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SOUTHERN COMPANY
Energy to Serve Your World℠

February 25, 2008

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50-366

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

Edwin I. Hatch Nuclear Plant
Request to Implement an Alternative Source Term
Response to Request for Additional Information
Regarding the Radiological Consequences Analyses

Ladies and Gentlemen:

On August 29, 2006, Southern Nuclear Operating Company (SNC) submitted a request to revise the Edwin I. Hatch Nuclear Plant (HNP) licensing/design basis with a full scope implementation of an alternative source term (AST). By letters dated November 6, 2006, November 27, 2006, January 30, 2007, June 22, 2007, July 16, 2007, August 13, 2007, October 18, 2007, December 11, 2007, January 24, 2008, and February 4, 2008 SNC has submitted further information to support the NRC review of the HNP AST submittal.

By letter dated May 8, 2007, the NRC requested additional information concerning the AST radiological consequences analyses for the four HNP design basis accidents described in enclosure 1 of the referenced AST submittal. The enclosure to this letter contains the SNC response to the referenced NRC request for additional information (RAI).

The 10 CFR 50.92 evaluation and the justification for the categorical exclusion from performing an environmental assessment that were included in the August 29, 2006 submittal continue to remain valid.

(Affirmation and signature are on the following page.)

Mr. L. M. Stinson states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

This letter contains no NRC commitments. If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

L. M. Stinson
Vice President Fleet Operations Support

Sworn to and subscribed before me this 25th *day of* February, *2008.*

Notary Public

My commission expires: July 5, 2010

LMS/CLT/daj

Enclosure: 1. Response to Request for Additional Information Regarding the Radiological Consequences Analyses

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. D. R. Madison, Vice President – Hatch
Mr. D. H. Jones, Vice President – Engineering
RType: CHA02.004

U. S. Nuclear Regulatory Commission
Mr. V. M. McCree, Acting Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Hatch
Mr. J. A. Hickey, Senior Resident Inspector – Hatch

State of Georgia
Mr. N. Holcomb, Commissioner – Department of Natural Resources

Edwin I. Hatch Nuclear Plant
Request to Implement an Alternative Source Term

Enclosure 1

Response to Request for Additional Information
Regarding the Radiological Consequences Analyses

Enclosure 1

Edwin I. Hatch Nuclear Plant
Request to Implement an Alternative Source Term

Response to Request for Additional Information
Regarding the Radiological Consequences Analyses

LOSS-OF-COOLANT ACCIDENT (LOCA) QUESTIONS

NRC Question 1

Please provide the equations and input parameters used to calculate the activity leak rates from containment and through the main steam isolation valve (MSIV) to the environment.

SNC Response

The equation used to determine the volumetric flow from the containment through the one closed outboard MSIV is based on the fact that for critical flow, the volumetric flow remains constant with respect to changes in upstream pressure as long as the gas composition and the gas temperature remain constant. Therefore, one may calculate the volumetric flow at any test pressure above the critical pressure for a given leak path, and it would be the same under accident conditions except for the change in gas composition (nitrogen to a nitrogen-steam mixture) and temperature (from a nominal test temperature to the maximum accident temperature).

The volumetric flow from the containment under test conditions is simply:

$$Q_{test} = w_{test}/\rho_{test}$$

where w_{test} and ρ_{test} are the tested mass flow rate and the density under test conditions, respectively. As previously noted, if the test is run at a pressure above the critical pressure, then both w and ρ will increase linearly with absolute pressure up to and including the accident pressure (assumed to be greater than the test pressure) as long as the temperature and the composition are constant. So as long as the accident temperature and composition are the same (i.e., only the accident pressure increases and $P_{accident} \geq P_{test} \geq P_{critical}$), then:

$$Q_{accident} = Q_{test} = w_{test}/\rho_{test}$$

Under accident conditions, however, both the temperature and the composition can change, usually in a way that increases volumetric flow out of the containment to a value defined as $Q'_{accident}$. The ratio of the sonic velocity at accident conditions to that at test conditions is very close to the ratio of the source volumetric flow at accident conditions to that at test conditions, and for HNP that ratio is as follows:

$$\frac{v'_{sonic}}{v_{sonic}} = \frac{\sqrt{(k'-1)\frac{c'_p}{M'}T'}}{\sqrt{(k-1)\frac{c_p}{M}T_{\text{standard}}}} = \sqrt{\frac{k'-1}{k-1}\frac{c'_p}{c_p}\frac{M}{M'}\frac{T'}{T}} = \left(\frac{(0.3)(8)(29)(800R)}{(0.4)(7)(18)(530R)}\right)^{0.5} = 1.444$$

where the prime indicates accident conditions, and it is assumed that the tested condition is air (k = 1.4) at 530 R and the accident condition is pure steam (k = 1.3) at 800 R. Then:

$$Q'_{accident} = 1.444Q_{accident} = 1.444Q_{test} = 1.444(w_{test}/\rho_{test}).$$

If w_{test} is measured in scfh, then ρ_{test} normalized by $\rho_{standard}$ is simply ($P_{test}/P_{standard}$) and:

$$Q'_{accident} = 1.444(w_{test}\ \text{scfh})(P_{standard}/P_{test})\ \text{cfh} = 0.0241(w_{test}\ \text{scfh})(P_{standard}/P_{test})\ \text{cfm}.$$

Beyond the outboard MSIV, it is expected that the pressure will be atmospheric (standard), but the temperature may still be that of the normally operating steam line. Here, the volumetric flow rate out of the containment $Q'_{accident}$ is increased by both ($P_{accident}/P_{standard}$) and by ($T_{steamline}/T_{accident}$) to calculate the residence time in the steam line. In the main condenser, it is assumed that both the pressure and the temperature are standard. Hence, the volumetric flow rate $Q'_{accident}$ is increased by ($P_{accident}/P_{standard}$) but decreased by ($T_{standard}/T_{accident}$) to calculate the residence time in the main condenser.

NRC Question 2

Please explain the significance of 144 standard cubic feet per hour (scfh) as it relates to the design basis allowable MSIV leakage. Also, please discuss how this value is calculated.

SNC Response

As noted in the response to NRC Question 1:

$Q'_{accident} = 1.444(w_{test}\ \text{scfh})(P_{standard}/P_{test})\ \text{cfh} = 0.0241(w_{test}\ \text{scfh})(P_{standard}/P_{test})\ \text{cfm}.$

If P_{test} were to be increased to $P_{accident}$, then (since the constants 1.444 and 0.0241 are independent of the test pressure) w_{test} would have to increase by the ratio $P_{accident}/P_{test}$ in order for $Q'_{accident}$ to remain the same (the basis for the dose analysis).

The $P_{accident}$ assumed in the calculation is 61.6 psia. Since the volumetric flow $Q'_{accident}$ is not determined by accident pressure (only by the test pressure), the $P_{accident}$ value is important only from the standpoint of calculating the residence time in the steam line beyond the outboard MSIV. In other words, it has been assumed that the volumetric flow in the steam line beyond the outboard MSIV is equal to $Q'_{accident}$ increased by ($P_{accident}/P_{standard}$) as well as by ($T_{steamline}/T_{accident}$). Therefore, the MSIV leakage must not be allowed to exceed that corresponding to an accident pressure of 61.6 psia.

The two conditions that must hold, then, if P_{test} is to be increased to $P_{accident}$ without changing the dose results of the LOCA calculation, are as follows:

1. The leak rate limit corresponding to $P_{test} = P_{accident}$ cannot be allowed to be greater than that corresponding to $P_{accident}$ = 61.6 psia even if $P_{accident}$ > 61.6 psia. If this condition does not hold, then the revised MSIV leak rate limit will result in a residence time in the steam line shorter than that taken credit for in the calculation.

