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Fred Dacimo
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March 14, 2005

Re: Indian Point Unit 3
Docket No. 50-286
NL-05-036

Document Control Desk
U.S. Nuclear Regulatory Commission
Mail Stop O-P1-17
Washington, DC 20555-0001

Subject: **Additional Information Regarding Indian Point 3 License Amendment Request Alternate Source Term (TAC MC3551)**

References: 1. Entergy letter NL-04-068 to NRC, dated June 2, 2004; regarding Alternate Source Term license amendment request.

Dear Sir:

Entergy Nuclear Operations, Inc (Entergy) is submitting additional information to support NRC review of the alternate source term (AST) license amendment request (Reference 1) for Indian Point 3 (IP3). This letter address questions identified in recent telephone conferences with NRC staff. The response to questions is provided in Attachment 1 and revised pages for the AST Licensing Report are provided in Attachment 2.

The additional supporting information, provided in Attachments 1 and 2, does not alter the conclusions of the no significant hazards evaluation that support the AST license amendment request. There are no new commitments identified in this submittal. If you have any questions or require additional information, please contact Mr. Kevin Kingsley at (914) 734-6695.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 3/14/05.

Sincerely,

Fred R. Dacimo
Site Vice President
Indian Point Energy Center

cc: next page

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ATTACHMENT 1 TO NL-05-036

ADDITIONAL INFORMATION REGARDING AST LICENSE AMENDMENT REQUEST FOR INDIAN POINT 3

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

NRC QUESTION:

The staff requests that Entergy perform dose calculations for the IP3 large break loss of coolant accident (LB LOCA) using a value of 10% for the iodine partition factor applied to the ECCS leakage outside containment, in lieu of the proposed value of 2.8%.

ENTERGY RESPONSE:

The dose results for LB LOCA, using an iodine partition factor of 10% instead of 2.8%, are provided in the following table. This new analysis also uses a value of 400 cfm for IP3 control room unfiltered inleakage, instead of the 700 cfm value previously used. The reduced inleakage value of 400 cfm bounds the recently reported (Entergy letter NL-05-020, dated 2/11/2005) tracer gas test results of less than 100 cfm. The new analysis for LB LOCA demonstrates that regulatory acceptance criteria remain satisfied with this revision to input assumptions.

Location	Calculated TEDE Dose (Rem)	TEDE Dose Limit (Rem)
Site Boundary	23.4	25
Low Population Zone	13.0	25
IP3 Control Room	4.98	5
IP2 Control Room	3.51	5

Entergy understands that additional time would be needed for staff review and acceptance of the proposed 2.8% partition factor. Therefore, using the above set of assumptions is the preferred approach for completing the AST license amendment review at this time. A new request regarding the partition factor assumption may be submitted in the future.

Applicable revised pages for the AST Licensing Report are provided in Attachment 2.

NRC QUESTION:

The staff requests that Entergy confirm the mode of operation being assumed for the Control Room Ventilation System (CRVS) for the Fuel Handling Accident (FHA) dose analysis.

ENTERGY RESPONSE:

The original license amendment request (Entergy letter NL-04-068, dated June 2, 2004) contained dose results which credited operator action to switch the CRVS from mode 2 to mode 3. CRVS mode 3 provides for filtration of outside air makeup to the control room, while CRVS mode 2 does not use filtration. Subsequent submittals (January 28 and February 11, 2005) also reported dose results based on operator action for switching the CRVS operating mode, while the latest submittal (February 22, 2005) included a scenario with no credit for operator action. Entergy now requests staff review and approval of the FHA analysis with no credit for operator action. This analysis assumes that the CRVS remains in mode 2. All other analysis assumptions are as described in NL-04-068, as supplemented. The analysis with no operator action yields an operator dose of 2.8 rem TEDE, as compared to a dose of 1.4 rem TEDE with operator action. The regulatory limit of 5 rem TEDE is satisfied with no credit for operator action.

Applicable revised pages for the AST Licensing Report are provided in Attachment 2.

NRC QUESTION:

The licensee proposes to include I-130 in the Tech Spec definition of Dose Equivalent I-131. This isotope is not included in the current version (Revision 3) of NUREG-1481, Standard Technical Specifications for Westinghouse plants.

ENTERGY RESPONSE:

Entergy's basis for including I-130 was provided in NL-04-162 dated December 22, 2004 (question 2) and this definition was previously approved by the NRC for IP2 (Amendment 241 dated October 27, 2004). However, Entergy acknowledges that the staff will issue the license amendment for IP3 without I-130 included in the definition.

