ENERGY NORTHWEST

P.O. Box 968 ■ Richland, Washington 99352-0968

August 29, 2002
GO2-02-136

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

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397**
LICENSE AMENDMENT REQUEST -- ALTERNATIVE SOURCE TERM
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Reference: Letter GO2-01-156, dated December 3, 2001, RL Webring (Energy Northwest) to NRC, "License Amendment Request -- Alternative Source Term"

Dear Sir or Madam:

During discussions with the NRC, staff members requested additional information in support of their review of the referenced license amendment request. The information is included as an attachment to this letter.

Should you have any questions or desire additional information pertaining to this letter, please call Ms. CL Perino at (509) 377-2075.

Respectfully,

DW Coleman, Manager
Performance Assessment and Regulatory Programs
Mail Drop PE20

Attachment:

- Response to Request for Additional Information

cc: EW Merschoff - NRC RIV
BJ Benney - NRC NRR
NRC Senior Resident Inspector - 988C
DL Williams - BPA - 1399
TC Poindexter - Winston & Strawn

A001

Question 1: Justify the reduction in containment leakage after 24 hours. Ensure that the post-accident procedures do not provide any mechanism for re-pressurizing the containment.

Response

FSAR Figure 6.2-7 shows that the containment response with two residual heat removal system heat exchangers in operation reduces the primary containment pressure to a few psig in 24 hours. Since the containment leak rate is specified as 0.5%/day with the peak containment pressure of 37.4 psig, this pressure reduction is sufficient to reduce any containment leakage by greater than the 50% credited in the analysis. The potential for re-pressurization is eliminated by the Columbia design for control of the postulated non-condensable hydrogen and/or oxygen generation with two 100% redundant hydrogen recombiners to ensure the oxygen buildup does not reach a flammable level. In addition, the emergency operating procedures direct the operators to vent through the standby gas treatment system. Purging (addition of a non-condensable gas) is allowed only after the vent process is in place and at a rate equal or less than the standby gas treatment system flow rate. This precludes re-pressurization of the primary containment.

Columbia starts with nitrogen content sufficient to control oxygen content generated per 10CFR50.44 and if necessary using the vent with purge operation long term. This prevents re-pressurization of non-condensables long term (>24 hours) and validates the 50% reduction in leak rate. Some BWRs have hydrogen control systems where post-accident nitrogen is added (purging) to dilute the hydrogen and oxygen generated post-accident. Those plants could see some re-pressurization and would not take the full 50% reduction in leak rate.

Question 2: The fuel handling accident assumes a water depth of 22 feet above the fuel assemblies and that the DF provided by 22 feet is equivalent to the DF at 23 feet discussed in RG 1.183. Please justify this statement. Provide the methodology utilized to make this determination.

Response

A small difference in water depth is minimal compared to conservatism assumed in the DF credited to the spent fuel pool. For example, if a decontamination factor is calculated using an equation from the Westinghouse WCAP 7828, the form of the equation for DF is:

$DF = a\, e^{bt}$ where a is a constant and b is dependent on bubble diameter.

The bubble rise time, t, in this equation is dependent on water level. Although 22 feet is less conservative than 23 feet, both result in a DF greater than DF = 500. The difference is small, less than 10%, compared to the conservatism inherent in assuming a DF of 500 for elemental iodine (overall DF of 200) in accordance with Regulatory Guide 1.183.

The following additional conservative assumptions were incorporated into the FHA analysis:

- The 34 foot drop ignores the buoyancy of the falling bundle.
- The number of damaged fuel pins assumed is greater that the number that would be damaged by the forces calculated.
- The distance the fuel bundle drops if the FHA occurred in the spent fuel pool is less than 34 feet.
- Fewer pins would be damaged if the bundle were dropped in the spent fuel pool.

Question 3: Attachment 1, page 7 of 30 states that only the steam lines between the MSIVs are used for deposition. Are these lines seismically qualified?

Response

Yes, the four main steam lines are seismically qualified. The four lines from the reactor to the turbine stop valves are Seismic Category I. The main steam lines are analyzed for all loads as specified in FSAR Chapter 3; the loads include seismic OBE and SSE loadings.

Question 4: Attachment 2, page 4 of 20 states that significant progenies are included. Does this include decay of radionuclides while residing on the charcoal filters? If so how much dose is attributed to progeny?