2. The w_{test} value for $P_{test} = P_{accident}$ = 61.6 psia (defined for convenience as "$w_{test\text{-}max}$") must not exceed the current test limit (w_{test}) times the ratio of $P_{accident}/P_{test}$ = (61.6 psia /42.7 psia) = 1.44; i.e., $w_{test\text{-}max}$ = 1.44(w_{test} scfh) = 1.44(100 scfh) = 144 scfh. As long as the MSIV leak rate limit for $P_{test} \geq$ 61.6 psia is 144 scfh, then the dose results of the LOCA calculation will not be exceeded. If P_{test} = 61.6 psia and the limit is 144 scfh, then the dose results will be identical to the dose results of the LOCA calculation which assumes P_{test} = 42.7 psia and a limit of 100 scfh.

NRC Question 3

Appendix A, Section 6.1, of Regulatory Guide (RG) 1.183 states that "the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Regulatory Position 3)." Regulatory Position 3 presents the source term, in the form of activity release fractions, released into containment. It is understood that this containment source term accounts for phenomena that would serve to inhibit activity release from the vessel, prior to transport through main steam and other bypass piping. The guidance of RG 1.183 further allows for the credit of other containment removal mechanisms (i.e., natural deposition and drywell spray); however, applying these additional removal mechanisms, prior, and in addition, to crediting pipe deposition, can substantially change the containment source term assumed to enter the main steam and other bypass piping, thus rendering the containment source term of Regulatory Position 3 inapplicable. Because the cumulative effect of these removal mechanisms was not explicitly addressed by the containment source term provided in Regulatory Position 3, consideration should be given to the interaction of each removal mechanism with the source term of RG 1.183 when modeling the transport of activity from the drywell through bypass pathways.

Therefore, please provide information to show that the cumulative effect of assuming the release of the containment source term, natural deposition, drywell spray removal, followed by pipe deposition, for the postulated LOCA at HNP, does not compromise the conservative characteristics of the dose analysis.

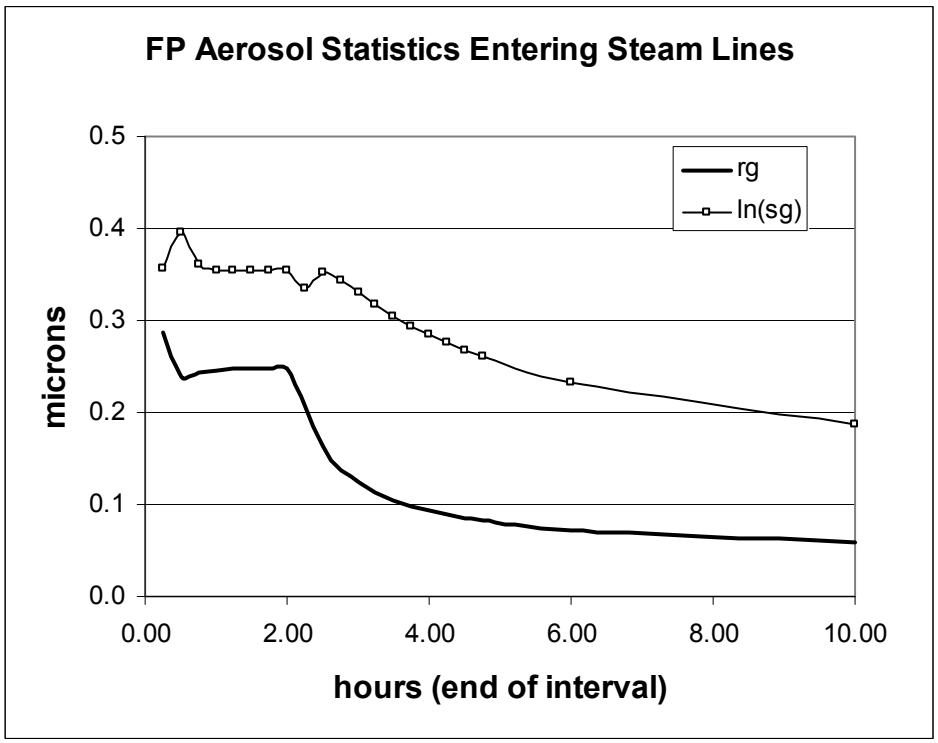
SNC Response

The limiting case for HNP includes failure of an MSIV valve to close (assumed to be an inboard MSIV) in the single line in which the leak rate is assumed to be 100 scfh. The source of the leakage into the MSIV is assumed to be the drywell. One loop of drywell sprays is credited.

The alteration of the containment source term by drywell spray removal has been taken into account when calculating natural removal by sedimentation in the steam line beyond the outboard MSIV. The Polestar computer code STARNAUA has been used to model both drywell and steam line aerosol processes, and indeed, the leakage from the STARNAUA problem representing the containment (with one loop of drywell sprays operating) becomes the STARNAUA source term for the steam line beyond the outboard MSIV with one additional modification. That additional modification is a factor of two reductions due to impaction of the mass entering the space between the closed outboard MSIV and the main condenser.

The post-LOCA size distribution of the aerosol entering the HNP steam line is shown in the following plot (geometric mean radius and natural log of the geometric standard deviation). This plot reflects the impact of drywell spray removal. Once the source ends at two hours, there is a significant decrease in

the aerosol size as the sprays remove the larger particles. There is also a general "tightening" of the size distribution (smaller sigma) during the entire spray removal period. However, there is a large decrease in the overall aerosol mass in the drywell associated with the transfer of mass from the drywell to the torus gas space immediately after the source is stopped (reflecting reflood and cooling of the damaged core). This transfer of mass reduces the effectiveness of the spray on the remaining airborne mass, and this effect is shown in the slight "spreading" of the size distribution (by aerosol agglomeration) as the calculated removal rate decreases.



Because of the rigorous capability of the STARNAUA code to calculate aerosol removal, and as importantly, the impact of that removal on the size distribution and deposition velocity of the aerosol remaining airborne, the cumulative effect of (1) assuming the release of the containment source term, (2) crediting drywell spray removal, and finally, (3) crediting removal by pipe deposition in the steam lines for the postulated LOCA at HNP does not compromise the conservative characteristics of the dose analysis.

NRC Question 4

The phenomena referred to as impaction is generally assumed to take place under conditions characterized by relatively high flow rates and turbulence, where the concentration of airborne particulate, or aerosols, is substantial. For the release model used for HNP, impaction is credited; however, the assumed flow rate is low and potentially laminar, and particles settling in the pipe have been credited as well.

Therefore, please provide the effective decontamination factors associated with the individual aerosol removal mechanisms that were credited for the HNP LOCA analysis, and explain how the credit taken for aerosol impaction accurately accounts for the relatively low flow rate assumed and the settling of large particulate in the pipe length.

SNC Response

Impaction has been credited for the Oyster Creek station alternative source term (AST) application, and the basis for impaction credit for that application (found acceptable by the NRC as documented in a letter from G. Edward Miller (NRC) to Christopher M. Crane (AmerGen) dated April 26, 2007) is essentially identical to that for HNP. In order to avoid a restatement of proprietary material, one may review request for additional information (RAI) responses supporting the Oyster Creek submittal made on February 9, 2001 (AmerGen Letter Number 2130-01-20023), in particular, the response to RAI 8.

