Entergy

Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

June 15, 2007

10 CFR 50.90

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

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

Response to Request for Additional Information Regarding Alternative Source Term License Amendment Request

Dear Sir or Madam:

By letter dated September 25, 2006, Nuclear Management Company, LLC (NMC), the former licensee for the Palisades Nuclear Plant (PNP), requested Nuclear Regulatory Commission (NRC) review and approval of a proposed license amendment request for PNP. The proposed license amendment modifies the PNP licensing basis to adopt the alternative source term methodology.

By electronic mail dated December 5, 2006, February 26, 2007, and March 29, 2007, the NRC sent requests for additional information (RAI) on the proposed amendment. On April 2, 2007, a teleconference was held with the NRC staff to discuss the RAIs. Enclosure 1 provides the responses to the RAIs.

Summary of Commitments

This letter contains no new commitments and one revision to existing commitments.

Commitment made by letter dated September 25, 2006:

2. Palisades will implement three distinct plant modifications to support the assumptions used in the radiological dose analysis:

    c. Installation of an alternate power source to allow the cross-tie of the low pressure safety injection suction piping.

This commitment is being revised because it was determined that the cross-tie of the low pressure safety injection suction piping can be achieved procedurally, as opposed to requiring a physical plant modification. Therefore, the commitment is being revised as follows:

A001

NRR

Revised commitment:

ENO will modify plant emergency operating procedures to allow the cross-tie of the low pressure safety injection suction piping post loss-of-coolant-accident following recirculation, prior to entering Mode 3 from the 2007 refueling outage at PNP.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 15, 2007.

Christopher J. Schwarz
Site Vice President
Palisades Nuclear Plant

Enclosure (1)

CC Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC

Requests for additional information received by electronic mail December 5, 2006.

Nuclear Regulatory Commission (NRC) Request

1. *On Section 2.1.2 of your application you stated that the sump pH is controlled at a value greater than 7 based on the addition of tri-sodium phosphate (TSP) baskets or an alternate buffer. Clarify what type of alternate buffer could be used to control containment sump pH.*

Entergy Nuclear Operations, Inc. (ENO) Response

1. Several alternate buffers could be used. ENO plans to submit a license amendment request for Palisades Nuclear Plant (PNP) prior to the 2007 refueling outage to replace the current TSP buffer with the alternate buffer sodium tetraborate (STB). The choice of buffer was based in large part on information contained in WCAP-16596-NP, "Evaluation of Alternative Emergency Core Cooling System Buffering Agents."

NRC Request

2. *In order to complete its evaluation, the NRC staff needs to review the general assumptions and calculations used to prove that the containment sump pH will be maintained above 7 for 30 days following a Loss-of-Coolant Accident (LOCA). Provide this information in sufficient detail for the NRC staff to perform independent calculations to evaluate the licensee's conclusion (if different buffers could be used, provide the information requested for each buffer). If the calculations were performed manually, describe the methodology and provide sample calculations. If a computer code was used, describe the code and provide the input values and how they were determined. Provide the results of pH calculations at different time intervals and explain how the time intervals were selected.*

ENO Response

2. The mass of STB required to raise the pH of the borated water in the containment sump to within a design range of 7.0 to 8.0 (analytical pH range of 7.0 to 7.8) for post-LOCA conditions has been calculated.

   A parametric analysis was performed to determine the required STB mass as a function of the quantity of borated water in the sump, the boron concentration in the sump, the sump water temperature, and the desired equilibrium pH value. The analysis was performed using equilibrium equations derived in the calculation which were iteratively solved with a computer algorithm developed by Sargent and Lundy, LLC, specifically for this purpose.

The dissolution rate of STB is acceptable if it is equal to or greater than the dissolution rate of TSP currently in use in the baskets. Dissolution rates for STB and TSP were compared using a standardized surface dissolution rate (SDR) that is characteristic of the material and is experimentally determined.

The analysis included computation of:

- Boron speciation
- Water dissociation
- Hydrochloric acid and nitric acid addition from radiolysis
- Iodine and cesium addition from core inventory

The equilibrium conditions for the resulting solution were determined as a function of STB addition and as a function of temperature. The evaluations were based on a steady state analysis of equilibrium for conditions that bound post-recirculation times. Boron species considered were $B(OH)_3$ (boric acid, H_3BO_3), $B(OH)^{-}_{4}$, $B_2(OH)^{-}_{7}$, $B_3(OH)^{-}_{10}$, and $B_4(OH)^{2-}_{14}$. Water dissociation was determined with activity coefficients using the extended form of Debye-Hückel theory. Total beta and gamma production of hydrochloric acid from radiolysis of cabling was calculated to be 1095 g-mole. Production of nitric acid from radiolysis of air and water was calculated to be 1.15E-3 g-mole/kg water. The total amount of iodine from core inventory was calculated to be 82.1 g-mole. Most of this iodine (95%) would be in the form of cesium iodide. The total amount of cesium from core inventory was calculated to be 407.6 g-mole. Some of this cesium would be in the form of cesium iodide. All iodine and cesium added from core inventory were included in the equilibrium computations to account for effects on ionic strength.

The required minimum mass of pure STB decahydrate to achieve a pH of 7.0 for the bounding condition of a maximum boron concentration (2790 ppm B), maximum borated water (2,565,532 lb), and minimum water temperature (77°F) was calculated to be 8186 lbs. The required maximum mass of pure STB decahydrate to achieve a pH of 7.8 for the bounding condition of a minimum boron concentration (1560 ppmB), minimum borated water (2,488,533 lb), and maximum water temperature (268°F) was calculated to be 10,553 lbs.

NRC Request

3. The discussion of backleakage to the safety injection refueling water tank (SIRWT) in Section 2.1.3 of the application provides the maximum pH, maximum iodine concentration and maximum elemental iodine fraction of the SIRWT at 30 days. Provide the general assumptions and calculations used to determine the maximum pH, maximum iodine concentration and maximum elemental iodine fraction of the SIRWT at 30 days. Provide this information in sufficient detail for the NRC staff to perform independent calculations to evaluate the licensee's conclusion. If the calculations were performed manually, describe the methodology and provide sample calculations. If a computer code was used, describe the code and provide the input values and how they were determined.

ENO Response

3. The SIRWT leakage model accounts for the dose due to leakage of sump fluid back into the SIRWT which subsequently is released to the environment through the SIRWT tank vent. The calculation of this dose is a complex process involving many variables. The sump fluid that leaks back into the SIRWT is assumed to mix with liquid in the SIRWT. The fraction of elemental iodine in the SIRWT is a function of the SIRWT pH and total iodine concentration. The elemental iodine in the SIRWT fluid is then assumed to enter the SIRWT air space as a function of the iodine partition coefficient. The iodine partition coefficient is a function of the SIRWT liquid temperature. The iodine in the SIRWT air space is then available for release via the SIRWT vent.

The flashing fraction for the leaking emergency core cooling system (ECCS) fluid is 0.02953, at the beginning of recirculation. Flashing would cease at approximately 4.9 hours into the event. Since the flashing fraction is very low and since flashing ceases at 4.9 hours, it is assumed that all of this backleakage would condense within the pipe leading into the SIRWT and mix with the water inventory of the tank. The duration of 4.9 hours was confirmed via a GOTHIC model, which was used to determine the heat loss from the leaking fluid as it traveled to the SIRWT.