Response

The complete list of decay chains utilized by STARDOSE is as follows:

Isotope	Parent
Y-90	Sr-90
Y-91	Sr-91
Y-92	Sr-92
Nb-95	Zr-95
Tc-99m	Mo-99
Rh-105	Ru-105
Te-127	Sb-127
Te-129	Sb-129
I-131	Te-131m
I-132	Te-132
Xe-133	I-133
Xe-135	I-135
Ba-137m	Cs-137
La-140	Ba-140
Ce-141	La-141
Pr-143	Ce-143

RADTRAD has additional decay chains represented; however, many of the daughter nuclides included in the RADTRAD decay schemes are not represented in the RADTRAD default nuclide files. Therefore, these nuclides have no impact. Moreover, RADTRAD 3.02 does not consider re-evolution from filters.

The only additional RADTRAD decay chains for which the daughters could have an impact are the following:

Isotope	Parent
Kr-85	Kr-85m
Te-127m	Sb-127
Te-129m	Sb-129
Te-127	Te-127m
Te-129	Te-129m
Pu-239	Np-239
Am-241	Pu-241
Pu-238	Cm-242
Pu-240	Cm-244

The metastable forms of Te are partially considered in STARDOSE because 100% of the Sb disintegrations are assumed to produce the non-metastable forms. To investigate the remaining five decay chains, a special library file was created for STARDOSE to include those chains. The Columbia LOCA dose analysis was rerun with those chains present. No impact was seen from adding these chains.

Decay of daughter products deposited on filters is included in the dose calculations. The STARDOSE code used to compute the dose has "re-suspension of daughter products" as a defined input fraction. The noble gas and gaseous iodine daughters are allowed to re-suspend from filters. The dose impact of the re-suspended daughter products may be determined by not allowing re-suspension in STARDOSE. It is observed when doing so that the dose impact is negligible for the EAB because of the little time available for decay. The dose impact for the LPZ and control room dose locations is $\leq$ 0.05 rem TEDE.

Question 5: Explain why the CRE unfiltered inleakage values differ for 2 trains and 1 train and between accidents. In table 7, for the control rod drop accident what is the basis for the control room unfiltered inleakage rate of 1800 cfm. Do the values given for unfiltered inleakage include 10 cfm to account for ingress and egress to and from the control room? If they do not the 10 cfm value traditionally used should be included. If the control room emergency filtration system allowable flow rate is between 900 cfm and 1100 cfm, justify why the use of 900 cfm is conservative for habitability determinations.

Response

The control room ventilation flow rate from outside air is 1000 cfm +/- 10%, and is the same whether or not the emergency filtration system is operating. For the control rod drop accident, the analysis does not take credit for the filtration, so the CREF flow rate is assumed as inleakage.

The measured control room in-leakage flow rate for the A train is 82 +/- 36 scfm, and the measured in-leakage flow rate for train B is 76 +/- 24 scfm as measured by tracer gas testing. The testing method restricted ingress/egress to a single door but did not interfere with normal control room traffic over the approximately 24 hours of testing performed for each train of the system. Therefore, the maximum measured in-leakage flow rate including ingress/egress is 118 scfm for one train. The analytical values assumed for each of the reported accidents are listed below.

Analytical CR HVAC Flow Values

Accident	Flow (cfm)	In-leakage (scfm)	Notes
CRDA	1800	1800	all flow is assumed to be in-leakage
FHA	1800	300	First 30 minutes
	900	150	30 min to 30 days
MSLB	1800	300	First 30 minutes
	900	150	30 min to 30 days
LOCA	1800	250	First 30 minutes
	900	125	30 min to 30 days

Columbia reduced the assumed control room in-leakage in the LOCA analysis in order to meet the dose acceptance criteria and, therefore, the in-leakage values assumed in the LOCA are the most limiting values. The control room ventilation flow rate was assumed to be the minimum design flow rate, i.e., 900 cfm per train, as this maximized the resident time of the radionuclides in the control room and maximized the dose to the control room operators.

Question 6: The Fuel Handling Accident takes credit for automatic initiation of the CREF system. Describe how this works and justify the value of 5 minutes.

Response

The Control Room Emergency Filtration System (CREF) is required to be operable during Modes 1, 2, 3 and during core alterations or movement of irradiated fuel assemblies in secondary containment. With reactor building ventilation operating, a release of radioactivity would reach the reactor building exhaust radiation monitors within seconds, initiating CREF. It would take less than 10 seconds at full flow to traverse the ducting to the release point. By assuming a very conservative five minutes to CREF operation, any anomaly delaying CREF initiation can be accounted for without detailed time line scenario development. The analysis, therefore, assumes source term release at time zero, no secondary containment isolation, no standby gas treatment system drawdown and filtration, and maximum control flow unfiltered for five minutes until CREF starts filtering inflow, except for the specified inleakage which remains unfiltered.