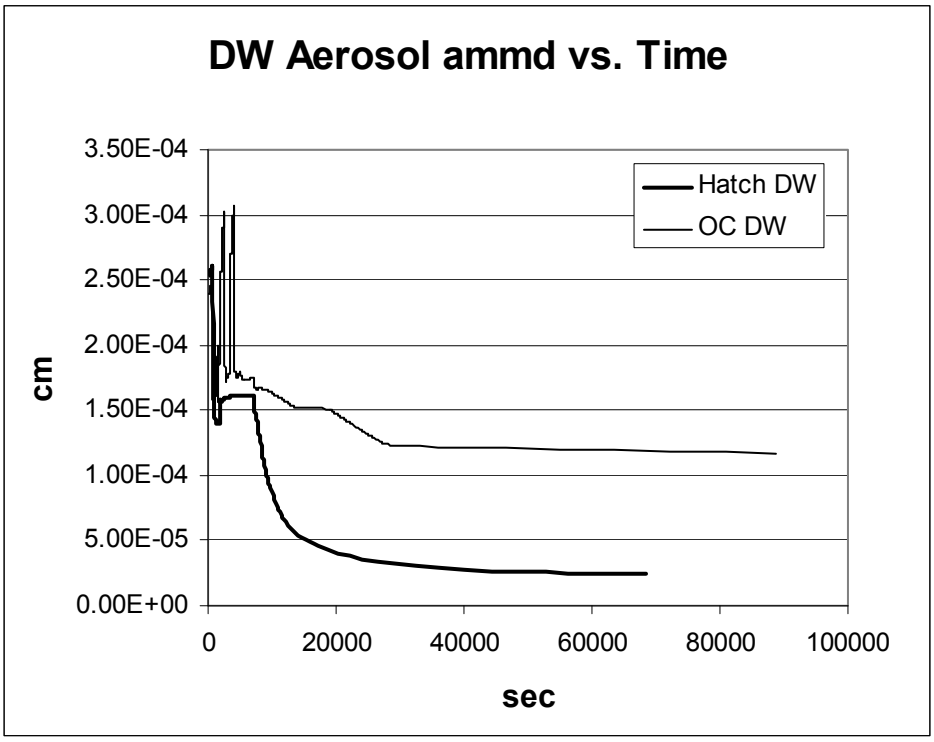
“Impaction” is really two, closely related phenomena. The first is the impingement of aerosol (and its removal from the flow stream) at the entrance to a small passage due to the compaction of streamlines as the flow accelerates to high velocity, even to sonic velocity if the upstream pressure is above the critical pressure. This phenomenon occurs in the immediate vicinity of the entrance and may be viewed as a removal efficiency of aerosol trying to enter the passage. The second is the phenomenon of aerosol plugging of small passages due to accumulated removal (by impaction) at the entrance to and within the passage. This phenomenon may easily produce a DF of 2 with respect to the calculated leakage (integrated over time) even though the first phenomenon, by itself, may be unable to do so (evaluated on an instantaneous basis).

This second phenomenon is covered in the non-Proprietary portion of the February 2001 Oyster Creek RAI response where it was documented that the passage of only a small fraction of the total calculated MSIV aerosol leakage (~2.4% in the Oyster Creek case) would be sufficient to plug the leak path. Any calculated leakage beyond that point would, in fact, not actually occur, so the actual DF for the Oyster Creek MSIV-leakage aerosol release would be 1/0.024 or 42. Only a DF of 2 was actually credited.

A similar situation exists for HNP. The February 2001 Oyster Creek RAI response shows that a leak path becomes plugged when “suspended mass

carried to or past plug" equals KD^3 where K equals 30 ± 20 g/cm^3. This "past plug" phrase is important because it clarifies that the plugging phenomenon is based on the amount of mass that would be leaking through the closed MSIV into the steam lines if there were no such plugging. In this calculation, the leak orifice diameter (as if it were a single hole, which is already a very conservative assumption) is 0.131 cm for the 100 scfh HNP MSIV leakage limit and a test pressure of 28 psig. Using the most conservative value for K, the plugging model would predict plugging to occur when a maximum of 0.11 g has traveled through the orifice, while the HNP aerosol transport analysis shows that a total of 2 g of aerosols would leak out of the drywell. Therefore, the DF that could be credited here would be closer to 20 (i.e., 2 g divided by 0.11 g), rather than just 2. Moreover, the analysis that predicts 2 g of aerosol leaking from the drywell and through the closed MSIV assumes two spray pumps running. Since only one spray pump is actually being credited in the dose analysis, the mass of aerosol being assumed to leak effectively is ~4 g. The actual DF, therefore, would be closer to 40 (approximately the same as that for Oyster Creek) rather than 20 if 0.11 g were sufficient to plug the MSIV leak path.

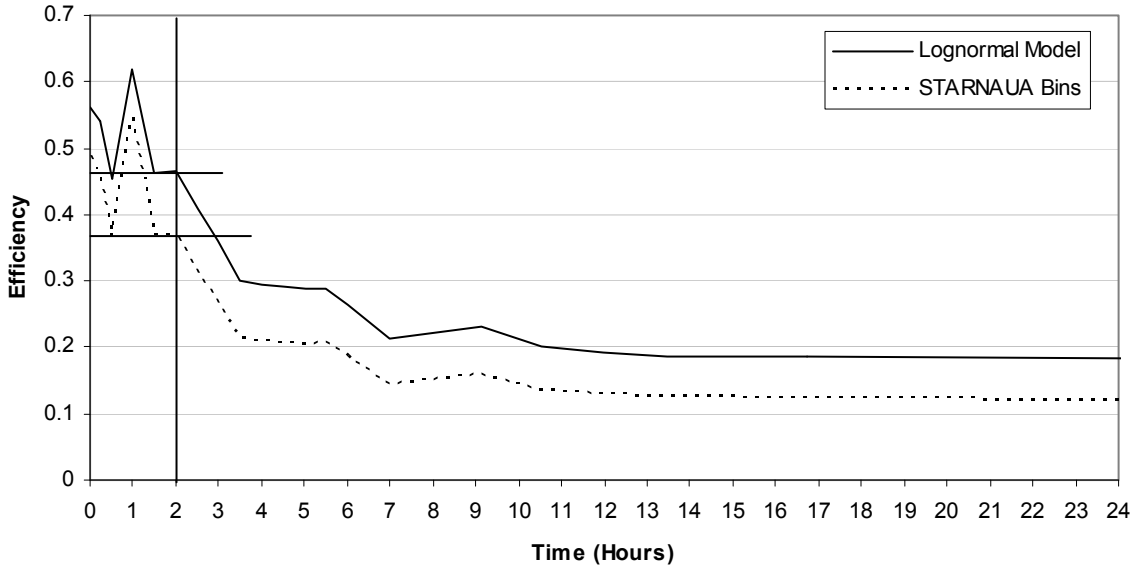
Looking at the first phenomenon (impaction at the entrance as opposed to plugging over time) in the context of the Oyster Creek RAI response, one may note a proprietary model was used to calculate a removal efficiency of ~50% (i.e., a DF of 2) for this phenomenon alone for the Oyster Creek single, closed MSIV. A comparison of the Oyster Creek drywell aerosol size distribution (expressed as the aerodynamic mass mean diameter or ammd) to that of HNP is as follows:



The distributions are initially similar except that the Oyster Creek sprays operate intermittently with the result that ammd "excursions" occur during the release phase at times that sprays are off. At the end of the release phase (t = 7200 seconds), the ammd values are only slightly different (1.75E-4 cm for Oyster Creek and 1.61E-4 cm for HNP). The aerosol size is the key parameter for impaction at the entrance (assuming the flow is accelerating to sonic velocity), so one would expect that impaction efficiency at the end of the release phase would be comparable for the two applications. During the release phase, however, the Oyster Creek application would be expected to exhibit higher removal efficiencies.

The entrance impaction removal efficiency for Oyster Creek at t = 7200 seconds (end of the release phase) is 37-47% and from 37-62% during the release according to the proprietary model previously applied to Oyster Creek (refer to "Figure 4," presented below, from the Proprietary portion of the February 2001 Oyster Creek RAI response). The HNP removal efficiency during the release phase (based on the model applied to Oyster Creek) would be expected, therefore, to be in the range of 37-47% rather than 37-62%. Moreover, after the end of the release phase, the much higher (and continuous) spray removal rate assumed in the HNP aerosol transport analysis brings about an obvious divergence of the aerosol size distribution beyond t = 7200 seconds. Taken at face value, this would mean that one could not claim for HNP what was claimed for Oyster Creek, namely that based solely on entrance impaction (i.e., neglecting any plugging effects), a DF of 2 could be justified.