The situation is further complicated by operator actions to partially refill the SIRWT post-LOCA. Water make-up to the SIRWT would likely have a low pH due to the presence of boric acid, which increases the SIRWT iodine volatile fraction; a low temperature (relative to backleaked sump fluid), which increases the SIRWT iodine partition coefficient; a high rate of addition (relative to backleakage rate), which increases the SIRWT vent release rate; and very low iodine concentration (relative to the sump fluid), which decreases SIRWT iodine concentration and, therefore, the iodine volatile fraction. Also, SIRWT water may leak out of the tank. With respect to shielded dose calculations, depending on the initial volume of water in the SIRWT and the rate of backleakage to the SIRWT the scenario in which all leaked back activity is retained in the tank may not bound the scenario in which outleakage results in the minimum water level in the tank for the duration of the event. As a result, two sensitivity cases are analyzed in addition to the design-basis case to ensure the limiting scenario has been considered for both the SIRWT leakage dose and shielded dose contributors. The design basis case assumes no SIRWT refill and that all activity leaked into the tank is retained, i.e., with no SIRWT outleakage. Additional sensitivity cases are analyzed: one assuming a bounding SIRWT refill and no SIRWT outleakage, and one assuming no SIRWT refill and full SIRWT outleakage.

Section 6.3 of calculation NAI-1149-014, "Palisades Design Basis AST MHA/LOCA Radiological Analysis," revision 3 (ADAMS Accession Number ML062830447), provided with the original submittal contains the detailed inputs, assumptions and calculations used to determine the pH, iodine concentration, elemental (volatile) iodine fraction of the SIRWT, and the resultant SIRWT iodine release over time. This information is provided in sufficient detail for the staff to perform independent calculations.

<u>Requests for additional information received by electronic mail February 26, 2007.</u>

NRC Request

1. *The AST dose consequence analyses are based on a power level of 2703 megawatt thermal (MWt) representing the original full power design rating of 2650 MWt, with a 2% margin for uncertainty. It appears that certain parameters used in the AST evaluations may be based on information from tables in the final safety analysis report (FSAR) with listed power levels other than 2703 MWt.*

1.1 *Please provide additional information to clarify whether the 2703 MWt power level used in the AST dose consequence analyses for the evaluation of the radiological source term, applies to all other power related aspects of the evaluations.*

ENO Response

1.1 Only the ORIGEN inventory calculations were done at 2703 MWth. All other (thermal-hydraulic) calculations are based on power level of 2580.6 MWth, which is the current licensed power level of 2565.4 MWth including the current licensed calorimetric uncertainty of 0.5925%. Note that primary coolant system (PCS) and secondary side activities are based on ORIGEN inventories at 2703 MWth, adjusted for equilibrium release/cleanup rates and normalization to Technical Specification (TS) dose equivalent I-131 and 100/E-bar. See Table 1 below for additional detail.

Therefore, the power level of 2703 MWth applies only to the core inventory calculations and does not apply to any other power related aspects of the AST submittal. Use of the power level of 2703 MWth for the AST core inventory calculations is not intended to imply that the AST calculations are valid for power levels other than the currently licensed power of 2580.6 MWth.

NRC Request

1.2 Please provide additional information to clarify the limiting power level for each of the AST dose consequence analyses considering all power related aspects of the evaluations.

ENO Response

1.2 The limiting power level of 2580.6 MWth, which is the current licensed power level of 2565.4 MWth including the current licensed calorimetric uncertainty of 0.5925%, is supported by the AST analyses.

Table 1. Radiological Consequences for FSAR Chapter 14 Analyses

FSAR Section	Event	Radiological Analyses	In AST Submittal?	Comments
14.2	UNCONTROLLED CONTROL ROD WITHDRAWAL	N/A Radiological dose consequence analysis not applicable since event does not result in failed fuel, breaches of fission product barriers, or ESF actuation.	No	N/A
14.3	BORON DILUTION	N/A Radiological dose consequence analysis not applicable since event does not result in failed fuel, breaches of fission product barriers, or ESF actuation.	No	N/A
14.4	CONTROL ROD DROP	N/A Radiological dose consequence analysis not applicable since event does not result in failed fuel, breaches of fission product barriers, or ESF actuation.	No	N/A
14.5	CORE BARREL FAILURE	N/A Radiological dose consequence analysis not performed since consequences are bounded by another event (control rod ejection).	No	N/A
14.6	CONTROL ROD MISOPERATION	N/A Radiological dose consequence analysis not applicable since event does not result in failed fuel, breaches of fission product barriers, or ESF actuation.	No	N/A

Table 1. Radiological Consequences for FSAR Chapter 14 Analyses

FSAR Section	Event	Radiological Analyses	In AST Submittal?	Comments
14.7	DECREASED REACTOR COOLANT FLOW	N/A Radiological dose consequence analysis not applicable since event does not result in failed fuel, breaches of fission product barriers, or ESF actuation.	No	N/A
14.8	START-UP OF AN INACTIVE LOOP	N/A Radiological dose consequence analysis not performed since the event not considered credible.	No	N/A
14.9	EXCESSIVE FEEDWATER INCIDENT	N/A Radiological dose consequence analysis not performed since consequences are bounded by another event (increase in steam flow).	No	N/A
14.10	INCREASE IN STEAM FLOW (EXCESS LOAD)	N/A Radiological dose consequence analysis not applicable since event does not result in failed fuel, breaches of fission product barriers, or ESF actuation.	No	N/A
14.11	POSTULATED CASK DROP ACCIDENTS	Yes	Yes	LTR Table 2.7-1 power level of 2703 MW_{th} applies to core inventory calculation with ORIGEN. Results not impacted by reactor power level during event.

Table 1. Radiological Consequences for FSAR Chapter 14 Analyses

FSAR Section	Event	Radiological Analyses	In AST Submittal?	Comments
14.12	LOSS OF EXTERNAL LOAD	N/A Radiological dose consequence analysis not applicable since event does not result in failed fuel, breaches of fission product barriers, or ESF actuation.	No	N/A
14.13	LOSS OF NORMAL FEEDWATER	N/A Radiological dose consequence analysis not applicable since event does not result in failed fuel, breaches of fission product barriers, or ESF actuation.	No	N/A
14.14	STEAM LINE RUPTURE INCIDENT	Yes	Yes	LTR Table 2.3-1 power level of 2703 MW_{th} applies to core inventory calculation with ORIGEN only. Thermal-hydraulic calculation (HFP SG inventory, steam release for cooldown, etc.) based on power level of 2580.6 MW_{th} (current licensed power level of 2565.4 MW_{th} with uncertainty 0.5925%). PCS and secondary side activity based on ORIGEN inventories, equilibrium release/cleanup rates, and normalization to TS DEI-131 and 100/E-bar.

Table 1. Radiological Consequences for FSAR Chapter 14 Analyses

FSAR Section	Event	Radiological Analyses	In AST Submittal?	Comments
14.15	STEAM GENERATOR TUBE RUPTURE WITH A LOSS OF OFFSITE POWER	Yes	Yes	LTR Table 2.4-1 power level of 2703 MW_{th} applies to core inventory calculation with ORIGEN only. Thermal-hydraulic calculation (HFP SG inventory, steam release for cooldown, etc.) based on power level of 2580.6 MW_{th} (current licensed power level of 2565.4 MW_{th} with uncertainty 0.5925%). PCS and secondary side activity based on ORIGEN inventories, equilibrium release/cleanup rates, and normalization to TS DEI-131 and 100/E-bar.

