this release at 0.533 hours as stated on page 5.10 of the LOCA calculation. Justify why this value should not be 0.5 hours as given in Regulatory Guide 1.183.

5) Regulatory Guide 1.183, Position 5.1.2, “Credit for Engineered Safeguard Features,” states:

Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.

Attachment 2, Page 9 of the CGS license amendment states that credited mitigation features meet these requirements and are automatic except RHR drywell sprays and SLC injection.

a) Please verify that RHR drywell sprays will be operable by technical specifications and are powered by emergency power sources.

6) The following questions pertain to the spray removal credit in containment and steam lines.

a) Per Technical Specification Bases B 3.6.2.2, "Suppression Pool Water Level," The suppression pool volume ranges between approximately 112,000 cubic ft at the low water level limit of 30 ft 9.75 inches and approximately 114,000 cubic ft at the high water level limit of 31 ft 1.75 inches.

Chapter 15, Table 15.8-1, states the minimum suppression pool liquid volume is 112,197 cubic feet. It also states that the water in the pedestal and water lower than 12' below the vent exit are not included in this volume.

The minimum volume of the suppression pool assumed in the LOCA calculation (NE-02-04-05) is 137,262 cubic feet. This value includes the pedestal and water 12' below the vent exits. Please justify the use of this volume and justify that the water in the pedestal and 12' below the vent exits can and will mix completely with the volume credited for determining the spray DF and for the determination of the doses from ECCS leakage.

b) Page 5.12 of NE-02-04-05 states:

No maximum DF is established for aerosol removal (as permitted by Reference 3), and there is no practical need to limit elemental iodine removal (since Revision I of Reference 13 establishes a minimum elemental iodine partition coefficient, H, of 300 as long as the pH is greater than approximately 7.3).

Please specify where Reference 13 provides the partition coefficient for sodium pentaborate at a pH of 7.3 and justify its use if it is not explicitly specified.

c) Page 5.12 of NE-02-04-05 states:

The limited amount of elemental iodine initially present (4.85%) means that once the DF is applied, the percentage of elemental iodine remaining airborne would be approximately 0.04% of the total release. This is only 27% of the organic iodine percentage; and therefore, this amount may be neglected (particularly because the pH reaches 7.3 only after 30 days when the dominant dose contributor 1-131 has already been through 3.7 halflives).

The staff would like to discuss this logic.

d) The staff would like to understand the logic behind treating elemental iodine removal and particulate iodine removal with the same removal coefficients.

e) The staff would like to understand the method used to determine the deposition velocity and flow rates used for aerosol, elemental and organic deposition in the steamlines.

f) Please confirm that the surface areas used for the aerosol removal in the steamlines includes only horizontal piping and is calculated using the following equation:

Area = $\text{Diameter}_{\text{Internal}}$.x Horizontal Length

g) Please confirm that the surface areas used for elemental iodine in the steamlines are calculated using the following equation:

Area = $\text{Diameter}_{\text{Internal}}$.x Length x π

h) The staff would like to understand more about how the MSIV leakage flow rates are determined.

7) CGS proposes to no longer credit the main steam isolation valve leakage control system and remove the associated operability requirements from technical specifications. How is potential leakage from this system accounted for in the LOCA dose model and controlled by technical specifications?

8) CGS has requested that credit for the SGTS is delayed for the first 20 minutes while a negative pressure condition is being established in secondary containment. The basis for this 20 minute drawdown needs to be clarified. Does the 20 minute drawdown include the time for the SGTS to become operational (time for an initiation signal etc.)?

9) For the calculation of the aerosol removal rates in the drywell, CGS uses guidance from NUREG-0800, Section 6.5.2. To calculate the spray removal rate for particulate iodine the sprayed volume is used. Please justify the volume used?

10) Attachment 1, page 47 states ESF "Leakage was assumed to start at t =15 minutes after the event." Please justify this assumption.

11) Attachment 2, page 9 states that CGS conforms to Regulatory Position 5.1.3. Regulatory Position 5.1.3 states:

The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. ...

The ranges of flow rates out the standby gas treatment system are given in the proposed Technical Specification 5.5 as 4320 to 5280 cfm for the SGTS and 900 to 1100 cfm for the CREF system.

a) The modeling of the SGTS appears to use a flow rate of 5000 cfm for the SGTS. Please justify why the nominal value is conservative when calculating a postulated dose.

b) The volumetric flow rate provided in Calculation NE-02-04-1 (Rev. 1), Section

3.10, page 5.003 states the actual flow rate is 5378 ± 433.5 cfm. Please clarify the actual range of flows for this system.