Figure 4
Collection Efficiency at the Inboard MSIV as a Function of Time



In fact, the proprietary analytical model applied to Oyster Creek (reflected in "Figure 4" above) was quite conservative, and it has been improved upon. It is

now recognized that a DF of 2 (removal efficiency ≥ 50%) can be justified any time that the aerodynamic mass median of the distribution (approximately the same as the ammd late in time) exceeds 3.8E-4√D (where D is the characteristic dimension of the leak path which for HNP will be less than 0.131 cm) as long as the upstream pressure remains above the critical pressure. As noted, the ammd for HNP at the end of the release phase is 1.61E-4 cm, while the critical value based on D = 0.131 cm is 1.38E-4 cm. Drywell pressure may be assumed to remain above critical pressure since the MSIV leak rate is assumed to remain at its maximum value for the first 24 hours. This means that the entrance impaction removal efficiency alone will provide a DF of 2 for HNP for at least the 2-hour duration of the release phase. Of the aerosol that leaks from the drywell, 99% leaks during the 2-hour release phase.

In determining the aerosol removal rate for the steam line, the aerosol mass added to the steam line downstream of the outboard MSIV is reduced by a factor of two (i.e., from 2 g to 1 g) to account for mass removed by impaction and/or plugging. However, the size distribution assumed to exit the MSIV leak path is not modified from that assumed to enter the MSIV leak path. Experimental evidence suggests that even though removal by impaction at the entrance is particle size-dependent (i.e., the critical ammd being approximately 3.8E-4√D cm for sonic velocity), particle agglomeration internal to the pathway actually tends to increase the size of the particles being released by the same mechanisms that ultimately bring about plugging. Therefore, there is no need to reduce the aerosol size distribution leaving the MSIV leak path to account for impaction at the entrance.

Summary

1. The HNP DF for impaction will exceed a value of 2 based solely on entrance effects for at least two hours after the start of the accident even for the case of drywell spray flow (and associated aerosol removal) being twice that credited in the dose analysis. Note that 99% of the aerosol leaked from the drywell occurs during the first two hours.

2. At least two grams of aerosol will be leaked through the outboard MSIV (by analysis) to the steam line if removal by impaction is ignored. This is nearly a factor of 20 greater than the maximum amount necessary to plug the leak path (0.11 g). If the actual drywell spray rate credited in the dose analysis were to be used in the aerosol transport analysis, the amount leaked without impaction/plugging considered would be nearly four grams. This is what was effectively used in the dose analysis, and this value is nearly a factor of 40 greater than the amount needed to plug the MSIV leak path.

3. The Oyster Creek precedent applies to HNP. In both cases, the leakage is directly from the drywell through a single, closed MSIV. In both cases, a DF of two can be justified solely on the basis of impaction effects at the entrance (plugging ignored). In both cases, the amount of aerosol calculated to be leaked without impaction or plugging considered is approximately a factor of

40 greater than that needed to plug the leak path. In both cases, the DF credit has been limited to a factor of 2 for the combined effects of impaction effects at the entrance and ultimate plugging of the leak path.

NRC Question 5

Please verify that all assumed secondary containment bypass pathways enter the condenser under the condenser tubes. If they do not, please justify the applicability of the condenser activity removal model, and activity removal credit associated with this release pathway.

SNC Response

The assumed secondary containment bypass pathways enter the condenser below the top of the tubes and the condenser model in Appendix C Section 7.0 of GE Topical Report NEDC-31858P-A is applicable as credited in the AST submittal.

Specifically for HNP, High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) system bypass leakage enters the condenser two feet below the center line of the tubes, the Reactor Water Cleanup (RWCU) system bypass leakage enters the condenser approximately two feet below the bottom of the tubes, and the MSIV alternate leakage treatment path inlet is six feet below the center line of the tubes and approximately 1'-8 ½" above the bottom of the tubes. All inlets enter the condenser near horizontal cooling surfaces well below the top of the tubes.

The relative inlet locations are shown on Figure 1.

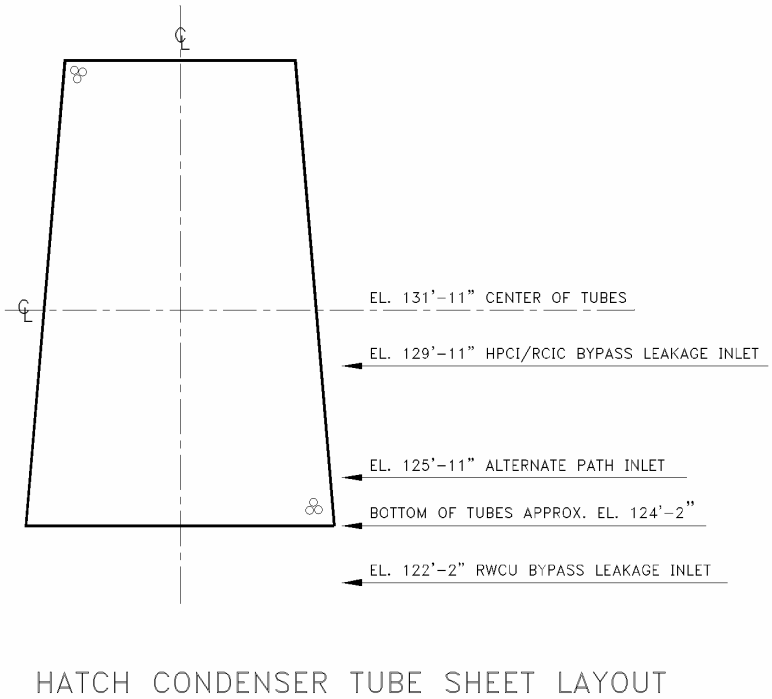

EL. 131'-11" CENTER OF TUBES
EL. 129'-11" HPCI/RCIC BYPASS LEAKAGE INLET
EL. 125'-11" ALTERNATE PATH INLET
BOTTOM OF TUBES APPROX. EL. 124'-2"
EL. 122'-2" RWCU BYPASS LEAKAGE INLET
HATCH CONDENSER TUBE SHEET LAYOUT

FIGURE 1

NRC Question 6

It is stated in Section 2.5.2.1 of the August 29, 2006, submittal that the reactor building (RB) draws down to negative pressure within 2 minutes of the "start of the accident." Please clarify whether the "start of the accident" indicates the start of gap release in this context.

SNC Response

AST submittal Enclosure 1, Section 2.5.2.1 provides the inputs and assumptions for the LOCA radiological consequences analysis. Implicit in these assumptions is that the start of the LOCA, specifically time = 0, is defined as the sudden circumferential severance of one recirculation line resulting in the release of reactor coolant to the primary containment. In accordance with RG 1.183 Table 4 guidance and as stated in the referenced Section 2.5.2.1, the referenced gap release is assumed to start 2 minutes after the start of the accident, specifically time = 2 minutes.

The initiation of the standby gas treatment system (SGTS) and secondary containment (reactor building) isolation will occur within seconds of the break occurring. In response to the rapid vessel depressurization and inventory loss through the break, one or both of the following signals, specifically low reactor vessel water level 2 or high drywell pressure, will initiate SGTS and the secondary containment isolation. SGTS draws down the secondary containment to the required negative pressure within 2 minutes as routinely verified by performance of Technical Specification surveillances. Consequently, the LOCA analysis assumes that with the beginning of the gap release, at time = 2 minutes, the release point will be through the main stack.

NRC Question 7

Please verify that, for the technical support center dose model, the RB vent is the most limiting release point for RB leakage that bypasses the standby gas treatment system.

SNC Response

In evaluating releases from the reactor building (RB) and the turbine building (TB), the following potential ground-level release points were identified for each unit: RB vent, TB railroad door, and TB supply louvers. Of these release points, the Unit 2 RB vent yields the highest atmospheric dispersion factors for both the control room and the technical support center (TSC).