ATTACHMENT 2 TO NL-05-036

ERRATA PAGES FOR
AST LICENSE AMENDMENT REQUEST
FOR INDIAN POINT 3

The following pages replace those previously provided in NL-04-068, dated June 2, 2004; Attachment III AST Licensing Report

Affected Page	Reason for Revision
11	Reflect new partition factor and unfiltered inleakage for LB LOCA analysis
12	Revised dose results for new LB LOCA analysis
35	Reflect FHA analysis assumption regarding no switchover from CRVS Mode 2
36	Updated dose result for FHA
37	Revised unfiltered inleakage used for LB LOCA analysis
42	Revised unfiltered inleakage used for LB LOCA analysis
50	Revised partition factor used for LB LOCA analysis
68	Reflect FHA analysis assumption of no credit for operator action regarding CRVS operation

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INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

reports for the application of the alternative source term methodology. It is also conservatively assumed that sedimentation removal does not continue beyond a DF of 1000.

2.2.4 Leakage of Sump Solution Outside of Containment

In accordance with RG 1.183, it is assumed that the iodine is instantaneously and homogeneously mixed in the primary containment sump water at the time of release from the core.

2.2.4.1 ECCS Leakage

When ECCS external recirculation is established following the LOCA, leakage is assumed to occur from ECCS equipment outside containment. The leakage goes into the auxiliary building and no filtration or holdup is credited for this release. Initially, the ECCS recirculation is internal to the containment and there is no potential for leakage outside containment. However, the switch to external recirculation occurs at 6.5 hours because of the need to switch from cold leg recirculation mode to hot leg recirculation mode. The ECCS leakage is modeled as 4.0 gallon/hr, which is doubled from the plant allowable leakage value of 2.0 gallon/hr consistent with RG 1.183. The leakage continues for the 30-day period following the accident considered in the analysis. The airborne fraction is modeled as 2.7% based on calculations taking into account solution pH, temperature of the leaked water, room volumes (where leakage occurs), and ventilation flows. A default value of 10%, however, has been used for the final reported doses.

2.2.4.2 Reactor Coolant Pump Seal Leak-off Line

During the first 4 hours of the LOCA event, leakage from the reactor coolant pump seal leak-off line is assumed at the rate of 1.0 gallon/hr which is doubled from the plant allowable leakage value of 0.50 gallon/hr consistent with RG 1.183. The airborne fraction is modeled as 10% consistent with RG 1.183.

2.2.5 Control Room Isolation

In the event of a large break LOCA, the low pressurizer pressure SI setpoint will be reached shortly after event initiation. The SI signal causes the control room heating, ventilation and air conditioning (HVAC) to switch from the normal operation mode to the accident mode of operation. It is assumed that the SI setpoint is reached immediately at the start of the event and a conservative 60-second delay time for switching from normal to accident operating mode (recirculation with filtered fresh air intake) is modeled. The assumed unfiltered inleakage has been reduced from 700 cfm to 400 cfm.

2.3 Acceptance Criteria

The offsite dose limit for a LOCA is 25 rem TEDE per RG 1.183. This is the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

2.4 Results and Conclusions

The calculated doses for the large break LOCA are:

Site Boundary	23.4 rem TEDE
Low Population Zone	13.0 rem TEDE
Control Room	4.98 rem TEDE

The control room dose from external sources (activity remaining in containment and activity outside the control room envelope) was determined to be negligible.

The acceptance criteria are met.

The site boundary dose reported is for the worst two-hour period, determined to be from 0.6 hours to 2.6 hours.

11.1.2 Iodine Chemical Form

The iodine released from the fuel was assumed to be 95-percent cesium iodide (CsI), 4.85-percent elemental iodine, and 0.15-percent organic iodine. It was assumed that all of the CsI was dissociated in the water and that the iodine re-evolved as elemental iodine. This was assumed to occur instantaneously. Thus, the FHA dose analysis was based on an initial iodine characterization of 99.85-percent elemental iodine and 0.15-percent organic iodine.

11.1.3 Water Scrubbing Removal of Activity

The activity released from the damaged fuel rods was assumed to be contained within gas bubbles that rise up through the water and are released into the atmosphere above the pit. As the bubbles pass through the water column, there is a significant removal of activity. RG 1.183 (Reference 2) identifies a DF of 500 for elemental iodine and no removal for organic iodine and noble gases. The DF of 500 for elemental iodine is based on having a water height of 23 feet or more. (Per the *Technical Specifications*, there are requirements for =23 feet of water above the stored spent fuel and above the reactor vessel flange during fuel-handling operations.)