12) Attachment 2, page 4, states that the nuclides used for CGS are *the 60 identified as potentially important contributors to TEDE in NUREG/CR-4691 (MACCS Users Guide) [less the two cobalt isotopes which have a minor impact] plus four additional noble gas isotopes from TID-14844, plus three other short-lived noble gas isotopes, plus Ba137m for a total of 66.*

The staff believes the list from NUREG/CR-4691 is the same list as the default list provided for the RADTRAD computer code. The 60 radionuclides that are contained in the RADTRAD code were selected based upon a study that determined that those 60 radionuclides have the greatest impact on offsite dose assuming that each individual element has an equal release fraction.

a) Attachment 2, page 7, Section 4.2.3, states that the models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel. It states that the CGS analysis conforms to this guidance. Confirm that the most conservative radionuclides were used to determine the source for the CGS shielding studies for the shine doses from external sources to the control room.

b) Please justify the changes to the default RADTRAD list of nuclides.

13) UFSAR Section 4.1.2.1.3, "Fuel Assembly Description," states:

The core is loaded with FANP and Westinghouse Electric Company reload fuel. The Westinghouse Electric Company reload fuel assemblies are composed of a 10 x 10 array of fuel rods with a central, cruciform water channel (see Reference 4.1-8). The FANP reload fuel assemblies are composed of 10 x 10 array of fuel rods with a single, large, central water channel (Reference 4.1-20).

Page 62 of Attachment 1 states that the Fuel Handling Accident (FHA) Analysis is based upon an 8 x 8 fuel pin array with 250 fuel pins that are postulated to break. On page 18 of Attachment 2, CGS states that they conform to Regulatory Position 1.1, Appendix B Regulatory Guide 1.183. This Regulatory Position states:

The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case.

For the allowed fuel designs in the CGS core please describe why the proposed analysis, based upon an 8x8 assembly, provides the limiting radiological consequences for the fuel handling accident.

14) Page 63 of 91 of Attachment 1 states that:

Based on the comparable water depth available for decontamination and the difference in the postulated drop distances, Energy Northwest concludes that the consequences of *an FHA over the reactor cavity bound those for an FHA over the transfer area or over the spent fuel pool. This conclusion is consistent with the NRC staff conclusion for a similar configuration at the Fitzpatrick plant as documented in a recent Safety Evaluation Report (SER) (Reference 32).*

The staff reviewed the Fitzpatrick SER and found that the basis of the staff's conclusion was as follows:

ENO stated that the implied reduction in scrubbing efficiency is offset by the reduced number of fuel rods (i.e., 81 vs. 125) that are projected to be damaged by a fuel assembly drop over the spent fuel pool.

CGS has not provided a similar argument because CGS has not provided a value (and justifying analysis) for the reduced number of damaged fuel rods for an FHA in the fuel transfer area or over the spent fuel pool. If CGS plans to use this method of analysis please provide the number of fuel rods and the analysis justifying this number.

15) Page 62 of Attachment 1 states:

The TS minimum required water depth available over the point of fuel assembly impact is approximately 22', just 1' lower than the 23' upon which a DF of 200 is based.

The discussion is focused on the water level above the point of impact for an FHA over the fuel transfer area or the spent fuel pool. It compares this depth of water to the 23' upon which the DF of 200 is based. The depth should be compared to a conservative release point of the radioactivity from the damaged fuel assemblies. Please provide details about the assumptions used to determine water depth above these release points. Please justify your assumptions. If the assembly is assumed to lie flat on top of the racks justify why it is not possible that the fuel assembly could be in any other position.

16) Page 19 of 22 of Attachment 2 provides Table 3, "Comparison with Regulatory Guide 1.183, Appendix B." In Table 3 CGS states that the application conforms with Regulatory Position 5.3. Regulatory Position 5.3 states:

If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open),[3] the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.

Footnote 3:
The staff will generally require that technical specifications allowing such operations include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such administrative controls will generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses should generally not credit this manual isolation.

Several proposed technical specification changes (for example TS 3.6.4.1, 3.6.4.2, 3.6.4.3) delete the requirement for these systems to be operable and do not include any controls for manual isolation if a fuel handling accident were to occur. Explain how the proposed specification and bases provide assurance that the intent of closure as a defense-in-depth measure is accomplished and that the open penetrations are closed.

17) General Design Criterion 61 and 64 are part of the CGS licensing bases. For the proposed technical specifications changes describe how these criteria continue to be met for these proposed changes.

18) Several proposed specification changes delete the Note: LCO 3.0.3 is not applicable. For an example see the required action for Technical Specification LCO 3.7.4, Action E. This change appears to deviate from the TSTF-51 traveler. Please provide a justification for this change.

19) Appendix B to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components (SSCs) that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Appendix B, Criterion III, "Design Control," requires that design control measures be provided for verifying or checking the adequacy of a design. GL 2003-01, "Control Room Habitability," addresses issues with respect to assumed values of unfiltered inleakage. Generally, these issues can only be resolved by inleakage testing.