NRC Question 8

Please provide the "rigorous analysis," referenced in the August 29, 2006, submittal, Enclosure 1, Appendix A, which led to the inclusion of an additional 0.5 correction factor being applied to the semi-infinite cloud dose conversion formula referenced in Section 4.2.7 of RG 1.183; and please discuss how this additional 0.5 correction factor was applied to the calculation of post-accident main control room (MCR) dose consequences.

SNC Response

MCR doses are calculated with the LocaDose computer program, which uses the formula specified in Section 4.2.7 of RG 1.183 to estimate the finite dose within a volume. This formula, from a paper presented by Murphy and Campe at the 13th AEC Air Cleaning Conference, is based on the average gamma energy of noble gases. Since the AST includes particulates, the average energy is different from that of noble gases only. To take into account the isotopic mixture specific to AST and to explicitly model the geometry of the HNP MCR, the finite cloud correction factor is rigorously calculated using Shield-SG, a point-kernel computer program that facilitates three-dimensional modeling of bodies.

Using energy- and time-dependent airborne source terms within the control room, Shield-SG is used to calculate the integrated dose over 30 days in the center of the HNP MCR. A second Shield-SG model is then used to calculate the 30-day dose in the center of an infinitely large volume. The ratio of the two doses yields a correction factor of 0.019. The formula from Section 4.2.7 of RG 1.183 yields a correction factor of 0.041. The ratio of the two correction factors is approximately 0.5. This factor of 0.5 is applied only to the deep dose equivalent (DDE) component of the total effective dose equivalent (TEDE) that is due to external exposure.

NRC Question 9

Please verify that all other potential contributors to post-LOCA MCR direct shine dose were evaluated and found to be negligible, including possible nearby core coolant carrying lines.

SNC Response

As covered in Section 2.5.2.3 of Enclosure 1 to the AST submittal, the post-LOCA MCR dose analyses included MCR internal TEDE and ingress/egress TEDE and the following post-LOCA MCR direct shine components, listed in more detail than in the referenced section: turbine building airborne, cloud outside turbine building, main steam lines and main condenser, main control room environmental control system filter, secondary containment airborne, and secondary containment door streaming contributions.

The other potential contributors to post-LOCA MCR direct shine dose include the turbine building ventilation filter, standby gas treatment system filter, and emergency core cooling system lines outside the containment. The MCR is shielded from the referenced filters by at least 4.5 feet of concrete. The MCR is shielded from the referenced core coolant lines outside containment by at least 6 feet of concrete. Therefore, these additional contributors are negligible.

NRC Question 10

Please provide a detailed sketch showing the specific geometries used to model the shine dose to the MCR from external sources, including the airborne activity in the Turbine Building (TB), RB cloud, condenser, MCR filters, TB heating, ventilation, and air conditioning filters, and external plume. Also, please provide the parameters that thoroughly describe source and receiver characteristics (i.e., activity, density, composition, etc…) used to calculate the shine dose from the aforementioned contributors.

SNC Response

Source terms and models for MCR internal TEDE, ingress/egress, and TB internal cloud are as described in AST submittal Enclosure 1 and as clarified in the response to NRC Questions 7, 8, and 11.

TB external cloud (external plume) shine contribution is developed from the unshielded outside cloud at the TSC, which uses the MCR X/Q value, as calculated by the LocaDose computer program. This value is then reduced assuming a factor of 10 per foot (ft) of concrete based on the shielding resulting from 2 ft concrete MCR walls and 6 inch TB walls.

Geometries used for the following shine evaluations are shown in Figures 2 - 4 below.

The main steam line (MSL) was modeled as an air-filled 2 ft diameter pipe with a 1.2 inch steel wall shielded from the control room by 6.1 ft of concrete as shown in Figure 2, with MSL source terms calculated as a function of time by the LocaDose computer program.

The condenser inventory as a function of time is also calculated by the LocaDose computer program. The condenser inventory is compared to the MSL activities (4.22 times higher at 8 hr, 15.4 at 24 hr, 44 at 96 hr, and 264 at 720 hr). The dose from the condenser is taken as the activity ratio times the MSL dose with no credit for the further distance from the MCR or additional slant path through concrete shielding.

RB shine was modeled as a rectangular slab shielded by 4 ft of concrete (2 ft MCR wall and >2 ft RB walls, ignoring the TB and RB internal structures) as shown in Figure 3. Source terms are from an older core inventory which displays a different isotopic mix than that modeled in LocaDose, but bounds the LocaDose activity.

Streaming through the RB door was modeled as a cylinder 36 ft long, 5.17 ft equivalent diameter with the same activities as above for the RB and a 2 ft concrete shield (MCR wall) ignoring RB internal structures, e.g. the 1 ft concrete door shield and width of the HVAC room in the RB shown in Figure 4.

The MCREC recirculation flow is about 2100 cfm / 9.35 x 10^4 ft^3 = 0.022/min. Then assuming all the activity has the half-life of I^{131}, the accumulation on the filter would be

$$A = S / \lambda = 0.022 / (.693/8.05/1440) \approx 370 \text{ times the MCR activity.}$$

A review of the LocaDose output activities in the MCR indicates the noble gas activity is more than 600 times the iodine activity, *i.e.* greater than the filter accumulation, from 8 hours to 720 hours, so the MCR whole body dose is a reasonable surrogate for the unshielded dose from the accumulation on the filter. Therefore, the HVAC filter is estimated as equal to the MCR whole body dose shielded by the 2 ft 6 inch MCR roof, assuming a 1/10th thickness of 12 inches for concrete.