Table 1. Radiological Consequences for FSAR Chapter 14 Analyses

FSAR Section	Event	Radiological Analyses	In AST Submittal?	Comments
14.16	CONTROL ROD EJECTION	Yes	Yes	LTR Table 2.6-1 power level of 2703 MW_{th} applies to core inventory calculation with ORIGEN only. Thermal-hydraulic calculation (HFP SG inventory, steam release for cooldown, etc.) based on power level of 2580.6 MW_{th} (current licensed power level of 2565.4 MW_{th} with uncertainty 0.5925%). PCS and secondary side activity based on ORIGEN inventories, equilibrium release/cleanup rates, and normalization to TS DEI-131 and 100/E-bar.
14.17	LOSS OF COOLANT ACCIDENT	N/A Radiological dose consequence analysis not performed since consequences are bounded by another event (MHA).	No	N/A
14.18	CONTAINMENT PRESSURE AND TEMPERATURE ANALYSIS	N/A Radiological dose consequence analysis not performed since consequences are bounded by another event (MHA).	No	N/A

Table 1. Radiological Consequences for FSAR Chapter 14 Analyses

FSAR Section	Event	Radiological Analyses	In AST Submittal?	Comments
14.19	FUEL HANDLING INCIDENT	Yes	Yes	LTR Table 2.2-1 power level of 2703 MW_{th} applies to core inventory calculation with ORIGEN. Results not impacted by reactor power level during event.
14.20	LIQUID WASTE INCIDENT	Yes	No	Not an accident analysis. Design ensures 10 CFR 20 limits are met.
14.21	WASTE GAS INCIDENT	Yes	No	No changes were proposed to the waste gas or volume control systems in the AST LAR. Analysis of the waste gas decay tank rupture and volume control tank rupture not included in the AST LAR based on guidance in RIS 2006-04.

Table 1. Radiological Consequences for FSAR Chapter 14 Analyses

FSAR Section	Event	Radiological Analyses	In AST Submittal?	Comments
14.22	MAXIMUM HYPOTHETICAL ACCIDENT	Yes	Yes	LTR Table 2.1-1 power level of 2703 MW_{th} applies to core inventory calculation with ORIGEN only. Thermal-hydraulic calculation (HFP SG inventory, steam release for cooldown, etc.) based on power level of 2580.6 MW_{th} (current licensed power level of 2565.4 MW_{th} with uncertainty 0.5925%).
14.23	RADIOLOGICAL CONSEQUENCES OF FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE CONTAINMENT	Yes	Yes	LTR Table 2.5-1 does not indicate a power level. Inventory calculations with ORIGEN based on 2703 MW_{th}. PCS activity based on ORIGEN inventories, equilibrium release/cleanup rates, and normalization to TS DEI-131 and 100/E-bar.
14.24	CONTROL ROOM RADIOLOGICAL HABITABILITY	Yes	Yes	Arguments for specific events discussed above apply to the control room dose calculations as well.

NRC Request

2. The staff will address the acceptability of the retention of the TID-14844 source term for both equipment qualification (EQ) purposes and for NUREG-0737 analyses other than for the control room envelope (CRE), based on the resolution of Generic Issue 187 and the logic provided in Enclosure 1 of the license amendment request (LAR). However, the staff is concerned that such an action may be interpreted as an acceptability for these analyses at the AST power level, especially if the AST power level exceeds the power level used in the EQ and NUREG-0737 analyses. Therefore:

2.1 Please verify the core thermal power level and the associated uncertainty used in the licensing basis EQ dose evaluations and the relationship, if any, to the power level used in the AST LAR.

ENO Response

2.1 The environmental equipment qualification (EEQ) analyses support power levels of 2580.6 MWth (2565.4 MWth + 0.5925%). No relationship between the AST core inventory calculation power level of 2703 MWth, and EEQ power level assumptions exists. Use of the power level 2703 MWth for AST core inventory calculations is not intended to imply that the EEQ calculations are valid for power levels other than the currently licensed power.

NRC Request

2.2 Please verify the core thermal power level and the associated uncertainty used in the licensing basis NUREG-0737 dose evaluations and the relationship, if any, to the power level used in the AST LAR.

ENO Response

2.2 NUREG-0737, "Clarification of TMI Action Plan Requirements," II-B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-Accident Operations," analyses support power levels of 2580.6 MWth. No relationship between the AST core inventory calculation power level of 2703 MWth and NUREG-0737 power level assumptions exists. Use of the power level of 2703 MWth for AST core inventory calculations is not intended to imply that the NUREG-0737 calculations are valid for power levels other than the currently licensed power.

NRC Request

3. The loss-of-coolant accident (LOCA) analysis credited an elemental iodine wall deposition removal coefficient of 2.3 hr^{-1} for the duration of the accident. EA-PAH-91-06, Revision 2, Fission Product Removal Coefficients for Design Basis Radiological Consequence Analyses, September 2006, was cited as the reference for the elemental iodine wall deposition removal coefficient of 2.3 hr^{-1}.

3.1 The cited reference is not readily available to the staff. Please provide additional information describing the technical basis for the elemental iodine wall deposition removal coefficient of 2.3 hr^{-1}.

ENO Response

3.1 Per Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," three forms of iodine are assumed to exist in a post-LOCA containment atmosphere: elemental, particulate, and organic iodine. Consistent with NUREG-0800, "Standard Review Plan," Section 6.5.2 (Section III.4.c (3)) no removal mechanism is modeled for organic iodine.

Four processes govern the removal of the other two forms of iodine and other particulates (aerosols) from the containment atmosphere: wall deposition or plate-out of elemental iodine, spray absorption of elemental iodine, natural deposition of particulate iodine and other particulates (aerosols), and spray washout of particulate iodine and other particulates (aerosols).

The four removal coefficients are defined as follows:

- λ_w models the removal of elemental iodine by wall deposition
- λ_s models the removal of elemental iodine by containment sprays
- λ_n models the removal of particulates (including particulate iodine) by natural deposition
- λ_p models the removal of particulates (including particulate iodine) by containment sprays

Methods for calculating λ_w, λ_s, and λ_p are given in SRP 6.5.2. The basis for determining λ_n is given in the Industry Degraded Core Rulemaking Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents and Fission Product Cleanup System," revision 2.

The removal of fission products from the containment atmosphere in the dose consequence calculations are modeled using these four removal coefficients. Each coefficient models one of the removal processes described above. An excel spreadsheet was developed to facilitate the calculation of the spray removal coefficients.

From SRP 6.5.2 and previous analyses, six plant-specific, fundamental parameters that are subject to change as a result of plant modifications or re-analysis and that impact the removal coefficients are:

- Containment spray flow rate

  A reduction in spray flow rate causes the calculated values of the spray removal coefficients to decrease, increasing calculated doses. Therefore, lower values for spray flows are conservative for the calculation of spray removal coefficients.

- Containment spray droplet mass-mean diameter

  An increase in spray droplet mass-mean diameter causes the calculated value of the elemental iodine spray removal coefficient to decrease, increasing calculated doses. The particulate spray removal coefficient remains unaffected due to the proscribed nature of the collection efficiency to drop diameter (E/D) ratio specified in SRP 6.5.2 for particulate iodine spray removal coefficients. Therefore, higher values for mass-median drop diameters are conservative for the calculation of spray removal coefficients.

- Containment Atmosphere Peak Pressure

  An increase in pressure causes the calculated value of the elemental iodine spray removal coefficient to decrease, increasing calculated doses. The particulate spray removal coefficient remains unaffected due its dependence only on fall height and not fall time as specified in SRP 6.5.2. Therefore, higher values for pressure are conservative for the calculation of spray removal coefficients.

- Containment Atmosphere Peak Temperature

  A decrease in temperature causes the calculated value of the elemental iodine spray removal coefficient to decrease, increasing calculated doses. The particulate spray removal coefficient remains unaffected due its dependence only on fall height and not fall time as specified in SRP 6.5.2. Therefore, lower values for temperature are conservative for the calculation of spray removal coefficients.

- Containment Atmosphere Air Mass at Peak Conditions

  A reduction in air mass causes the calculated value of the elemental iodine spray removal coefficient to decrease, increasing calculated doses. The particulate spray removal coefficient remains unaffected due its dependence only on fall height and not fall time as specified in SRP 6.5.2. Therefore, lower values for air mass are conservative for the calculation of spray removal coefficients.