The DF of 500 for elemental iodine is also based on fuel rod pressure of =1200 psig. There is the potential for fuel rod pressures to exceed 1200 psig (but remain less than 1500 psig). With this increase in fuel rod pressure, the DF is determined to remain above 400. Using a DF of 400 for elemental iodine and the defined iodine species split of 99.85-percent elemental and 0.15-percent organic, the overall DF would be 250. However, RG 1.183 (Reference 2) also specifies the overall DF for iodine to be 200. The overall DF of 200 has an associated elemental iodine DF of 285, and this value was used in the analysis together with a DF of 1.0 for organic iodine and noble gases.
The cesium released from the damaged fuel rods was assumed to remain in a nonvolatile form and not be released from the water.

11.1.4 Filtration of Release Paths

No credit was taken for removing iodine by filters, nor was credit taken for isolating release paths.

Although the containment purge will be automatically isolated on a purge line high-radiation alarm, isolation was not modeled in the analysis. The activity released from the damaged assembly was assumed to be released to the outside atmosphere over a 2-hour period. Since no filtration or containment isolation was modeled, this analysis supports refueling operation with the equipment hatch and the personnel air lock remaining open.

11.1.5 Control Room Isolation

It was assumed that the control room HVAC System is not isolated.

11.2 Acceptance Criteria

The offsite dose limit for an FHA is 6.3 rem TEDE per RG 1.183 (Reference 2). This is ~25 percent of the guideline value of 10CFR50.67. The limit for the control room dose is 5 rem TEDE, per 10CFR50.67.

11.3 Results and Conclusions

The calculated doses due to the FHA are:

Site Boundary	5.7 rem TEDE
Low Population Zone	2.1 rem TEDE
Control Room	2.8 rem TEDE

The acceptance criteria were met.

The SB dose reported was for the worst 2-hour period, determined to be from 0 to 2 hours.

The control room dose from activity outside the control room envelope was determined to be negligible.

12.0 Conclusions

RG 1.183 (Reference 2) defines an alternative source term model for use in evaluating the radiological consequences of a postulated large break loss-of-coolant accident with core melt. This alternative source term model also forms the basis for determining the radiological consequences for other design basis accidents as provided in RG 1.183.

The alternative source term methodology, as defined in RG 1.183 and its appendices, has been incorporated into the Indian Point 3 Nuclear Power Plant's design basis accident analyses to support control room habitability. Analyses of the radiological consequences of the large break LOCA, steam generator tube rupture, locked rotor, rod ejection, steamline break, small break LOCA, gas decay tank rupture, volume control tank rupture and holdup tank failure have been made using the RG 1.183 methodology. The calculated doses do not exceed the defined acceptance criteria. The fuel handling accident, which had previously been analyzed using the alternative source term methodology (with an SER issued by Reference 14), was reanalyzed using revised source term modeling and revised control room HVAC flow rates and allowable inleakage.

This report supports the following changes to Indian Point 3 Nuclear Power Plant's design and operation:

- The elimination of a requirement for filter efficiency operability requirement for the charcoal filters on the containment fan cooler units.
- An allowable unfiltered inleakage into the control room of up to 700 cfm (400 cfm for the LBLOCA).

This report also supports the complete removal of all filters from the fan cooler units (both the HEPA and the charcoal filters) if plant changes are also made to implement a passive means for pH adjustment of the post-accident sump solution (i.e., the installation of baskets filled with trisodium phosphate). With the addition of trisodium phosphate for sump solution pH control, the spray additive can be deleted.

