June 30, 2008

Mr. Keith J. Polson
Vice President Nine Mile Point
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P.O. Box 63
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SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT NO. 2 - CORRECTION TO SAFETY EVALUATION SUPPORTING AMENDMENT NO. 125 RE: IMPLEMENTATION OF ALTERNATIVE RADIOLOGICAL SOURCE TERM (TAC NO. MD5758)

Dear Mr. Polson:

On May 29, 2008, the Nuclear Regulatory Commission (NRC) issued Amendment No. 125 to Renewed Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station, Unit No. 2 (NMP-2). This amendment changed the NMP-2 TSs by revising the accident source term in the design basis radiological consequence analyses in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67.

Subsequent to the issuance, Mr. Dennis Vandeputte of your staff pointed out a number of errors in the safety evaluation (SE) supporting the amendment. We agree that editorial errors had been inadvertently made, resulting in several inaccurate statements in the SE. Enclosed please find the corrected pages 18, 23, 27, Table 3.2, and Table 3.2.3 of the SE, with side bars highlighting the areas of correction.

The NRC regrets any inconvenience that these editorial errors may have caused. If there are any questions regarding this matter, please contact me at 301-415-1030.

Sincerely,

/RA/

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-410

Enclosure:
As stated

cc w/encl: See next page

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reflect the effectiveness of drywell spray activity removal in containment upstream of this pathway. Though, as discussed earlier, the LOCA activity leak rates are reduced by a factor of 2 after 24 hours, based on decreasing containment pressure, the licensee conservatively does not credit this reduction to increase removal efficiency by natural deposition in the bypass lines.

For elemental iodine, the licensee assumed that a DF of 2 applies for natural deposition in the bypass piping. This is consistent with the licensee's assumption of elemental iodine plate-out on aerosol particulate, as was used in their drywell spray calculation. The NRC staff notes that the conservatively calculated aerosol activity removal by settling in the piping exceeds a DF of 2, and finds that the DF of 2 for elemental iodine is acceptable. The licensee took no credit for organic iodine removal.

3.2.1.3 Direct Shine Dose

The licensee's evaluation of post-LOCA shine doses to control room personnel from the RB airborne activity cloud, the passing external activity plume, and the activity loaded control room filters was based on the historical NMP2 design basis. The historical external shine doses for NMP2 were calculated using the release characteristics associated with a TID-14844 source term and model based on RG 1.3, "Assumptions for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors." The licensee compared AST-calculated activity releases at NMP2, based on NUREG-1465 and RG 1.183 methodology, to those historically calculated, and showed that the historically calculated values will bound. The NRC staff agrees that the AST methodology is bounded by the TID-14844 source term, based on the following three (3) reasons, as documented in the licensee's RAI response of January 7, 2008 (ML080140133):

- The TID-14844 activity releases are instantaneous, whereas NUREG-1465 allows for a release linearly distributed over a period of 2 hours.
- Removal of iodine by sprays and deposition is credible in AST-based models, where no such removal mechanisms were historically applied.
- RG 1.183 allows for a leak rate reduction of 50% at 24 hours, but analyses performed consistent with RG 1.3 take no such reduction.

Though it is agreed that the activity released and available to contribute the NMP2 control room shine dose is bounded by the historical analysis, the NRC staff does note that the historical shine dose calculation implemented the QADMOD point-kernel code. Also, verification of the historically calculated doses was performed using the MicroShield point-kernel code. Both the QADMOD and MicroShield codes are point-kernel integration codes used for general purpose gamma shielding analyses. The potentially complex geometries associated with the direct shine dose assessments, such as those performed for NMP2, are generally more effectively modeled using more powerful particle transport codes. Specifically, MicroShield sacrifices accuracy in lieu of simplicity when modeling complex multidimensional systems of sources, shields, and receivers. However, though it also uses a point-kernel method that implements buildup factors and is subject to mistreatment of albedo effects, QADMOD does allow for the modeling of complex geometries using combinatorial geometry.