- Containment Atmosphere Steam Mass at Peak Conditions

  A reduction in steam mass causes the calculated value of the elemental iodine spray removal coefficient to decrease, increasing calculated doses. The particulate spray removal coefficient remains unaffected due to its dependence only on fall height and not fall time as specified in SRP 6.5.2. Therefore, lower values for steam mass are conservative for the calculation of spray removal coefficients.

Conservative, bounding values for these parameters are used to determine conservative removal coefficients.

The plate out of elemental iodine onto containment surfaces is accounted for by the first-order removal coefficient for wall deposition, λ_w. Following the SRP 6.5.2, λ_w can be estimated by the following:

$$\lambda_w = \frac{k_w A}{V} \qquad \textbf{(1)}$$

where:

A	=	wetted surface area, ft^2
V	=	containment building net free volume, ft^3
k_w	=	wall deposition mass-transfer coefficient, ft/hr.

In NUREG/CR-0009, "Technological Bases for Models of Spray Washout of Airborne Containments in Containment Vessels," pages 65 and 90, the wetted surface area is described as encompassing all containment surfaces that are wetted. Containment sprays will automatically be initiated following any event that generates a containment high pressure (CHP) signal. All surfaces in containment can be assumed wetted for CHP/CHR [containment high radiation] events in which sprays are activated. However, horizontal slabs and heated surfaces such as the steam generators and associated piping in containment will generally be excluded as a wetted surface. In addition, any surface area below the maximum sump flood plane elevation of 597' is assumed to be flooded and is also generally excluded.

From a listing of the consolidated heat sinks from the containment response analyses of record, a spreadsheet tabulates the wetted surface area in containment by using a 0 or 1 to exclude or include, respectively, the associated constituent heat sink for the surface area summation. Note that substantial conservatism in the heat sink consolidation calculation itself exists in that much in-containment surface area has been neglected in the underlying containment heat sink calculation. Also, a conservative judgment on inclusion/exclusion has been employed. In addition, the primary system drain tank has been excluded.

The conservatisms ensure that even though some portions of some surface areas that are included may contain some horizontal or heated areas, the overall result remains conservative. This results in a total wetted surface area of 233,343 ft^2.

From SRP 6.5.2, Page 6.5.2-11, the wall deposition mass-transfer coefficient, k_w, is conservatively taken as 4.9 m/hr, or 16.076 ft/hr. Using this value for k_w, the wetted surface area from above, and the containment net free volume of 1.64E+06 ft^3, the calculated value for the elemental iodine removal coefficient for wall deposition is:

$$\lambda_w = 2.3\ hr^{-1}$$

NRC Request

3.2 *Please provide additional information describing the technical justification for the use of the elemental iodine wall deposition removal coefficient of 2.3 hr^{-1} for the duration of the accident. In particular, please provide additional information describing the technical justification for the application of this coefficient during the period in which credit for elemental iodine removal from the operation of the containment sprays is taken currently.*

ENO Response

3.2 The elemental iodine spray removal mechanism is separate and distinct from elemental iodine wall deposition mechanism, as demonstrated by the prescription to sum the spray and wall deposition removal coefficients to determine overall removal (see SRP 6.5.2, Section III.4.b).

Elemental iodine spray removal is the result of the absorption of elemental iodine by spray drops, and is a function of spray flow rate, equilibrium partition coefficient applicable to spray absorption, absorption efficiency, and the volume of the sprayed region. The absorption efficiency is based on a model of absorption through liquid and gaseous films called the stagnant film model. This model forms the basis of the SRP 6.5.2 method for calculating elemental iodine spray removal coefficients. This model is predicated on an instantaneous source term and a cut-off concentration to account for equilibrium affects (see SRP 6.5.2, III.4.d). The instantaneous source term assumption is conservative with respect to the time dependent release of the AST (see NUREG/CR-0009, Section 6.1.2).

Elemental iodine wall deposition is the result of transport through the bulk gas phase, the gas boundary layer, the liquid film, and the solid wall surface. Transport in the gas boundary layer has been shown to be controlling (NUREG/CR-0009, Section 5.1.2). The model for calculating elemental iodine wall deposition is called Knudson-Hilliard, and views deposition as a gas film transport process, with gas film mass transfer coefficients based on natural convention heat transfer correlations and on mass transfer-heat transfer analogy. A conservative value for the gas film mass transfer coefficient is used as prescribed in SRP 6.5.2 (see response to RAI 3.1). No cut-off value is used since the overall absorption capacity of containment surfaces is much larger than that required to absorb all core iodine not retained by the molten core or sump fluid (NUREG/CR-0009, Section 6.1.9).

In summary, elemental iodine spray removal and wall deposition are distinct processes. The removal effects are additive when occurring simultaneously, as prescribed by SRP 6.5.2. Elemental iodine wall deposition process does not saturate and therefore does not require a cut-off.

Note that elemental iodine removal has less of an impact on dose results for AST-based source terms than for TID-based source terms, given the chemical forms prescribed by RG 1.183 (95% particulate, 4.85% elemental, 0.15% organic).

NRC Request

3.3 *The staff notes that the aerosol natural deposition coefficient of 0.1 hr^{-1} is specified only for the time period after containment spray credit ends. Please provide additional information describing the differences in the application of the natural removal mechanisms and their relationship to the containment spray removal assumptions. Also, please note that the table on page 17 of NAI-1149-014 Revision 3, describing the timing of the credited aerosol natural deposition coefficient of 0.1 hr^{-1}, appears to contain a typographical error in the title "Particulate Spray Removal Coefficients."*

ENO Response

3.3 See response to RAI 3.1. SRP 6.5.2 provides a method for calculating the spray washout of particulate iodine and other particulates (aerosols) but does not discuss the natural deposition of particulate iodine and other particulates (aerosols). The natural deposition removal coefficient of 0.1 hr^{-1} is assumed (based on the Industry Degraded Core Rulemaking Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," Atomic Industrial Forum, December 1983) for all aerosols. SRP 6.5.2 does not specify the summation of the natural deposition and spray washout coefficients for particulates as it does for wall deposition and spray removal of elemental iodine. Therefore, for conservatism these removal coefficients are not added together during the time of spray operation. Note that for particulates, the natural deposition coefficient is 0.1 hr^{-1} and the spray removal coefficient is 1.8 hr^{-1}, initially.

The terms "aerosol" and "particulate" are used interchangeably above. The use of the term "aerosol" has evolved and now embraces liquid droplets, solid particles, and combinations of these. As a common term, aerosol would be interpreted by the general population as meaning an aerosol spray can or output from such a can. In scientific terms it refers to airborne solid particles also called dust or particulate matter as well as liquid droplets.

The heading for the second table in Section 6.1.1.2 "Natural Deposition and Wall Deposition," of NAI-1149-014, revision 3, indicated as "Particulate Spray Removal Coefficient" is a typographical error and should be "Particulate Natural Deposition Coefficient."

NRC Request

4. *Containment purge is not considered as a means of combustible gas or pressure control in the AST LOCA analysis. In addition, routine containment purging is not active for the AST LOCA analysis.*

   Please provide additional information describing the controls that are in place to preclude the use of on-line containment purging and containment purging for post LOCA hydrogen control.

