Entergy

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

September 7, 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

Supplemental Information Regarding Alternative Source Term License Amendment Request (TAC No. MD3087)

Dear Sir or Madam:

By letter dated September 25, 2006, Nuclear Management Company, LLC, the former licensee for the Palisades Nuclear Plant (PNP), requested Nuclear Regulatory Commission (NRC) review and approval of a license amendment request (LAR) for PNP. The LAR would modify the PNP licensing basis to adopt the alternative source term methodology. On June 15, 2007, Entergy Nuclear Operations, Inc., (ENO) responded to requests for additional information (RAI) on the LAR.

In a July 17, 2007, telephone conference call discussion regarding the information provided in the June 15, 2007, RAI response, the NRC requested the plant drawings used to calculate the distances between the release and receptor locations in the determination of atmospheric dispersion factors. These drawings are included in Enclosure 1 of this letter.

Enclosure 2 provides information about three issues related to the release and receptor locations used to determine the atmospheric dispersion factors for PNP. The enclosure also includes data gathered from use of surveying techniques, which provide an independent verification of the release and receptor locations.

Enclosure 3 describes the results and planned action resulting from a review for limiting high burnup fuel rods that exceed a stipulation in the Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

On June 29, 2007, ENO submitted a LAR for replacement of the containment sump buffer. In the conference call on July 17, 2007, ENO agreed that the implementation of the alternative source term LAR should be contingent on the approval of the LAR for

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replacement of the containment sump buffer. ENO will work with the NRC project manager, as necessary, to coordinate the changes.

No other information in the September 25, 2006, LAR is affected by this additional information. The No Significant Hazards Consideration and the Environmental Consideration provided in Enclosure 1 of the September 25, 2006, submittal are not affected by this additional information.

Summary of Commitments

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

Prior to loading the Cycle 21 core, ENO will revise the procedure used to check the adequacy of a core design to include an evaluation on the pin power/burnup of the design core against the following criteria:

- Fewer than 21 rods in any one assembly violate the "54/6.3" criterion.
- Fewer than 20 assemblies in any core design contain at least one rod that violates the "54/6.3" criterion.
- All rods that violate the "54/6.3" criterion have a rod average linear heat generation rate of less than 6.7 kW/ft.
- All rods that violate the "54/6.3" criterion have a rod burnup of less than 58.5 GWD/MTU.
- In any assembly containing any rods that violate the "54/6.3" criterion there are at least four times as many rods that have total radial peaking factor of less than ¾ of the total radial peaking factor limit of 2.04.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 7, 2007.

Christopher J. Schwarz
Site Vice President
Palisades Nuclear Plant

Enclosures (3)

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

ENCLOSURE 1
PLANT DRAWINGS

In an electronic mail message dated March 29, 2007, the Nuclear Regulatory Commission (NRC) requested that the release and receptor component locations be added to a drawing to assist in the review of the license amendment request (LAR). Entergy Nuclear Operations, Inc. (ENO) provided drawings on June 15, 2007, that had approximate locations of the release and receptor components. However, the drawings did not include the needed detail for the NRC's review. The plant drawings listed below were used to determine the distances between the release and receptor components. The drawings are those referenced in Numerical Applications Inc. (NAI) report number NAI-1149-002, revision 0, "Determination of Atmospheric Dispersion Factors for Palisades," Attachment D, "ARCON96 Cases Input Reference Compilation." The NAI report was provided to the NRC with the September 25, 2006, LAR for Alternative Source Term.

Enclosed Plant Drawings

C-2 sheet 1 revision 16
C-18 sheet 55 revision 3
C-18 sheet 67 revision 6
C-38 revision 6
C-50 revision 20
C-51 revision 30
C-127 revision 2
C-141 revision 5
M-15 revision 12
M-29 revision 13
M-40 revision 29
M-42 revision 32
M-65 revision 13
M-79 revision 3
M-113 revision 11
M-119 sheet 3 revision 7
M-119 sheet 4 revision 8
M-140 revision 7
M-990 revision 0

19 Drawings Follow

ENCLOSURE 2
PALISADES RELEASE AND RECEPTOR LOCATION DATA

In a September 25, 2006, letter "Alternative Source Term - Proposed License Amendment," the licensee provided a Numerical Applications Inc. (NAI) report/calculation NAI-1149-002, "Determination of Atmospheric Dispersion Factors for Palisades," revision 0. The dimensional input and calculations in Attachment D of NAI-1149-002, revision 0, have been re-verified, resulting in corrections to distance and direction indicated in Table 1 below.