It is the NRC staff's judgment that the licensee's direct shine dose model implements sufficient and substantial conservatism that compensates for potential non-conservative treatment of the resulting from the spiked activity meets the lower acceptance criterion for the equilibrium activity case that is suggested in Table 6 of RG 1.183, so would therefore meet the higher acceptance criterion for the iodine spike case in RG 1.183 and SRP 15.0.1. Also, because the radiological consequences are directly related to the coolant activity released, and since the equilibrium concentration case has a lower coolant activity release than the iodine spike case, the equilibrium concentration case would meet the equilibrium concentration acceptance criterion.

Consistent with RG 1.183 guidance, the licensee assumed that the speciation of radioactive iodine released from failed fuel is 95% aerosol (particulate), 4.85% elemental, and 0.15% organic. However, the speciation of radioactive iodine released by coolant blowdown is 97% elemental and 3% organic. Because no fuel failure was assumed, the coolant iodine speciation was used for this DBA analysis. The licensee also considered the maximum TS noble gas and cesium activity to be available for release from the steam blowdown and coolant, respectively. This treatment is conservative, with respect to the RG 1.183 guidance, and acceptable to the NRC staff, because the guidance does not explicitly suggest that cesium activity be considered as a dose contributor.

3.2.3.2 Transport Methodology and Assumptions

The licensee has defined the design-basis MSLB accident as an instantaneous circumferential break of one main steam line outside the secondary containment, downstream of the outside isolation valve. It is assumed that pipe end displacement due to this double ended guillotine break is such that the maximum blowdown rate is permitted to occur. The licensee assumed that the break flow is terminated by closure of the MSIVs, and that the coolant mass released through the break includes the line inventory plus the system mass released through the break prior to isolation. The radiological consequences of an MSLB outside secondary containment will bound the consequences of a break inside containment. Thus, only an MSLB outside of containment was considered with regard to the radiological consequences.

Consistent with the current NMP2 licensing basis, the licensee assumed break isolation in 5.5 seconds, corresponding to the maximum MSIV closing time of 5 seconds, plus an assumed closure signal delay time of 0.5 seconds. The licensee took no credit for reduction in break flow as the valves are closing. In their LAR (ML071580314), the licensee stated a total assumed coolant mass release is 4.85 E+07 gm, consisting of 2.56 E+07 gm of liquid, 1.58 E+07 gm of flashed liquid, and 7.10 E+06 gm of steam. The licensee also assumed that, following accident initiation, the radionuclide inventory from the released coolant reaches the environment instantaneously, taking no credit for holdup in the turbine building. An infinite exchange rate between the control room and the environment was assumed, and no credit was taken for control room filtration, other iodine removal mechanisms, or decay. The release modeled by the licensee was assumed to waft over the control room intake at a rate of 1 m/s, leaving it resident and contributing to dose for 124 seconds, which is based on the size of the activity "puff" that results from the released mass of coolant. The NRC staff finds the use of this puff release model to be acceptable because of the very short duration of the MSLB release and inherent conservatism of the instantaneous release and intake assumed by the licensee.

The licensee used a spreadsheet to perform the calculations for their analysis of the dose consequences resulting from this design-basis MSLB. This spreadsheet was provided for NRC were found to meet the applicable accident dose acceptance criteria and are, therefore, acceptable.

3.3 Control Room Habitability and Modeling

The current NMP2 DBA analyses, as described in USAR Chapter 15, do not consider unfiltered inleakage in calculating control room dose; therefore, the control room dose model provided in the revised DBA accident analyses that support this AST-based LAR represents a change in the NMP2 licensing basis.