ENO Response

4. PNP does not perform routine containment purges. Clean waste receiver tank rupture disk, RUD-1018, is removed and continuous containment venting via control valves, CV-1064 and CV-1065, occurs on-line. This alignment is established by general operating procedure GOP-2, "Mode 5 to Mode 3 ≥525°F." Post-LOCA purge is not done unless directed by the Technical Support Center (TSC), via reference emergency operating procedure EOP-4, "Loss of Coolant Accident Recovery." Post-LOCA hydrogen is not a design basis issue as recombiners have been eliminated from the design basis. Online continuous venting is acceptable since first clad burst is at ~50 seconds for the limiting LOCA. RG 1.83 proscribes initial fuel failure to occur at 30 seconds, and CV-1064 and CV-1065, (and other containment isolation valves) are isolated in <30 seconds. Stroke time testing for CV-1064 and CV-1065 is performed by several TS surveillance procedures, including QO-5, "Valve Test Procedure (Includes Containment Isolation Valves)," RO-11, "Containment High Radiation Tests," and RO-12, "Containment High Pressure (CHP) and Spray System Test," to verify CV-1064 and CV-1065 close with the appropriate signal.

NRC Request

5. *Regulatory Guide (RG) 1.183, Appendix A, Regulatory Position 3.3 states that, "The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown." The AST LAR references Amendment No. 31 and states that, "Per the current licensing basis, there is at least 90% spray coverage of the containment (Reference 5.30); therefore, the containment is treated as a single well mixed volume." The safety evaluation (SE) for Amendment No. 31 used a containment free air volume of 1.64 million cubic feet and sprayed volume of 1.48 million cubic feet, thereby establishing the 90% spray coverage. However, the SE for Amendment No. 31 also assumed an air exchange between the unsprayed and sprayed regions of two unsprayed region volumes per hour.*

   Please provide additional information addressing the mechanisms credited for providing adequate mixing of the unsprayed compartments to substantiate the assumption of a single, well-mixed containment building atmosphere.

ENO Response

5. Adequate mixing is assured based on the operation of sprays, thermally driven natural convection currents, and blowdown induced flow. No credit for containment air cooling fans is taken.

   NUREG-0800, SRP 6.5.2, Section III.1.c indicates the containment building atmosphere may be considered a single well-mixed space if the spray covers at least 90 percent of the containment building space and if a ventilation system is available for adequate mixing of any unsprayed compartments. However, RG 1.183, Appendix A, Section 3.3, states that the containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown, i.e., dropping the requirement for a ventilation system.

   There are several industry data sources that indicate ventilation fans are not necessary for containment mixing:

   - Experimental data indicates that spray flow induces significant air currents inside containment NUREG/CR-5966.

   - NUREG/CR-5662, "Hydrogen Combustion, Control, and Value-Impact Analysis for PWR dry containments," examined PWR hydrogen issues. Experimental data from three studies were examined specifically the Hanford Engineering Development Laboratory (HEDL) ice condenser simulation, the Nevada Test Site (NTS) large single volume simulation and the German Heissdampfreaktor (HDR) facility. The results of the HEDL tests require some caution as the test vessel, simulating an ice condenser design, included a recirculation fan. However, both the NTS and HDR facilities closely represented a large dry containment design such as PNP. In fact, the German HDR facility has a multi-compartment geometry (72 sub compartments). The HDR tests simulated a large break LOCA with an initial blowdown and a subsequent smaller steam release with a hydrogen component. The results showed after the initial transient buildup of non-condensible gases that the difference in hydrogen concentration in the various compartments was small. The results demonstrated that when steam and hydrogen were injected at high rates, rapid transport and mixing throughout the vessel were observed. Moreover, when sprays were turned on, the subsequent turbulent conditions created uniform hydrogen conditions immediately.

- NUREG/CR-4102, "Air Currents Driven by Sprays in Reactor Containment Buildings," provides considerable indication that spray induced mixing is substantial relative to mixing resulting from fan operation. Adequate mixing can reasonably be assumed due to the operation of sprays, LOCA hydraulic forces and thermal convection currents. Iodine removal due to containment sprays is only credited in the analysis during the time the sprays are operating so that termination of the sprays may affect mixing but mixing no longer impacts spray effectiveness. Note that spray induced mixing enhances elemental iodine wall deposition (NUREG/CR-0009, Section 6.1.9, Page 93). This enhancement has not been included in the calculation of the elemental iodine wall deposition removal coefficient (NUREG/CR-0009, Section 5.1.2 and Section 6.1.9, Tables 10 and 15).

In summary, the tests and analytical investigations demonstrate gas transport behavior in large-scale multi-compartment facilities in the presence of steam under natural convection and sprayed conditions and show excellent mixing.

NRC Request

6. *The PNP LOCA analysis credits a 50% reduction of the emergency core cooling system (ECCS) leakage into the auxiliary building as per current design basis. The staff notes that Table 9.0-1 of the SE supporting license Amendment No. 31, Section A, LOCA, Item 11, states, "Iodine plateout factor due to high-radiation trip of engineered safety feature (ESF) cubicles ventilation system if significant leakage occurred: 2." If the basis for the 50% reduction for plateout is due to the lack of ventilation in the cubicle area, the staff would assume that the ESF leakage would be evaluated using ground level meteorology. Based on an examination of Table 1.8.1-2 of the technical report and Section 6.2.6 of NAI-1149-014 Rev. 3, it appears that the ESF leakage into the auxiliary building is being evaluated as a stack release.*

Please provide additional information on the basis for the 50% reduction of the ECCS leakage into the auxiliary building and the relationship of this credited reduction to the assumed stack release pathway.

ENO Response

6. ESF room ventilation is normally aligned to the stack. ESR room ventilation is isolated on high radiation signals from RIA-1810 and RIA-1811. In the isolated condition, very little leakage is expected to occur. However, since the radiation detectors, closure signal, dampers and ducting are not classified as safety-related and are not single active failure proof, failure of a damper to close resulting in significant leakage must be evaluated. The issue was evaluated in Systematic Evaluation Program (SEP) NUREG-0820, Topics IX-5, "Ventilation Systems," and XV-19, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary – Radiological Consequences," which indicate significant leakage of the ESF room is equated to damper failure and the assumption of an iodine plate-out factor of 2 was accepted.

   Therefore, in design basis radiological analyses, significant leakage must be considered, i.e., the failure of the ESF room isolation dampers to close on high radiation. An iodine plate-out factor of 2 when significant leakage from the ESF rooms occurs has been accepted as PNP design basis, and is therefore used in the AST submittal.

   If significant leakage is not assumed, leakage may be into other areas of the auxiliary building and a plate-out factor of 2 has not been accepted (although a higher plate-out factor and hold-up would be applicable for small leakage conditions). This scenario would be bounded by the significant leakage case considered.

   The physical basis for the plate-out factor can be found in NUREG/CR-0009, Section 6.1.9. The models, theories and experimental data apply in general to the walls and equipment in the ESF rooms as well as containment.

   All stack releases are treated as ground level releases and not elevated releases. Therefore, calculated atmospheric dispersion factors (X/Q) for stack releases versus an ESF room release differ only in the straight line distance between source and receptor. The stack is closer to normal intakes than the ESF room and is therefore conservative for control room doses. Also, ESF room leakage could reasonably credit the tortuous path that leakage would have to take if not exhausted from the stack, making the stack release assumption even more conservative. Offsite dose calculations involve X/Q that are not source location dependent and utilize the minimum distances from containment to the exclusion area boundary and the low population zone.

NRC Request

7. The AST fuel handling accident (FHA) evaluation assumes a minimum water cover depth of 22.5 feet over the damaged fuel and adjusts the allowed iodine decontamination factor accordingly. Numerical Applications, Inc. Calculation Number: NAI-1149-016 Rev. 1, Palisades Design Basis Fuel Handling Accident AST Radiological Analysis, states that, "Per the inputs listed in Reference 5, 22.5 feet of water will be maintained above the damaged fuel; therefore, the decontamination factor for elemental iodine must be adjusted."