Additionally, information is provided on three issues related to the release and receptor locations used to determine the atmospheric dispersion factors for the Palisades Nuclear Plant (PNP). Item a. is a description of an error in reading one of the drawing dimensions used in the release to receptor locations calculation submitted in the licensee's September 25, 2006, letter. Item b. describes a small drawing distance discrepancy and provides results using surveying techniques. Item c. discusses the extent of condition walkdowns performed in response to the drawing reading error.

The re-verification found the following issues:

a. The distance between reference lines 29 and 21 on drawing C-51 (see Enclosure 1) was incorrectly read and transcribed as 30 feet on page D6 of Attachment D to NAI-1149-002, revision 0. The error carried forward on several of the calculated distances in Attachment D of the report. The correct distance is 50 feet. This distance has been incorporated into the results in Table 1 below. The error is a conservative error in that the affected release to receptor distances are all increased and the directions are not greatly impacted. Table 1 reproduces and corrects release/receptor data from Attachment D to NAI-1149-002, revision 0.

   Results in Table 1 are given to two decimal places for distances in meters to zero decimal places for direction in degrees. Direction is given with respect to (wrt) true north. The crossed out distances and directions in Table 1 are those that have been corrected due to the error described above.

   The drawing reading error is a conservative error, resulting in a conservative over-prediction of the X/Q by at least 20%, based on the increased actual distance between release and receptor and the small change in direction.

b. The distance between reference line "G" and the containment centerline is indicated as 26'-0" on drawing M-990, while the same distance is indicated as 26'- 6" on drawings C-50 and C-51 (See Enclosure 1 for the referenced drawings). This impact is less than three inches on distance and results in slightly worse agreement with the survey data. The impact on results is insignificant and is not incorporated into the results in Table 1 below.

The distances and directions (except for the containment equipment door) have also been independently confirmed by data provided by a surveying firm. The relative release and receptor locations were determined by surveying techniques. Differences are within three percent for all locations for distance and direction from those calculated. A summary of the survey data is compared to the drawing dimension data in Table 2 below.

The survey data confirm accuracy of the distances calculated from the drawings. Therefore, no changes to the analysis are required.

c. During the extent of condition walkdowns performed in response to the drawing reading error discussed above, two additional release points for steam line breaks outside containment were identified. They are the turbine building feedwater exhaust fans V-22A and V-22B. These release points were not identified as turbine building release points and were not included in the submitted X/Q calculations. The new release points are applicable only to the primary coolant system release and faulted steam generator release paths for the main steam line break (MSLB) control room dose analysis. Control room dose analysis receptors are the normal and emergency control room intakes.

   As discussed below, the submitted MSLB analysis contains substantial conservatisms that can be demonstrated to address any loss of margin due to the new release points.

   Based on the X/Qs and the distances and directions between various release-receptor pairs in the vicinity of the new release points, the X/Q for the new release points could potentially be a factor of two or three greater than those used in the analysis. Release points considered are the atmospheric dump valves, the main steam safety valves, the turbine building roof exhausters, and the turbine building feedwater exhaust fans. Receptors considered are the normal and emergency control room intakes.

   The predicted number of fuel failures based on the MSLB core response analysis of record is zero. To avoid consideration of pre-existing and event generated iodine spiking cases, a bounding fuel failure of two percent was used to calculate the source term for the MSLB analysis. Based on the submitted MSLB analysis and a conversion of the source term to dose equivalent iodine-131, it is seen that the two percent fuel failures results in a source term that is between five and six times larger than the source term from the event generated iodine spike case. A reduction in fuel failures by a factor of five would result in a source term based on 0.4% fuel failures, which bounds both the predicted number of fuel failures (zero) and the pre-existing and event generated iodine spike source terms. A reduction in fuel failures by a factor of five decreases the source term for the failed fuel by a factor of five, and decreases the dose results affected by the turbine building exhaust fans by a factor greater than three.