This is very conservative when it is expected CREF would initiate in less than half a minute.

Question 7: Per RG 1.183, consideration should be given to design leakage through valves isolating ESF recirculating systems from tanks vented to the atmosphere. In Attachment 2 page 13 of 20, the dose from ESF leakage to the CST is said to have a negligible contribution to dose. The calculation shows that this contribution pathway contributes 8% to the control room dose and 2% to the LPZ dose. These contributions are not considered negligible to the staff and should be added to the total dose.

Response

The values in Attachment 4 of the calculation referred to as contributing 8% of the control room dose and 2% of the LPZ dose were part of a sensitivity study in which Energy Northwest and Polestar compared the AST methodology to the existing results with the TID methodology. The sensitivity study used unrealistic conservative assumptions of leak rate and atmospheric dispersion that were not part of the final determination (see below).

In Attachment 2 of the submittal, Page 13 of 20, two different liquid leakages paths from the primary are discussed. As discussed in Attachment 1, page 12 of 30 of the submittal, the ESF leakage is from the recirculating ECCS systems into the secondary containment. The liquid leakage is collected in the reactor building sumps. The leak rate assumed was twice the design value and the partitioning coefficient assumed was 10%. The vapor leakage is filtered by the standby gas treatment system after building drawdown prior to release. These assumptions meet the guidance provided in Regulatory Guide 1.183 for ESF leakage.

Also discussed in Attachment 1, page 12 of 30, the secondary containment bypass leakage is the primary containment vapor that leaks directly to the environment, bypassing the secondary containment. The vapor paths contribute to the dose and are accounted for in the analysis. The secondary containment liquid bypass leakage paths (which include but are not limited to ECCS paths) are not part of our current design basis. The leakage paths were initially evaluated using TID source terms. This evaluation supported excluding the secondary containment liquid leakage from the offsite does consequences.

Regulatory Guide 1.183 states: "Consideration should be given to design leakage through valves isolating ESF recirculation systems from tanks vented to the atmosphere." This leakage out of ESF systems, through double isolation valves to tanks vented to the atmosphere was considered and evaluated using AST. The analysis showed the dose contribution of liquid bypass paths to be negligible (less than 1% of the RG 1.183 limits). This information provides additional support to the Energy Northwest determination that the secondary containment liquid bypass leakage is not part of the current license basis for Columbia Generating Station.

In summary, the contribution of the ESF leakage to the control room and the LPZ were not added to the total dose because:

- Values cited by the staff were not derived from the design bases dose calculations;
- The calculated contribution from the ESF leakage to the dose consequences is negligible; and
- Secondary containment liquid bypass leakage is not part of the Columbia Generating Station design basis.

Question 8: Attachment 1 page 4 of 30 states that the source term used is comprised of a 66 isotope subset that is different from that given in NUREG/CR-4691. Energy Northwest determined that the difference between the two sets was negligible. Please provide the difference in dose between the two sets and justify that this difference is negligible.

Response

The differences between the NUREG/CR-4691 isotope subset and Columbia's can be categorized as two types; neglect of activation products that are not part of the fuel, and addition of noble gases that do have an impact. The dose impact of neglecting Co-58 and Co-60 is approximately 5E-4 rem TEDE to the control room. The dose impact of adding Kr-83m, Kr-89, Xe-131m, Xe-133m, Xe-135m, Xe-137, and Xe-138 is the addition of approximately 4E-3 rem TEDE to the control room. The resulting changes are more conservative than the NUREG/CR-4691 (or the NUREG/CR-6604; i.e., RADTRAD) isotope subset, and this isotope subset also more accurately reflects the contents of the fuel.

Question 9: The LOCA calculation (PSAT 206CT.QA.01.10 "onsite and control room dose calculation for Energy Northwest's Columbia Generating Station using the Alternate Source Term and RADTRAD Methodology") states that the Sr-90 (strontium) dose conversion factor used in the Columbia LOCA analysis is different from the value used in the RADTRAD code. Justify the Sr-90 dose conversion factor used by Energy Northwest. Provide proof that the isotopic Sr source is not in the form of $SrTiO_3$.