7.1 The cited reference, [Email Jeffery Voskuil (Nuclear Management Company) to Jim Harrell (Numerical Applications, Inc.)], dated June 28, 2004, Subject: FHA Inputs], is not readily available to the staff. Please provide additional information describing the basis for the 22.5 feet of water cover used in the FHA analysis.

ENO Response

7.1 GOP-11, "Refueling Operations and Fuel Handling," section 5.4.1, TS LCO 3.9.6, "Refueling Cavity Water Level," and TS LCO 3.7.14, "Spent Fuel Pool (SFP) Water Level," state that the administrative low water level in the reactor cavity and spent fuel pool during refueling is 647' which corresponds to a water level 1' below the spent fuel pool skimmers located at an elevation of 648'. If the inventory decreases below 647', a low water level alarm indicates and core alterations are immediately suspended (TS LCO 3.7.14 for the spent fuel pool and LCO 3.9.6 for the refueling cavity water level). Since the reactor cavity floor is at the 624'6" elevation, it is ensured that approximately 22'6" of water remains above a potentially damaged fuel assembly. For a water height of 23' above the potentially damaged fuel assembly the effective decontamination factor for iodine is 200 (RG 1.183, Appendix B, Section 2). For lower water heights, the method of Burley, "Evaluation of Fission Product Release and Transport," Staff Technical Paper, 1971, (NRC Accession number 8402080322) is utilized to determine the reduction in effective decontamination factor.

The FHA analysis assumes that a fuel handling accident occurs at the location where the lowest water height exists, i.e. the reactor cavity floor. In reality, the limiting fuel handling accident is most likely to occur in the containment tilt pit or above the core where the water height above the potentially failed fuel is above 23 ft. The likelihood of a fuel handling accident occurring on the reactor cavity floor is very small due to the fact that the fuel handling machine clearance above the floor is only a few inches. The impact from a drop from such a small distance would not result in the failure of all rods in an assembly, which is assumed in this analysis. However, this analysis is intended to be bounding for all possible events, and therefore the smaller water height is utilized in the calculation of the iodine decontamination factor.

NRC Request

7.2 Please provide additional information describing the differences in the water cover assumptions for the FHA vs the cask drop accident.

ENO Response

7.2 The fuel handling accident is postulated to occur in the area where the lowest water height exists: over the reactor cavity floor inside containment. Spent fuel casks are not brought into containment so the cask drop event is not postulated inside containment. The cask drop event postulated is that the 96-ton transfer cask falls in the westerly direction, onto the 7x11 Westinghouse spent fuel pool (SFP) fuel rack. It is assumed that 73 peak fuel assemblies with 30 (and 90) days decay are damaged and release their fuel rod gap gas inventories. Note: The 7x11 Westinghouse SFP fuel rack has four locations obstructed by piping associated with spent fuel pool cooling system. No fuel which may be present in the cask is assumed to fail and the drop of the cask into the pool is conservatively not assumed to change the water level.

The elevation of the spent fuel pool floor is 611' 0" and the normal elevation of the water level in the spent fuel pool is 648' 0". GOP-11 specifies that the minimum water level, when handling spent fuel is 1' below the skimmers, which corresponds to the 647' elevation. The NUS (region I) racks have a 10.25" center to center spacing (FSAR, Section 9.11.3.2) and are 149.50" or 12.46' tall. The Westinghouse (region II) racks have a 9.17" center to center spacing and are 151.25" or 12.60' tall. Hence, more than 23' of water exists above the top of the fuel assemblies.

NRC Request

8. RG 1.183, Appendix F, Regulatory Position 5.3, states that, "The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212° F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated." The AST steam generator tube rupture (SGTR) analysis assumes that the release of radioactivity from both the ruptured steam generator (SG) and the unaffected SG continues for 8 hours, until shutdown cooling is in operation, and steam releases from the steam generators have been terminated. The AST main streamline break (MSLB) analysis assumes that both primary-to-secondary leakage and releases from the faulted SG continue for 12 hours at which time the temperature of the leakage is projected to be less than 100°C (212°F) and the faulted SG is completely isolated. Both analyses assume that the release of radioactivity from the unaffected SG continues for 8 hours until shutdown cooling *is in operation and releases from the unaffected steam generator have been terminated.*

Please provide additional information describing the difference in the transport assumptions used in the MSLB and the SGTR accidents regarding the time duration for releases from the affected SG, i.e., the ruptured SG in the SGTR and the faulted SG in the MSLB.

ENO Response

8. In general, FSAR Chapter 14 transient thermal-hydraulic analyses are only run out until the nuclear steam supply system (NSSS) has recovered and stabilized. For MSLB analyses, this is generally after the blowdown is over, SG inventories have stabilized, primary pressure and temperatures recover and any return to power has been suppressed. For PNP, this is from 350 to 700 seconds into the MSLB event (FSAR Figures 14.14-1 through 14.14-22). SGTR analyses are run out until shutdown cooling conditions are reached (FSAR Figure 14.15-12).

   Once entry into shutdown cooling is achieved, releases from unfaulted steam generators cease because the SGs can be effectively isolated and are not needed for steaming. For the MSLB, since the faulted generator cannot be effectively isolated, the release is continued until the primary coolant system temperature is taken below 212°F. SGTR analysis demonstrates that the assumption to reach shutdown cooling in 8 hours is bounding. The time to cool the PCS to less than 212°F is based on an additional cooldown from SDC entry conditions at 300°F to 212°F over 4-hour period, or a cooldown rate of 22°F/hr. Note that maximum allowable cooldown rates based on TS LCO 3.4.3 are 60°F/hr from 300°F to 250°F and 40°F/hr below 250°F, and that operators can be expected to attempt to terminate the leak as quickly as is allowed by TS.

NRC Request

9. In the MSLB, SGTR and control rod ejection (CRE) analyses, the time specified to establish shutdown cooling is 8 hours. In the MSLB, the time specified for the cessation of both primary-to-secondary leakage and releases from the faulted SG is 12 hours, at which time the temperature of the leakage is projected to be less than 100°C (212°F) and the faulted SG is completely isolated.

9.1 Please provide additional information to verify that the 8-hour time period for alignment to residual heat removal (RHR) is based on the time required to reduce the system heat load to the point where the RHR system can remove all the decay heat using only safety grade equipment.

ENO Response

9.1 The assumption of 8 hours to reach shutdown cooling entry conditions bounds actual time to shutdown cooling entry, as indicated by the thermal hydraulic analysis for the SGTR which calculates a conservative time to shutdown cooling as 23,300 seconds (6.5 hours). A longer cooldown results in a greater integrated steam release and is conservative with respect to radiological consequences. The increased stored energy removal for the main steam line break due to the faulted SG blowdown and for the control rod ejection due to the induced loss of coolant accident have shorter cooldown times.

NRC Request

9.2 Please provide additional information to verify that the 12-hour time period for the cessation of both primary-to-secondary leakage and releases from the faulted SG in the MSLB analysis, is based on the time required to reduce the temperature of the primary-to-secondary leakage to less than 100°C (212°F), using only safety grade equipment.

ENO Response

9.2 The shutdown cooling system is safety-related. The additional cooldown of 4 hours from shutdown cooling entry conditions to less than 212°F represents a cooldown rate of less than 25°F/hr. This cooldown rate is much less than the TS allowed cooldown rates of 60°F/hr or 100°F/hr for PCS temperatures between 200°F and 300°F. Operators can be expected to cool the plant down as soon as achievable after a MSLB event has occurred. Therefore, the 12-hour time to 212°F is conservative.