FIGURE 2 – MSL to CR Elevation

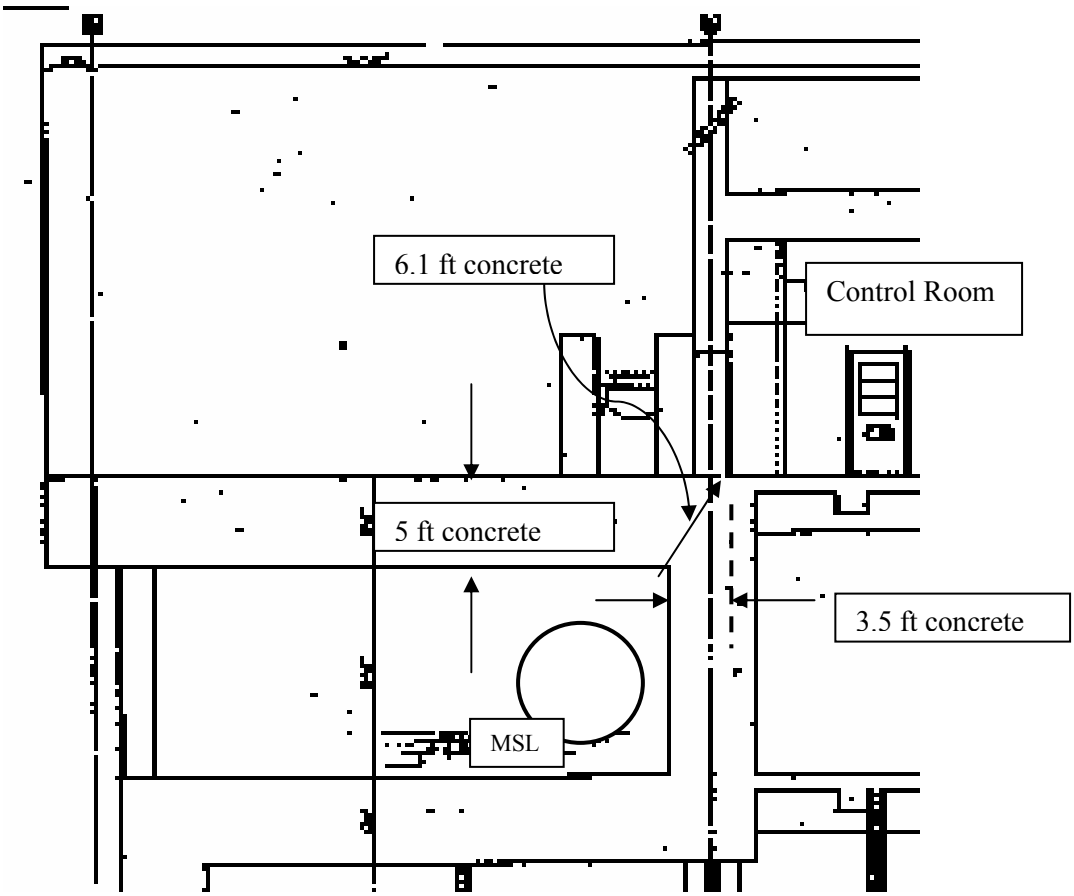
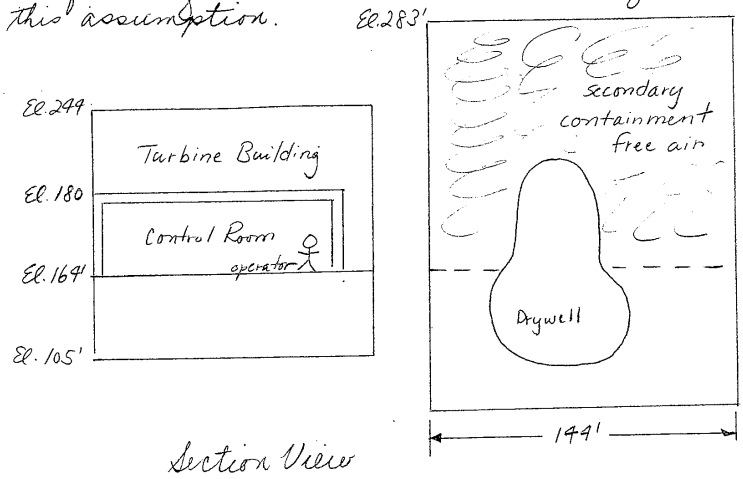
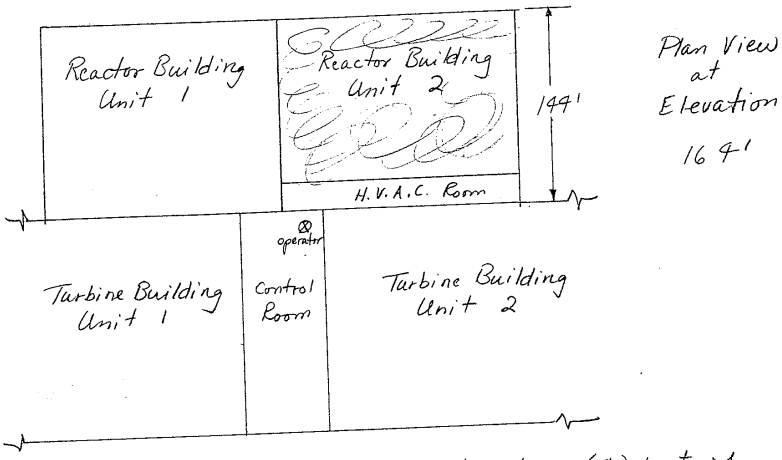


FIGURE 3 – RB air model

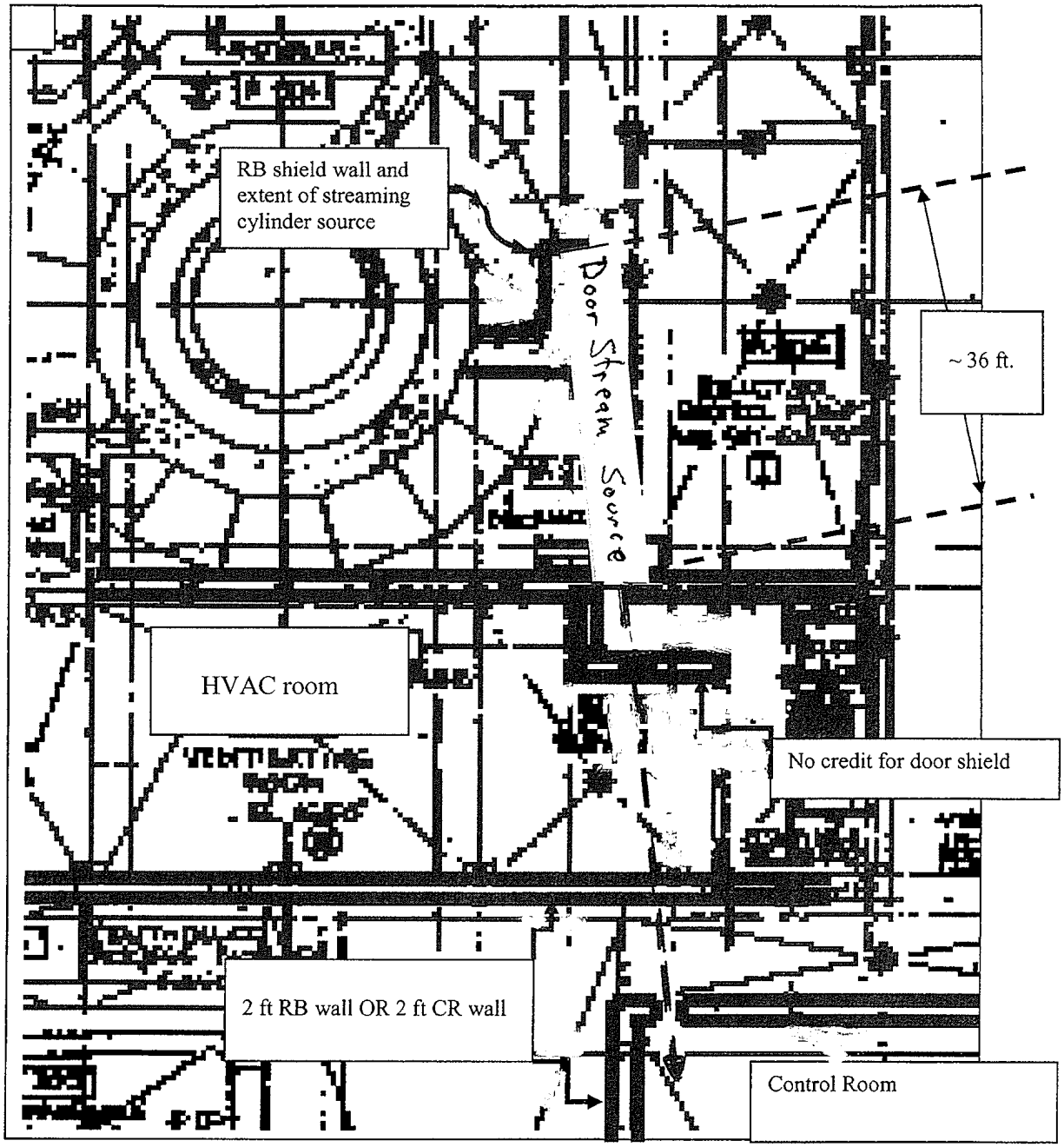
(3) The reactor containment building will be approximated as 1/4 of an infinite slab source and the doses will be calculated from an infinite slab and divided by four. The following sketch shows the reasoning behind this assumption.





(4) Credit will be taken for four (4) feet of ordinary concrete between the operator and the secondary containment free air volume. It can be seen upon inspection that more concrete actually exists. (This assumption via R.C. Boles, Jr.) No credit will be taken for any other shadow shielding or attenuation other than air self attenuation.

FIGURE 4 – RB Door Streaming Model Plan View



NRC Question 11

Please provide all parameters, justification for all assumptions, and methodology, used to calculate the MCR ingress/egress dose described for this postulated accident.