Response

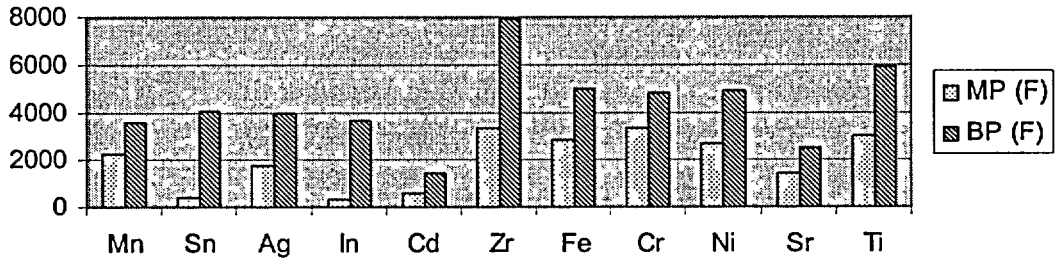
The expected chemical form for the release of fission product strontium in a core damage accident is SrO. This is primarily the result of the moderate volatility of strontium (moderate release rate from an overheated core) and its very high oxidation potential (eighth lowest electro-negativity). The SrO form of fission product strontium is the form assumed in all versions of the MAAP code, and it is the form previously assumed in NRC and NRC-sponsored severe accident studies. The dose conversion factor (DCF) used for strontium in all versions of the NRC-sponsored MACCS code (e.g., NUREG/CR-6613) is consistent with the oxide form, not with the titanate form. The DCF consistent with the oxide form is the one that has been used for Columbia.

The potential for fission product strontium being released in the chemical form $SrTiO_3$ is affected by two factors: (1) the availability of titanium in a core damage accident and (2) the volatility of $SrTiO_3$ relative to that of SrO. These factors are addressed in what follows.

Availability of Titanium for Formation of $SrTiO_3$

Titanium is a trace constituent of some austenitic stainless steels that is added to reduce susceptibility to inter-granular corrosion. It is generally less than 0.5% of the steel composition. This may be compared to other minor constituents such as manganese, which is typically two percent. Oxides of manganese and of tin (a minor constituent of zircaloy) are two of the major components of structural aerosols (along with silver, indium, and cadmium from control rods for certain plants) predicted to be released along with fission products as a result of a core damage accident. Manganese, tin, silver, indium, and cadmium are important contributors to the predicted structural aerosol release as opposed to the much more abundant zirconium, iron, chromium, and nickel, because of the relative volatility of these materials. The melting points and boiling points of manganese, tin, silver, indium and cadmium (the predicted components of core damage structural aerosols) are compared to those of the much more abundant zirconium, iron, chromium, and nickel in Figure 1 (in degrees F). Also shown are the melting and boiling points of strontium and titanium.

Figure 1- Melting and Boiling Point Comparison



As indicated by a comparison of the boiling points, the volatility of strontium is greater than all of the trace alloy and control rod materials except for cadmium. Strontium, therefore, is expected to be released earlier and in relatively larger amounts than most of the volatile structural materials. Titanium, on the other hand, has a boiling point substantially greater than all of the materials shown (including all of the major structural materials that are predicted to make up only a small part of the released structural aerosol because of their low volatility) except for zirconium. Therefore, very little titanium metal is expected to be released to the gas phase.

The low-volatility titanium metal will exist primarily in the metallic phase of any molten core debris while SrO will exist in the ceramic phase. It is unlikely that these two materials, both at very low concentrations and in separate phases, will react in the condensed state. TiO_2 may be oxidized from molten steel and be released to the gas phase (as may very small amounts of titanium metal). In the vapor state, SrO and Ti, or TiO2, would be very, very dilute so the rate of reaction would be strongly suppressed. Because of: 1) the trace amounts of titanium; 2) the substantial difference in volatility between it and strontium (approximately 3400 F difference in boiling point); and 3) the ready oxidation of strontium; the expected chemical form of fission product strontium is SrO.