NRC Request

10. In Section 1.6.3 of the AST Technical Report, "Control Room Heating, Ventilation, and Air-Conditioning System Description," the net volume of the CRE is given as 76,451 ft^3. The control room (CR) volume used in the inhalation dose consequence analyses, as shown in Table 1.6.3-1 of the AST Technical Report, is 35,923 ft^3.

10.1 Please provide additional information describing the basis for the use of the CR volume of 35,923 ft^3, as opposed to the CRE volume of 76,451 ft^3 in the CR inhalation dose consequence analyses.

ENO Response

10.1 In Section 1.6.3 of the AST Licensing Technical Report the phrase "control room envelope" refers to the total volume of air serviced by the control room heating, ventilation and cooling (HVAC) system. In Table 1.6.3-1 of the AST Licensing Technical Report the phrase "control room" also refers to the total volume of air. For PNP, the total volume of air serviced by the control room HVAC system consists of the control room proper, the viewing gallery, the technical support center and the mechanical equipment room. See response to RAI 10.2.

A smaller control room envelope volume generally results in higher doses, although dose results are not very sensitive to control room envelope volume.

The control room envelope volume of 35,923 ft^3 is current licensing basis value. It is based on a conservative calculation of the control room envelope volume that ignores the air space above the drop ceiling in the control room area and ignores the volume of the mechanical equipment room. The control room envelope volume of 76,451 ft^3 is a best-estimate value that includes all of the air space within the control room envelope. The best-estimate volume is needed for ASTM E-741 compliant tracer gas testing and the accuracy of the estimate is confirmed during testing.

The statements below regarding compartment doses follow from the equations in RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors" (ADAMS Accession Number ML031490640) and NUREG/CR-6604:

- For given volumetric flow rates for filtered and unfiltered inleakage and recirculation (and exhaust), a larger compartment volume results in greater compartment equilibrium activity, a longer time to reach equilibrium, and the same compartment equilibrium activity concentration as compared to a smaller compartment volume.
- For a larger compartment, the greater compartment activity and longer time to reach equilibrium is due to the fact that the given recirculation and exhaust volumetric flow rates are not as efficient in removing activity from the larger volume. The same compartment activity concentration is due to the fact that the equilibrium activity concentration does not depend on volume.
- For doses that do not utilize a geometric correction factor (i.e., for inhalation and beta submersion doses), a larger compartment volume results in lower doses since the larger time constant results in lower activity concentration at a given point in time, resulting in a lower integrated activity concentration for any finite time period.

- For doses that utilize a geometric correction factor (i.e., for gamma submersion doses), a larger compartment volume has competing effects: the larger time constant which reduces doses and the reduced geometric correction factor which increases doses.

NRC Request

10.2 Please provide additional information defining the boundaries of the CR proper, the CRE, and the CR/technical support center (TSC) envelope if the latter is a separately designated area.

ENO Response

10.2 The control room envelope consists of the CR, the TSC, the viewing gallery and the mechanical equipment room. The viewing gallery includes offices and a bathroom. The mechanical equipment room is divided into two separate equipment compartments and a common duct chase. The CR HVAC system has separate and redundant air handling units, separate and redundant air filtering units, separate and redundant condensing units, separate and redundant steam humidifiers, and separate and redundant continuous air monitors. Each mechanical equipment room equipment compartment has one air handling unit, one air filtering unit, one condensing unit and an electric unit heater. There are two normal outside air intakes, one associated with each of the air handling units. A single common remote emergency outside air intake serves both of the air filtering units. Common supply and return ducts serve the CR, the TSC and the viewing gallery. The control room HVAC system also includes a smoke purge exhaust fan with duct and a toilet exhaust fan with duct.

The control room envelope boundaries include the vestibules on the four CR/TSC/ viewing gallery entrances, the mechanical equipment room doors and the CR HVAC side of all electrical and mechanical penetrations in the CR, TSC, viewing gallery and mechanical equipment room. See RAI 11.2 for additional information.

NRC Request

11. For the LOCA CR habitability analysis, the LOCA analysis assumes an unfiltered in-leakage of 10 cfm after CR isolation. Page 9.8-12 of the FSAR, Section E, Control Room/TSC Envelope, states, "Four vestibules are used to provide egress and ingress to the control room/TSC during post-accident operations. These vestibules are adjacent to Doors 108, 115, 175 and 52. Their function is to prevent air in-leakage."

11.1 Please provide additional information describing the area for which the 10 cfm unfiltered in-leakage restriction is to apply.

ENO Response

11.1 The area (volume) to which the 10 cfm unfiltered in-leakage applies is the entire volume enclosed within the control room envelope boundary. The area is described in RAI 10.2 and indicated on plan view drawings in RAI 11.2.

NRC Request

11.2 Please provide a plan view of the boundary of the area to which the 10 cfm unfiltered in-leakage restriction is to apply. Please indicate all doorways into the area showing that they are equipped with double door vestibules to preclude unfiltered in-leakage from ingress/egress.

ENO Response

11.2 A plan view of the control room envelope boundary, which is the boundary of the area to which the 10 cfm unfiltered in-leakage restriction applies, is shown on drawing M-4, which is provided in Attachment 1. Doors 108, 115, 175 and 52 are located approximately as indicated by the mark-up of drawing M-4, and have associated vestibule doors to preclude unfiltered in-leakage from ingress/egress. Doors 15 and 16, into the mechanical equipment room, are located approximately as indicated by the mark-up of drawing M-4, but do not have vestibule doors. However, following an accident, the mechanical equipment room doors have their security card-readers deactivated to preclude personnel entering the mechanical equipment room without the knowledge of the control room.

NRC Request

12. The term reactor building is used the technical report NAI-1149-027 in the description of the small line break outside containment (SLBOC). FSAR Section 14.23, describing the SLBOC, uses the term auxiliary building.

Please provide clarification of the use of the terms reactor building versus auxiliary building at PNP.

ENO Response

12. The term reactor building is not formally defined for PNP. The use of the term reactor building is a typographical error in Section 2.5 – Small Line Break Outside Containment and Table 2.5-1 of the licensing technical report NAI-1149-027, "AST Licensing Technical Report," revision 1 (ADAMS Accession Number ML062830424), and should be corrected to read auxiliary building. Note the underlying calculation NAI-1149-20, "Palisades Design Basis Small Line Break Outside Containment AST Radiological Analysis," revision 0, correctly uses the term auxiliary building.

ENCLOSURE 1
RAI RESPONSE ON ALTERNATIVE SOURCE TERM LAR
PALISADES NUCLEAR PLANT

Requests for additional information received by electronic mail March 29, 2007.

NRC Request

1. *ARCON96 was used to calculate new atmospheric dispersion factors (X/Q values) for use in evaluating the radiological consequences of design-basis accidents on the control room. Hourly onsite meteorological data (e.g., wind speed and direction, temperature) that were collected during the calendar years of 1999-2003 at 10.1-m and 57.8-m levels were used as input into the ARCON96 code. PAVAN was used to generate X/Q values at the exclusion area boundary and low population zone using a joint frequency distribution (JFD) of wind direction and wind speed with respect to atmospheric stability class. The JFD was derived from the onsite meteorological data collected at 10.1-m.*

   According to the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.194 (Table A-1), ARCON96 requires wind directions to be represented from 1°-360°, with a wind blowing from the north representing 360°. However, the meteorological data set submitted contains wind direction values ranging from 0°-360°. If the 0° wind direction values are intended to represent valid wind direction observations from the north, they will be misinterpreted by ARCON96 as invalid data. Please explain this apparent deviation from the regulatory guidance.