For their revised analyses where control room isolation and/or filtration is credited, the licensee assumed an emergency mode control room intake flow rate of 2500 cfm ±10%, and assumed 99% filtration efficiency for elemental iodine, organic iodine, and particulate forms of radionuclide activity. For conservatism, the upper flow uncertainty value, 2750 cfm, is used for modeling, then, as a design basis, reduced to 1650 cfm at 20 minutes. Where control room filtration is credited, the licensee assumed that the control room was automatically isolated on a LOCA signal, and that filtration was delayed for 80 seconds.

In a letter dated January 31, 2005, from the licensee to the NRC staff (ML050460309), it is indicated that the highest measured unfiltered inleakage into the NMP2 control room is 174 cfm. For the DBA analyses that model actual NMP2 control room functionality, the licensee assumed an unfiltered inleakage of 250 cfm, to bound the worst-case unfiltered inleakage as tested. This value is conservative and provides margin for future measurements of control room inleakage. The major parameters and assumptions used by the licensee for modeling of the control room, and found acceptable to the NRC staff, are presented in Table 3.3.

3.4 Technical Specification Changes

3.4.1 Revision to the TS 1.0 Definition of "*Dose Equivalent I-131*"

The licensee has proposed to add the definition of *Dose Equivalent I-131* to NMP2 TS Section 1.0. The licensee's revised DBA dose consequence analyses use committed effective dose equivalent dose conversion factors from Table 2.1 of Federal Guidance Report (FGR) 11, ORNL, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," as the source of thyroid dose conversion factors instead of the current TID-14844, RG 1.109, Rev. 1, and ICRP 30 referenced dose conversion factors.

With the implementation of the AST, the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC 19, are replaced by the TEDE criteria of 10 CFR 50.67(b)(2). This new definition reflects adoption of the dose conversion factors and dose consequences of the revised radiological analyses. Thus, this proposed revision to the definition of *Dose Equivalent I-131* is supported by the justification for the proposed licensing basis revision to implement the AST, and conforms to the implementation of the AST and the TEDE criteria in 10 CFR 50.67. Therefore, the NRC staff finds the proposed revision to the TS 1.0 definition *Dose Equivalent I-131* acceptable.

Table 3.2

Licensee Calculated Radiological Consequences of Design Basis Accidents at NMP2

Design Basis Accident	Control Room		[a]EAB		LPZ	
	[b]Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)	[c]Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)	[d]Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)
LOCA	1.65E+00	*5.0*	6.57E-01	*25*	7.69E-01	*25*
FHA	3.15E+00	*5.0*	4.50E-01	*6.3*	6.13E-02	*6.3*
MSLB	2.96E+00	*5.0*	3.92E-01	*25*	5.34E-02	*25*
CRDA						
Case 1	1.26E+00	*5.0*	5.68E-01	*6.3*	7.73E-02	*6.3*
Case 2	2.31E+00	*5.0*	1.03E+00	*6.3*	1.17E+00	*6.3*

Revised by letter dated June 30, 2008

[a] The licensee calculated the EAB dose for the worst 2-hour period of the accident duration.
[b] The licensee's control room dose results have been rounded to three significant digit precision.
[c] The licensee's EAB dose results have been rounded to three significant digit precision.
[d] The licensee's LPZ dose results have been rounded to three significant digit precision.

Table 3.2.3

Key Parameters Used in Radiological Consequence Analysis of Main Steam Line Break Accident

Parameter	Value
Reactor Core Power, MWth	4067
Failed Fuel, %	0
Reactor Coolant Activity, µCi/gm DE I-131 Equilibrium Iodine Activity Pre-accident Iodine Spike Activity	 0.2 4.0
Iodine-131 DCF, rem/Ci	3.29E+04
Iodine Speciation from Coolant, % Elemental Organic	 97 3
Time Until MSIV Isolation, sec	5.5
Coolant Mass Blowdown, gm Liquid Steam Total	 4.1E+07 7.1E+06 4.9E+07
Time for Puff to Traverse Control Room Intake, sec	124
Atmospheric Dispersion Factors	Tables 3.1.1 and 3.1.2