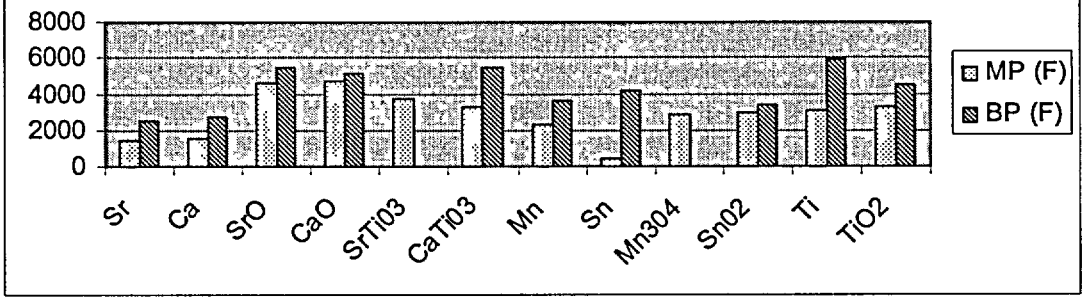
Relative Volatility of SrO and $SrTiO_3$

Not withstanding the above, it remains a possibility that some $SrTiO_3$ might be formed in small amounts. The question then arises, "Is it possible that, because of a difference in volatility, $SrTiO_3$ might be released in relatively large fractional amounts as compared to the release of SrO?"

The answer to this question is complicated by a lack of data of boiling point data on $SrTiO_3$. This material is used extensively in the manufacture of high-density computer memory, but Polestar was unable to obtain boiling point information. However, it is possible to infer its physical behavior by a chemical analogy to calcium.

Figure 2 shows a comparison of the available melting point and boiling point data for Sr, Ca, SrO, CaO (lime), $SrTiO_3$ and $CaTiO_3$ (perovskite). Also shown for comparison are the metallic and oxide forms of Mn, Sn, and Ti.

Figure 2 - Melting and Boiling Point Comparison



From Figure 2, it can be seen the metallic and oxide forms of strontium and calcium are quite comparable in terms of melting point and boiling point. The boiling point for $CaTiO_3$ is somewhat higher than that for CaO, and one would expect similar behavior for strontium. This indicates that, if anything, the fractional release of $SrTiO_3$ might actually be less than the fractional release of SrO; it would certainly be expected to be similar. Therefore, there would be no preferential release of $SrTiO_3$.

With respect to the release of structural oxides, it is noted that MnO_2 decomposes to Mn_3O_4 at about 1000 F. No boiling point information for Mn_3O_4 was found, but the boiling point for SnO_2 is observed to be lower than that of Sn, adding to its fractional release. The same behavior is observed for TiO_2 (rutile), in fact, the volatility of TiO_2 is greater than that of SrO. Therefore, what little metallic titanium is available in the molten core debris will most likely be released subsequently as TiO_2 with a fractional release greater than that of SrO and $SrTiO_3$. The difference in volatility between SrO and TiO_2 also serves to reduce the availability of titanium and the likelihood for the formation of $SrTiO_3$ in the ceramic phase.

Summary

$SrTiO_3$ is not expected to be a major part of the Sr release. Rather, the expected chemical form (as anticipated by the severe accident research community for two decades) is SrO. This supports our decision to have used the CEDE DCF for the oxide form of Sr-90 consistent with the previously published NRC MACCS code guidance.

Question 10: The Technical Specifications allowable flow rate for the standby gas treatment system is from 4500 cfm to 5500 cfm. Justify the use of 5000 cfm in the design bases accident analysis. The use of 5000 cfm yields non-conservative doses.

Response to Question 10

Higher standby gas treatment system flow rates result in the activity being released quicker to the atmosphere but result in shorter drawdown times (i.e., less time for the unfiltered portion of the release). Lower standby gas treatment system flow rates result in lower release rates but result in more time for drawdown and unfiltered release. Previous studies with the TID source term showed slightly less than 5000 cfm was the most conservative standby gas treatment system flow rate. Therefore, 5000 cfm was used in the AST analysis.

A sensitivity study was done with the AST varying the standby gas treatment system flowrate from 4500 to 5500 cfm. Ignoring the conservatism of a shorter drawdown time, the change in SGT flow rate did not change the reported dose values for the LOCA analysis. The FHA has a forced release rate over a two-hour period independent of standby gas treatment system flow, and the CRDA and MSLB do not credit secondary containment and standby gas treatment system operation. Therefore, use of 5000 cfm in the AST analysis is acceptable.

Question 11: For the CRDA please confirm that there are no forced flow paths from the turbine or condenser, such as unisolated motor vacuum pumps or unprocessed air ejectors that would provide an additional pathway for radiation to the environment.

Response

There are no forced flow paths from the turbine. The mechanical vacuum pumps are used to draw a vacuum on the main condenser when steam pressure is inadequate for air ejector operation, and are limited to less than 5% power. The steam jet air ejectors are in use from less than 5% to full power operation. The steam jet air ejectors remove non-condensables from the condenser, but process the flow through the off-gas system prior to release from the elevated release point.