    Section 4.2 of Attachment 1 provides the test conditions and results of inleakage testing. This section does not provide a description of the inleakage testing performed in the normal operating mode credited in the FHA and CRDA. In light of your Appendix B requirements, GL 2003-01, and because the 1100 cfm value for unfiltered inleakage is not based upon a measurement during this mode of operation, justification should be provided to explain why this number is appropriate. Please provide information as to how CGS has confirmed the inleakage characteristics of the control room envelope in the normal operating mode credited for the duration of the FHA and CRDA. Please provide details regarding your control room, design, maintenance and assessments to justify the use of and any plans to verify this number.

    a. Does the 1100 cfm unfiltered inleakage include 10 cfm for ingress and egress into and out of the control room over the duration of the accident?

    b. How much of the 1100 cfm is due to forced design flow and how much is assumed for unfiltered inleakage due to inflow other than egress and ingress.

    c. The use of 1100 cfm of unfiltered inleakage for the design bases FHA and CRDA meets the requirements of 10CFR 50.36, Criterion 2. This value is “a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.” Since use of the 1100 unfiltered inleakage value meets 50.36, Criterion 2, and the system which supplies this inflow essentially replaces the technical specification for the Control Room Emergency Filtration system (TS 3.7.3), please justify why CGS has not proposed a limiting condition for operation for this flow rate or provide a limiting condition for operation for this value.

20) Regulatory Position 5.1.2 of Regulatory Guide 1.183 states:

Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.

Attachment 2, Table 1 states that the CGS analysis conforms to this regulatory position. For the FHA and CRDA accident please provide additional details how CGS control room HVAC credited for these accidents conforms to Regulatory Position 5.1.2. State whether the system operation credited is operable by technical specification, is powered by emergency power sources, credits the worst case single failure and models the occurrence and timing of a loss of offsite power. Please provide justification for these answers.

21) Page 19 of Attachment 2 states that CGS conforms to Regulatory Position 5.3. Regulatory Position 5.3 states:

If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open),[3] the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.

Page 67, Section 4.7.4 states:

For modeling purposes, a fractional release rate of 2.3 volumes per hours was utilized to ensure that at least 99% of the activity was released from the reactor building during the first 2 hours.

If less than 100% is released from building how does this conform to Regulatory Position 5.3?

22) CGS proposes to remove the following words from page B.3.6.4.1-3 of the technical specifications:

CORE ALTERATIONS and movement of irradiated fuel assemblies must be immediately suspended if the secondary containment is inoperable.

Attachment 1, page 62 states:

The analysis assumed a ground level release from the reactor building over a 2-hour period. No credit was taken for secondary containment, the SGT system or the CREF system. The assumptions used in this analysis are consistent with RG 1.183.

a) The analysis that supports the removal of this statement assumes a certain containment configuration and resulting leakage pathways out of containment.

Please confirm that this configuration provides the most bounding atmospheric dispersion factors for all possible release paths allowed by removing this statement. For example, if this statement is removed there would appear to be no controls on the configuration of containment. It might be possible that hatches or scuttles could be opened that would lead to a more direct leakage pathway to the control room. Please verify that your analysis bounds all possible containment configurations.

23) Page 5.0 of Calculation NE-02-04-08 states that the *X/Q values take into account that (the) release (of radioactivity) begins 24 hours after the accident*. The accident is typically assumed to begin when the fuel assembly is dropped and not at the time of shutdown. Please justify using a X/Q that models the accident starting at the shutdown of the reactor.

24) Please justify the removal of the fuel handling accident reference from the bases of technical specifications 3.3.6.2 (page B 3.3.6.2-12, "Secondary Containment Isolation Instrumentations.")

25) Page 82 of the submittal states:

Based on the overall reduction in CR operator dose due to AST methodology, similarities in ventilation systems, and the ability to evacuate the TSC, an updated quantitative assessment of the TSC dose based on the AST source term was not performed.

Likewise, the EOF doses were not reassessed based upon a quantitative assessment of the doses. The NRC staff requests further information regarding the quantitative assessment of the TSC and EOF doses.

a) Does Columbia currently have in its licensing bases commitments to meet GDC 19 for the TSC and EOF without compensatory actions such as evacuations?

b) Please provide enough detail regarding the quantitative assessment for the NRC staff to come to the same conclusion as the Columbia evaluation. Consider that the reduction in doses due the AST may be due to factors that are independent of the TSC and EOF doses (such as atmospheric dispersion factors etc. before isolation).

26) Please provide a justification for the removal of the sentences on pages B 3.3.7.1-1 and B 3.3.7.1-2.