Therefore, margin in the two percent fuel failure input for the MSLB, which bounds the thermally-hydraulically calculated fuel failure of zero, is sufficient to address a loss of margin associated with the new release points and substantial conservatism remains in the submitted analysis as discussed below.

In addition to the conservatisms described in the three items above, there is additional margin available that is not credited. Conservatisms used in the calculation of the onsite atmospheric relative concentrations for the MSLB X/Q that are not used to offset the loss of margin due to the new release points include the following:

- The use of ARCON96 to calculate dispersion factors, given that Palisades wind tunnel testing results indicate substantial general conservatism in the ARCON96 calculated results.

- The use of only ground level releases, i.e., no vent releases, are used, and

- Diffuse area releases are not used for the release, even though a MSLB in the turbine building would result in multiple turbine building release points, many of which would be much further away from the receptors than the release points considered.

In addition, no credit is taken for plume rise. The high velocity, buoyant release from the main steam safety valves or atmospheric dump valves, and turbine building pressurization and hot steam releases, would result in a significant fraction of the release bypassing the normal intakes. Also, no credit is taken for building wakes. Moreover, only straight-line distances are used in determining the input for the release to receptor distances; no "stretched-string" distance is credited for the intervening structures.

Table 1: Palisades Revised Release and Receptor Data

Release Point	Receptor Point	Release Height (meters)	Receptor Height (meters)	Distance (meters)	Direction wrt True North (degrees)
Closest Containment Point	Normal Control Room Intake "A"	22.53	22.53	~~24.20~~ 29.35	~~168~~ 172
Closest Containment Point	Normal Control Room Intake "B"	22.53	22.53	~~21.49~~ 26.94	~~174~~ 178
Closest Containment Point	Emergency Control Room Intake	14.94	14.94	95.05	202
SIRW Tank Vent	Normal Control Room Intake "A"	24.51	22.53	10.56	157
SIRW Tank Vent	Normal Control Room Intake "B"	24.51	22.53	7.71	184
SIRW Tank Vent	Emergency Control Room Intake	24.51	14.94	81.19	214
Stack Vent	Normal Control Room Intake "A"	58.52	22.53	~~22.27~~ 27.53	~~169~~ 176
Stack Vent	Normal Control Room Intake "B"	58.52	22.53	~~19.83~~ 25.59	~~181~~ 186
Stack Vent	Emergency Control Room Intake	58.52	14.94	97.45	209
ADV	Normal Control Room Intake "A"	17.37	22.53	~~20.08~~ 26.40	~~192~~ 195
ADV	Normal Control Room Intake "B"	17.37	22.53	~~19.78~~ 26.17	~~207~~ 206
ADV	Emergency Control Room Intake	17.37	14.94	100.39	214
SSRV (West Bank)	Normal Control Room Intake "A"	17.37	22.53	~~17.65~~ 23.57	~~187~~ 191
SSRV (West Bank)	Normal Control Room Intake "B"	17.37	22.53	~~16.96~~ 23.06	204

Release Point	Receptor Point	Release Height (meters)	Receptor Height (meters)	Distance (meters)	Direction wrt True North (degrees)
SSRV (East Bank)	Emergency Control Room Intake	17.37	14.94	96.78	213
Containment Equipment Door	Normal Intake 'A'	20.42	22.53	~~26.84~~ 31.05	~~152~~ 161
Containment Equipment Door	Normal Intake 'B'	20.42	22.53	~~23.06~~ 27.85	~~161~~ 169
Containment Equipment Door	Emergency Intake	20.42	14.94	95.42	204
Turbine Building NE Roof Exhauster	Normal Intake 'A'	27.43	22.53	~~20.79~~ 24.85	~~257~~ 246
Turbine Building NE Roof Exhauster	Normal Intake 'B'	27.43	22.53	~~25.19~~ 28.63	~~264~~ 253
Turbine Building NE Roof Exhauster	Emergency Intake	27.43	14.94	99.09	227
Turbine Building NW Roof Exhauster	Normal Intake 'A'	27.43	22.53	~~22.29~~ 25.65	~~266~~ 253
Turbine Building NW Roof Exhauster	Normal Intake 'B'	27.43	22.53	~~27.00~~ 29.84	~~271~~ 260
Turbine Building NW Roof Exhauster	Emergency Intake	27.43	14.94	98.55	229