Each of the two mechanical vacuum pumps can be isolated on the suction and discharge sides. Gasses in the vacuum pump common are monitored by a radiation element that is installed in the air removal discharge piping to the reactor building elevated release duct (a common exhaust line to the mechanical vacuum pumps and the gland seal steam condenser exhaust).

A high-radiation signal from the monitor will trip both mechanical vacuum pump motors. The mechanical vacuum pump trip, in turn causes the suction and discharge valves to close and trips the mechanical vacuum pump seal water pumps. The safety related main steam line radiation monitor high-radiation signal also trips both mechanical vacuum pump motors. As mentioned previously, this causes the suction and discharge valves to close.

Therefore, there are no unaccounted for forced flow paths from the condenser during the CRDA.

Question 12: Explain how the source term for the control rod drop accident (CRDA) is calculated. Justify credit for reduction in the source term due to transport in the steamline and the condenser. Are the steamlines and condenser seismically qualified?

Response

In Attachment 2, Table 4, Items 3.3 and 3.4 of the submittal, it is noted that the release fractions for the control rod drop accident are calculated using direct percentages as specified in RG 1.183. Each isotope group activity is multiplied by: 1) the sum of the release fraction from gap to coolant and the release fraction from fuel to coolant; 2) the fraction of activity that reaches the condenser; and 3) the fraction of activity available for release to the environment to produce the total release fraction from the condenser. The condenser leaks to the environment at a rate of 1% per day during 24 hours.

The methodology and inputs for the pin damage in this analysis are exactly as presented in the current Columbia licensing basis. No reduction was credited in the steam lines or the condenser other than the fractional release from the condenser permitted by RG 1.183, but decay in the condenser was allowed. The steam lines are seismically qualified, but the condenser is not.

Question 13: Justify the use of 40% of the reactor building volume being credited for mixing during the calculation of the design basis accident doses.

Response to Question 13

Analytical results using our GOTHIC model justifying the use of 40% mixing credit in the reactor building was presented to the staff in Letter GO2-99-107 dated June 10,1999, DW Coleman (EN) to NRC, "Request for Amendment to Secondary Containment and Standby Gas Treatment System Technical Specifications (Supplemental Information)." As shown in the analysis, in a post-accident environment, the HVAC ducts provide flow paths to all parts of the building. Also, some forced diffusion occurs because of the standby gas treatment system operation. Releases on one side of the reactor building tend to rise through all sections of the building on that side. Less mixing occurs around the primary containment on the same floor as the release when normal building ventilation is not in operation. Mixing values between 20 and 50% were examined; 40% provided the best estimate of mixing with standby gas treatment system operational due to passive design features of the station.

Question 14: Provide a table of the fission products used as source terms and their activities. Explain and justify your conformance to RG 1.183 Section 3.1 if the long lived isotopes are adjusted for 24 month cycles (as described in Attachment 2 page 1) rather than calculated by the ORIGEN-2 code. Provide enough of the methodology utilized so that the source term can be replicated. Explain what is meant by ORIGEN-2 based" in Table 1 of Attachment 2.

Response to Question 14

The source term isotopes and their activities are in Appendix B of the LOCA calculation, PSAT 206CT.QA.01.10. The first column is the isotope label, and the first column of numbers is the Ci/MW_{th} values.

Our conformance statement to RG 1.183 Section 3.1 is in Attachment 2 of the submittal letter. What is meant by "ORIGEN-2 based" in Table 1 of Attachment 2 is that it is a plant-specific pre-1995 ORIGEN-2 run that has been adjusted. The three adjustments were: 1) a power uprate scale factor of +5.28% to bound an uprate in power level from 3486 to 3556 MW_{th}; 2) a +25% correction for abnormally low krypton values (based on comparisons to other core inventory tables); and 3) a +60% increase in the activity of longer lived isotopes (half-lives greater than one year). This last correction is based on the ratio of the burn-up being assumed for the current calculation and that used as input to the available ORIGEN analysis (a ratio of 1.6). The one-year half-life threshold for applying

the 1.6 factor is based on the assumption that isotopes with less than one-year half-lives will have reached equilibrium in the core. The use of the 1.6 multiplier for isotopes with half-lives greater than one year is conservative. These source term changes result in a conservative set of source term isotopes.