SNC Response

It is assumed that the MCR operator takes two trips through the TB each day for the accident duration of 30 days. Using a walking speed of 3 mi/hr and a travel distance of 321 ft and adding a margin of 45 sec, the duration of each trip is estimated to be 2 min. The LocaDose computer program is used to calculate the dose rate in the TB as a function of time. The ingress/egress dose is calculated by multiplying the time-dependent TB dose rate during each trip by the exposure time of 2 min. This methodology yields a total dose of 0.57 rem TEDE for 60 trips over 30 days.

NRC Question 12

Under the assumed conditions, the activity leakage through the steam lines, to the condenser, and into the TB will take place at a relatively low flowrate (<3.0 cfm). Additionally, there is no credited safety, or non-safety grade, recirculation system in the TB volume. Therefore, this release scenario does not make it intuitively obvious that any thorough, flow-based, mixing will take place in such a way that the released activity becomes "uniformly mixed in the volume of the TB."

So, please provide the justification for assuming that activity leaked into the TB will be uniformly mixed in the 6.5E6 ft3 volume above the 164 ft elevation. Also, please provide justification for the assumed timing of any credited mixing in the TB volume.

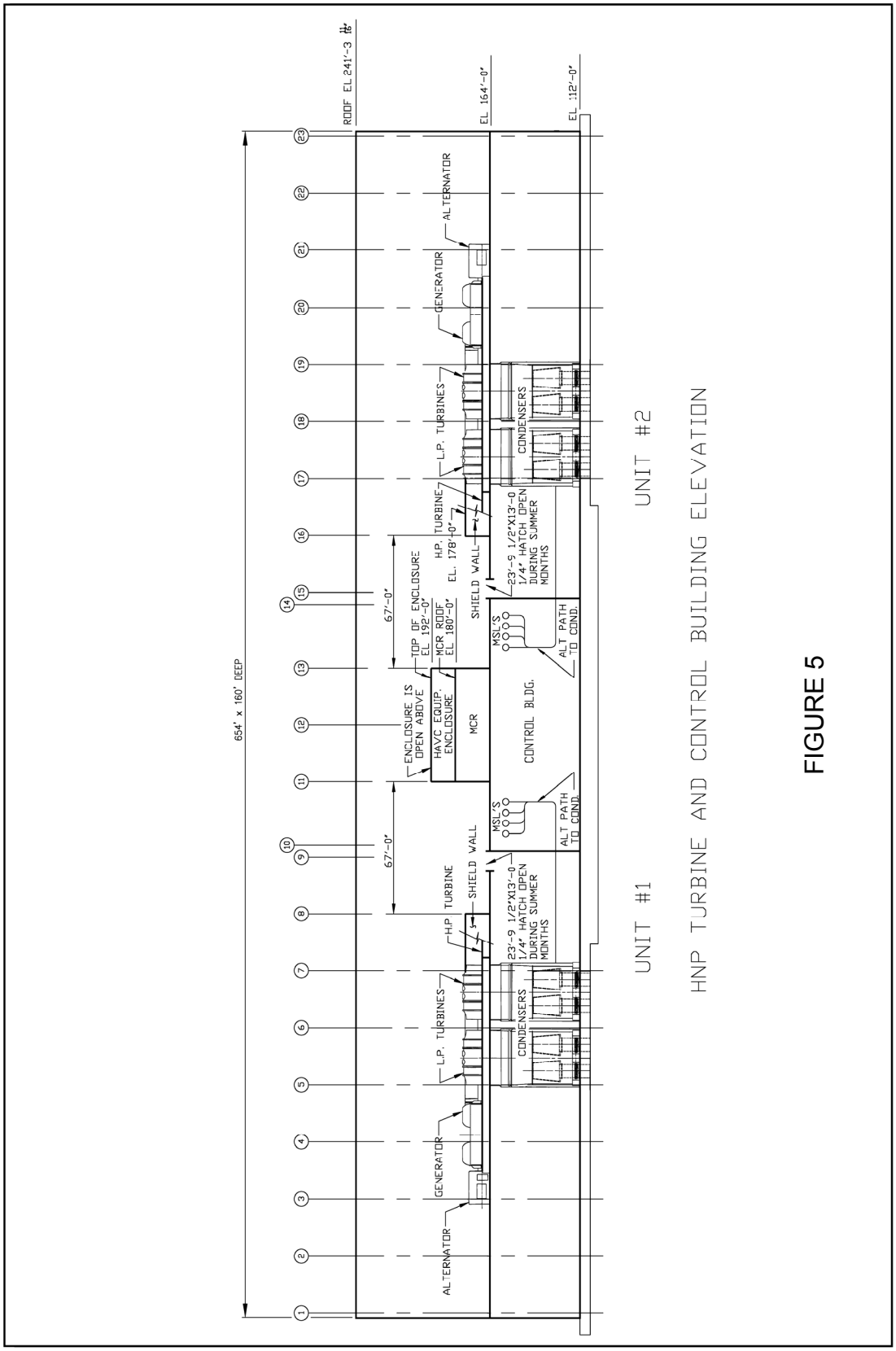
SNC Response

The HNP MCR is the top elevation of the Control Building, at 164'. It is the same elevation as the operating floor of the turbine building, which is open to both units. The relative location of structures and components is shown in Figure 5. All steam line piping, the MSIV alternate leakage treatment pathway piping, the bypass piping, and the main condenser are located below elevation 164'. The top of the condenser is open to the turbine, which is an enclosed piece of equipment. The turbine is completely surrounded by a 14' high shield wall, the base of which is elevation 164'. Although most releases would originate below elevation 164', no credit was taken for any leakage into the volume below elevation 164'. The volume assumed available for mixing is limited to the volume above elevation 164' and below the TB roof.

The shield wall at the end of each high pressure turbine is approximately 67' from the MCR. As can be seen in Figure 5, migration of leakage toward the MCR would be from approximately 40% of the volume above the 164' elevation on the accident unit to the approximately 60% of the volume containing the MCR and the non-accident unit. The majority of leakage being below elevation 164', not crediting the volume below elevation 164', and the distance from the turbine enclosure shield wall to the MCR will all contribute to inhibiting transport of the leakage to the MCR. Thus, the leakage would have to disperse through the TB and spread toward the MCR. Other than the MCR and the turbine shield walls, there are no major obstructions or discontinuities to diffusion or air flow on the operating floor of the TB. Thus, transport from the TB toward the MCR would continue to the remaining >60% volume of the turbine building, assuming no mixing in the opposite direction.

Based on the dimensions shown in Figure 5, the free volume of the TB above elevation 164' is approximately 8.1E6 ft3. The LOCA radiological consequence analysis uses a more conservative TB free volume input of 6.5E6 ft3. The LOCA analysis also assumes uniform mixing within the volume of the TB occurs instantaneously without holdup or decay during transport to the TB volume containing the MCR. Thus, based on the physical configuration, the

conservatisms in the LOCA analysis, and the information discussed in AST submittal Enclosure 1, SNC believes that uniform mixing is a conservative assumption, whether or not forced ventilation is present to promote mixing.



HNP TURBINE AND CONTROL BUILDING ELEVATION

FIGURE 5

FUEL-HANDLING ACCIDENT QUESTION

NRC QUESTION 1

Please verify that HNP has no fuel, and does not intend to use fuel, that exceeds the burnup parameters specified in Footnote 11 of RG 1.183.

SNC RESPONSE

Appropriate measures will be implemented to assure that HNP will operate in compliance with the fuel burnup parameters delineated in Footnote 11 to Table 3 of RG 1.183. These measures will be applied to GE14 fuel and future fuel designs used in the HNP Units 1 and 2 cores as part of AST implementation by May 31, 2010.

Upon AST implementation, fuel previously discharged from either unit for at least one month is bounded by the source term used in the AST radiological analyses irrespective of burnup parameters.