   The staff notes that the onsite wind directions from 0° <u>always</u> have a corresponding wind speed of 0 m/s. Table 1.1 below illustrates the number of occurrences where the wind direction is from 0° and the wind speed is 0 m/s. Are the meteorological data where wind direction equals 0° and wind speed equal to 0 m/s actually representing calm meteorological conditions or should these data be considered invalid? What is the impact, if any, on both the ARCON96 and PAVAN X/Q values if these data are invalid values and are identified as such with a field of "9"s?

   Table 1.1: Palisades (1999-2003) Number of Occurrences of Wind Direction from 0° and Wind Speed equal to 0 m/s

Palisades (1999-2003) Counts of Wind Direction from 0° and Wind Speed = 0 m/s.		
	10.1-m	*57.8-m*
1999	*19*	*41*
2000	*19*	*23*
2001	*38*	*6*
2002	*15*	*16*
2003	*33*	*11*
Total	*124*	*97*

ENO Response

1. PNP's contracted meteorologist confirmed the validity of the zero wind direction data sets in the 1999-2003 MET data. A wind speed of 0 results in undefined wind direction. Common meteorological practice is to use 0 for wind direction when wind speed is 0. Zeroes in direction represent actual calm conditions and are not invalid data. Invalid data would be marked by padding fields with 9s. Note that wind speeds less than 0.15 m/s will register as 0 due to bearing friction in anemometer. PNP replaces MET tower anemometer bearings every 6 months (every quarter during the 1999-2003 time frame). Hourly averages are average of 15 minute averages, which are averages of 1 minute data collection.

   During processing, ARCON96 source code treats wind direction of 0 as invalid data and the number 9 is entered. However, as indicated in NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes," revision 1 and from the ARCON96 source coding in subroutine XOQCALC5, for calm conditions (wind speeds less than 0.5 m/s as per RG 1.194) the wind direction is not relevant and the low wind speed correction is applied, the X/Q are calculated, and calm conditions count is updated.

   In summary, the zero wind speed MET data is valid and ARCON96 appropriately utilizes the zero wind speed data entries.

NRC Request

2. *ARCON96 was used to calculate new X/Q values for use in evaluating the radiological consequences of design-basis accidents on the control room. Table 1.8.1-1, on page 56 of 84, in Enclosure #4 (NAI Report No. NAI-1149-027 AST Licensing Technical Report for Palisades Revision #1) of the Palisades Alternative Source Term Proposed License Amendment lists release-receptor combination parameters for analysis events. Of particular interest to the staff is the release-receptor combination involving the safety injection and refueling water storage tank vent and the normal control room intake "B" which has a separation distance of approximately 7.7 m. According to the NRC RG 1.194 Regulatory Position C.3.4, if the distance to receptor is less than about 10 m, ARCON96 should not be used to assess relative concentrations. Please explain this deviation from regulatory guidance and justify the use of ARCON96 in your analysis for this release-receptor combination.*

ENO Response

2. RG 1.194, Regulatory Position C.3.4, states that if the distance to the receptor is less than about 10 m, ARCON96 should not be used. However, no alternative method is indicated so no provision for remediation exists.

Technical justification for conservatism of the ARCON96 X/Q values for SIRWT vent to control room normal intakes exists. The table below indicates the conservatism in the ARCON96 results for the SIRWT source and normal intake receptor:

	X/Q		
Time Interval	FSAR Prior to Wind Tunnel Normal Intake [s/m^3]	Wind Tunnel SIRWT To Normal Intake [s/m^3]	ARCON96 SIRWT To Normal Intake [s/m^3]
0 – 8 hrs	3.85×10^{-3}	1.32×10^{-2}	9.57×10^{-2} (0 – 2 hrs) 7.59×10^{-2} (2 – 8 hrs)
8 – 24 hrs	3.50×10^{-3}	7.78×10^{-3}	2.87×10^{-2}
1 – 4 days	2.85×10^{-3}	4.95×10^{-3}	2.19×10^{-2}
4 – 30 days	2.15×10^{-3}	2.18×10^{-3}	1.65×10^{-2}

The table indicates the ARCON96 values are more conservative than the previous Palisades Final Safety Analysis Report values.

NRC Request

3. *Table 1.8.1-3, on page 59 of 84, in Enclosure #4 of the Palisades Alternative Source Term Proposed License Amendment shows the release-receptor point pairs assumed for analysis events (e.g., loss-of-coolant accident (LOCA), main steamline break (MSLB), steam generator tube rupture (SGTR), small line break outside containment (SLBOC), control rod ejection (CRE), fuel handling accident (FHA), and spent fuel cask drop) prior to and following control room isolation. For the small line break outside containment event listed in Table 1.8.1-3, the release-receptor point pairs prior to and following control isolation appear to be reversed. More specifically, should the release-receptor pair prior to control*

room isolation be E (plant stack-normal intake "B") and the release-receptor pair following control isolation be F (plant stack-emergency intake)?

ENO Response

3. Yes, release receptor pairs should be reversed for Table 1.8.1-3 of NAI-1149-027, revision 1. The correct X/Q values were used in calculation NAI-1149-020, revision 0.

NRC Request

4. *The staff finds the licensee's description of the Radiological Consequences – Event Analyses in Section 2.0, beginning on page 16 of 84, in Enclosure #4 (National [sic] Applications, Inc. (NAI) Report No. NAI-1149-027 AST Licensing Technical Report for Palisades Revision #1) of the Palisades Alternative Source Term Proposed License Amendment to be very thorough and adequate. However, the staff had difficulty interpreting the summary of the cask drop event description presented in Table 1.8.1-3 Release-Receptor Point Pairs Assumed for Analysis Events. The cask drop event is characterized by multiple cases which do not necessarily correspond to the format of Table 1.8.1-3, with regard to the other event descriptions (i.e., LOCA, MSLB, SGTR, SLBOC, CRE, and FHA). Therefore, to clarify the description of the cask drop event, please provide a separate table which completely describes the event.*

ENO Response

4. For cases 1 and 2, emergency mode of the CR HVAC system is the initial condition of the event since heavy load procedures require CR HVAC to be in emergency mode. For case 3, emergency mode of the CR HVAC is not credited at any time during the event. Therefore, there is no "prior to" control room isolation for cases 1 & 2, and no "following" control room isolation for case 3.

Spent Fuel Cask Drop Release – Receptor Pairs	Case 1	Case 2	Case 3
Filtered Release – Unfiltered Makeup and Inleakage	E	E	n/a
Unfiltered Release – Unfiltered Makeup and Inleakage	L	L	L
Filtered Release – Filtered Makeup	F	F	n/a
Unfiltered Release – Filtered Makeup	M	M	n/a

NRC Request

5. *The plant drawings that were submitted as part of the Palisades Alternative Source Term License Amendment Request are somewhat hard to decipher. In order to perform and adequate review of the release-receptor data provided as input to ARCON96, please provide a plant drawing, to scale, with all the release and receptor locations clearly labeled.*

ENO Response

5. Figure 1.8.1-1 of NAI-1149-027, revision 1, is roughly to scale. Plant drawing C-3 indicates the approximate to scale release-receptor pairs. This drawing is provided in Attachment 1.

ATTACHMENT 1

Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant

RAI RESPONSE ON ALTERNATIVE SOURCE TERM LAR

PLANT DRAWINGS
M-4 and C-3

Provided in response to RAI 11.2 and RAI 5

2 Drawings Follow

THIS PAGE IS AN OVERSIZED DRAWING OR FIGURE,

THAT CAN BE VIEWED AT THE RECORD TITLED: "EQUIP. LOC. - REACTOR & AUX. BLDG. RADWASTE MODIFICATIONS PLAN OF EL 625'-0."

WITHIN THIS PACKAGE... OR BY SEARCHING USING THE DOCUMENT/DRAWING NO. M-4, REV. 24

D-01

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D-02X