Table 4: Control Room Parameters

Volume	47,200 ft^3	
Control Room Unfiltered In-Leakage	=700 cfm(1) (except LBLOCA used 400 cfm)	
Normal Ventilation Flow Rates Filtered Makeup Flow Rate Filtered Recirculation Flow Rate Unfiltered Makeup Flow Rate Unfiltered Recirculation Flow Rate	 0.0 cfm 0.0 cfm =1500 cfm (Not modeled - no impact on analyses)	
Post Accident Recirculation Flow Rates Filtered Makeup Flow Rate Filtered Recirculation Flow Rate Unfiltered Makeup Flow Rate Unfiltered Recirculation Flow Rate	<u>Option 1</u> =400 cfm =1000 cfm 0.0 cfm (Not modeled - no impact on analyses)	<u>Option 2</u> =1500 cfm 0.0 cfm 0.0 cfm (Not modeled - no impact on analyses)
Filter Efficiencies Elemental Iodine Organic (Methyl) Iodine Particulate	 90% 90% 99%	
R33 CR Radiation Monitor Setpoint	3.33E-4 µCi/mL (2)	
R33 CR Radiation Monitor Location	Ventilation Line drawing from CR bulk air	
R1 CR Gamma Dose Area Monitor Setpoint	1 mrad/hr	
R1 CR Gamma Dose Monitor Location	Wall in the control room outside of duct	
Delay to Switch CR HVAC from Normal Operation to Post Accident Operation after receiving a High Alarm Signal (radiation monitor) Based on Manual Action (min)	20 minutes	
Delay to Switch CR HVAC from Normal Operation to Post Accident Operation after receiving an SI signal (sec)	60 seconds	
Breathing Rate - Duration of the Event	3.5E-4 m^3/sec	
Occupancy Factors 0 - 24 hours 1 - 4 days 4 - 30 days	 1.0 0.6 0.4	

Notes:

1. All of the reported doses modeled a maximum 700 cfm unfiltered inleakage modeled in both CR modes of operation, with the exception of the tank ruptures (Sections 8, 9 and 10) which conservatively ignored inleakage.
2. The monitor setpoint is based on a 0.2 Mev/disintegration source, which is similar to Xe-133.

Table 12 (Cont.): Assumptions Used for Large Break LOCA Dose Analysis	
Containment spray DF Elemental Particulate	 200 1000
Credited containment sump volume	374,400 gal
Leakage of sump solution outside of containment 0 – 4 hours 4 – 6.5 hours > 6.5 hours	 1.0 gph 0.0 gph 4.0 gph
Iodine airborne fraction for leakage of sump solution outside of containment 0 – 4 hours 4 – 6.5 hours > 6.5 hours	 10.0% NA 2.7% (10% is used in the final calculation)
Control Room atmospheric dispersion (χ/Q) factors Releases from containment surface[1]: 0 – 2 hours 2 – 8 hours 8 – 24 hours 24 – 96 hours 96 – 720 hours	 3.57E-4 sec/m^3 3.12E-4 sec/m^3 1.24E-4 sec/m^3 1.06E-4 sec/m^3 7.99E-5 sec/m^3
Control Room atmospheric dispersion (χ/Q) factors Releases from containment vent[2][3]: 0 – 2 hours 2 – 8 hours 8 – 24 hours 24 – 96 hours 96 – 720 hours	 5.93E-4 sec/m^3 4.92E-4 sec/m^3 2.06E-4 sec/m^3 1.69E-4 sec/m^3 1.26E-4 sec/m^3

Notes:

1. Used for activity released via containment leakage
2. Used for activity released via leakage of sump solution outside of containment (RCP seal leak-off and ECCS)
3. Subsequent to submittal of the original version of this report, the χ/Q values were corrected from the original determination. The revised values presented below are slightly higher than those used in the original dose analysis and were incorporated in the final calculation:

0 – 2 hr	6.00E-4 sec/m^3
2 – 8 hr	5.20E-4 sec/m^3
8 – 24 hr	2.12E-4 sec/m^3
24 – 96 hr	1.76E-4 sec/m^3
96 – 720 hr	1.30E-4 sec/m^3

Table 24: Assumptions Used for FHA Analysis	
Source Term	
Core Total Fission Product Activity (84 hrs decay)	See Table 25
Number of Fuel Assemblies	193
Radial Peaking Factor	1.70
Fuel Rod Gap Fraction	
I-131	12%
Kr-85	30%
Other Iodines and Noble Gases	10%
Fuel Damaged	One assembly
Time after Shutdown	84 hours
Water Depth	23 feet
Overall Iodine Scrubbing Factor	200
Noble Gases Scrubbing Factor	1
Filter Efficiency	No filtration of releases assumed
Isolation of Release	No isolation of releases assumed
Time to Release All Activity	2 hours
Time to Start Crediting Emergency Control Room HVAC	No Credit Taken
Control Room Atmospheric Dispersion (χ/Q) Factor Containment Vent(1): 0 - 2 hours	5.93E-4 sec/m^3

Note:

1 Subsequent to submittal of the original version of this report, the χ/Q value was corrected from the original determination. The revised value increased slightly to 6.00E-4 sec/m^3 and was used in the final calculation.