CONTROL ROD DROP ACCIDENT (CRDA) QUESTIONS

NRC Question 1

Please verify that there are no other forced flow activity release paths at HNP, in addition to the mechanical vacuum pump that would contribute to a post-accident dose (i.e., steam jet air ejectors, etc…).

SNC Response

During normal operation, the steam jet air ejectors (SJAEs) remove air and noncondensable gases from the main condenser. These gases are processed through the off-gas system and then exhausted out the plant stack. In response to a CRDA and subsequent reactor scram, operation of the SJAEs is terminated by terminating steam to the SJAEs. Therefore, the SJAEs would not contribute to a post-accident dose.

To demonstrate that forced flow activity release paths would not result in doses that exceeded regulatory limits, the CRDA analysis includes a case of the mechanical vacuum pump running for a period of 24 hours following a CRDA (See AST submittal Enclosure 1, Section 2.5.4). The dose results of this analysis are given in AST submittal Enclosure 1, Tables 25 and 27. These doses are bounding (i.e., higher) for the doses that would result from the SJAEs, even if the SJAEs were to remain in operation for 24 hours following a CRDA, based on the following:

1. The effluent from the SJAEs travels out the plant stack, and is therefore treated as an elevated release. Therefore, the same atmospheric dispersion factors would apply to both the mechanical vacuum pump release and the release from the SJAEs.
2. The volumetric flow rate, and therefore the activity release rate from the condenser into the atmosphere, for the SJAEs is significantly less than the volumetric flow rate for the mechanical vacuum pump.
3. The release from the mechanical vacuum pump is an untreated release. The release from the SJAEs passes through a charcoal adsorber and high-efficiency particulate filter prior to entering the plant stack.

Forced flow activity release paths were considered only for offsite and TSC doses. Any forced flow from the condenser would exit through the plant stack, resulting in less activity in the main condenser available for release to the turbine building. Because the source of MCR dose is inleakage from the turbine building, this would result in lower MCR doses. Therefore, forced flow activity release paths were not considered for the MCR.

NRC Question 2

Please provide all parameters, justification for all assumptions, and methodology, used to calculate the MCR ingress/egress dose described for this postulated accident.

SNC Response

The response for LOCA Item 11 is also applicable for CRDA.

NRC Question 3

Under the assumed conditions, the activity leakage into the TB will take place at a relatively low flowrate (<2.0 cfm). Additionally, there is no credited safety, or non-safety grade, recirculation system in the TB volume. Therefore, this release scenario does not make it intuitively obvious that any thorough, flow-based, mixing will take place in such a way that the released activity becomes "uniformly mixed in the volume of the TB."

So, please provide a discussion of the justification for assuming that activity leaked into the TB will be uniformly mixed in the 6.5E6 ft3 volume above the 164 ft elevation. Also, please provide justification for the assumed timing of any credited mixing in the stated TB volume.

SNC Response

The response for LOCA Item 12 is also applicable for CRDA.

MAIN STEAMLINE BREAK (MSLB) ACCIDENT QUESTIONS

NRC Question 1

Please provide the basis for assuming a mixture quality of 7% for the post-accident coolant blowdown.

SNC Response

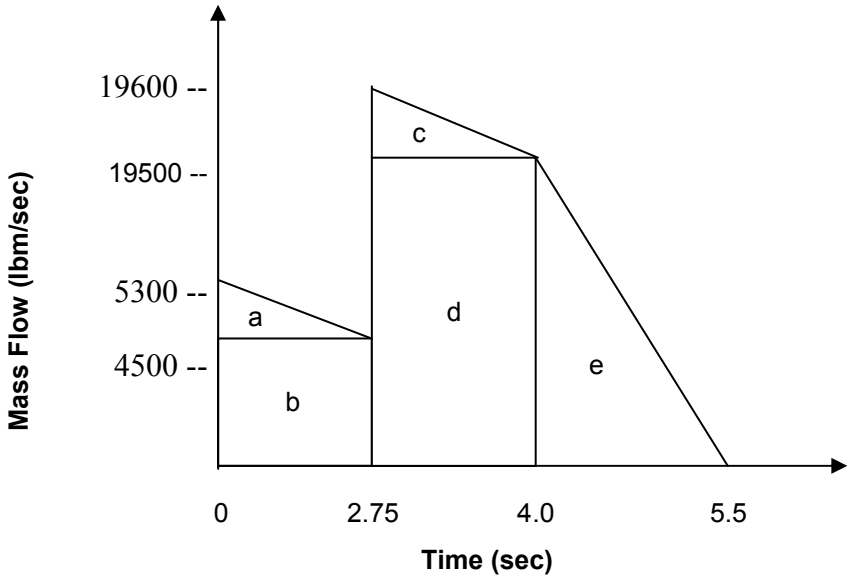
The mixture quality of 7% is obtained from the FSAR (Unit 1 Table N.5-1, Unit 2 Table 15A-2) and is consistent with the current licensing basis.

NRC Question 2

Enclosure 1, Table 30, of the August 29, 2006, submittal indicates that the total mass of the post-accident steam blowdown is 68,174 lbm. Please clarify what fraction of the total coolant blowdown mass forms the post-accident plume of activity.

SNC Response

In Table 30, the fluid mass shown as being released at an enthalpy of 1191.5 Btu/lbm within the first 2.75 sec of the break is steam. The subsequent release to 5.5 sec is a mixture of steam and liquid. The following figure shows the mass flow rate as a function of time:



The sections of the figure are integrated as follows to obtain the total mass released:

Section	Mass Release Formula	Mass Release (lbm)	Fluid Phase	Release by Phase (lbm)
a	(0.5)(800 lbm/sec)(2.75 sec)	**1.10E+03**	Steam	**1.35E+04**
b	(4500 lbm/sec)(2.75 sec)	**1.24E+04**		
c	(0.5)(100 lbm/sec)(1.25 sec)	**6.25E+01**	Mixture	**3.91E+04**
d	(19500 lbm/sec)(1.25 sec)	**2.44E+04**		
e	(0.5)(19500 lbm/sec)(1.5 sec)	**1.46E+04**		
Total		**5.25E+04**		**5.25E+04**

Assuming 7% of the mixture mass of 3.91E4 lbm is steam and adding this to the direct steam release yields a total steam release of 1.62E4 lbm. For the remaining 93% of the mixture that is in liquid form, a flashing fraction of 0.42 is calculated assuming a constant enthalpy process at 14.7 psia and 212 °F, yielding a flashed steam mass of 1.54E4 lbm. The total mass of steam that is released is 3.16E4 lbm, thereby releasing all the activity contained within this mass.

NRC Question 3

Please verify that assuming dilution of post-accident releases in the TB bounds an assumption that the activity plume is released directly to the environment and available for intake to the MCR.

SNC Response

As noted in Section 2.5.5.2 of Enclosure 1 of the AST submittal, the dose in the TSC is based on the assumption that the activity plume is released directly to the environment and is available for intake into the TSC. Based on this release model, Table 33 of Enclosure 1 shows a TSC dose of 0.43 rem TEDE. Tables 4 and 6 of Enclosure 1 indicate that the TSC and the MCR have the same atmospheric dispersion factors but the TSC has higher filtered intake, much higher unfiltered inleakage, and lower recirculation rate and filter efficiency. If the TSC release model were used for the MCR, the dose would be lower than 0.43 rem TEDE because the parameters for the MCR are bounded by the TSC. With the assumption of dilution and holdup in the TB, the dose due to airborne activity within the MCR is calculated as 3.7 rem TEDE, as indicated in Table 32 of Enclosure 1. This dose is nearly ten times higher than the TSC dose, confirming that the TB dilution model bounds the direct release model.

NRC Question 4

Please provide all parameters, justification for all assumptions, and methodology, used to calculate the MCR ingress/egress dose described for this postulated accident.

SNC Response

The response to LOCA Item 11 is also applicable for MSLB.