TABLE 2
DRAWING AND SURVEY DATA COMPARISON

Release	Receptor	Calculated Distance (meters)	Calculated Direction (degrees)	Survey Distance (meters)	Survey Direction (degrees)	Distance Difference (%)	Direction Difference (%)
Closest Cont. Pt.	Normal CR Intake "A"	29.35	172	29.23	174	0.4%	-1.2%
Closest Cont. Pt.	Normal CR Intake "B"	26.94	178	26.84	179	0.4%	-0.6%
Closest Cont. Pt.	Emergency CR Intake	95.05	202	95.48	204	-0.5%	-1.0%
SIRW Tank Vent	Normal CR Intake "A"	10.56	157	10.84	156	-2.7%	0.6%
SIRW Tank Vent	Normal CR Intake "B"	7.71	184	7.81	182	-1.3%	1.1%
SIRW Tank Vent	Emergency CR Intake	81.19	214	81.57	215	-0.5%	-0.5%
Stack Vent	Normal CR Intake "A"	27.53	176	27.43	177	0.4%	-0.6%
Stack Vent	Normal CR Intake "B"	25.59	186	25.50	187	0.4%	-0.5%
Stack Vent	Emergency CR Intake	97.45	209	97.88	210	-0.4%	-0.5%
ADV	Normal CR Intake "A"	26.40	195	26.34	196	0.2%	-0.5%
ADV	Normal CR Intake "B"	26.17	206	26.10	207	0.3%	-0.5%
ADV	Emergency CR Intake	100.39	214	100.86	215	-0.5%	-0.5%
MSSV (West Bank)	Normal CR Intake "A"	23.57	191	23.48	192	0.4%	-0.5%
MSSV (West Bank)	Normal CR Intake "B"	23.06	204	22.97	205	0.4%	-0.5%
MSSV (East Bank)	Emergency CR Intake	96.78	213	97.23	214	-0.5%	-0.5%
Cont. Equip. Door	Normal CR Intake "A"	31.05	161	N/A	N/A	N/A	N/A
Cont. Equip. Door	Normal CR Intake "B"	27.85	169	N/A	N/A	N/A	N/A
Cont. Equip. Door	Emergency CR Intake	95.42	204	N/A	N/A	N/A	N/A
TB NE Roof Exhauster	Normal CR Intake "A"	24.85	246	25.22	246	-1.5%	0.0%
TB NE Roof Exhauster	Normal CR Intake "B"	28.63	253	28.89	253	-0.9%	-0.0%
TB NE Roof Exhauster	Emergency CR Intake	99.09	227	100.10	228	-1.0%	-0.4%
TB NW Roof Exhauster	Normal CR Intake "A"	25.65	253	25.58	255	0.3%	-0.8%
TB NW Roof Exhauster	Normal CR Intake "B"	29.84	260	29.71	261	0.4%	-0.4%
TB NW Roof Exhauster	Emergency CR Intake	98.55	229	99.03	230	-0.5%	-0.4%

ENCLOSURE 3
LINEAR HEAT GENERATION RATE FOR HIGH BURNUP RODS

Nuclear Management Company, LLC, the former licensee for Palisades Nuclear Plant (PNP) submitted an alternative source term (AST) license amendment request (LAR), on September 25, 2006. The AST LAR uses Table 3 of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," explicitly for the main steam line break (MSLB), fuel handling accident, and spent fuel cask drop events. All other events use either Table 1 release fractions or release fractions specified in the specific event guidelines provided in the applicable Appendices of RG 1.183. The control rod ejection accident uses gap release fractions specified in Appendix H of RG 1.183, but the Table 3 restriction described below is conservatively assumed to apply to the control rod ejection source term as well.

Table 3 of RG 1.183 provides the release fractions associated with the fission product inventory residing in the fuel rod gap for non-loss-of-coolant accident (LOCA) events. Footnote 11 of Table 3 restricts the use of the gap fractions to currently licensed light water reactor fuel that do not exceed a peak rod burnup of 62 gigawatt-days per metric ton uranium (GWD/MTU) and provided the maximum linear heat generation rate does not exceed 6.3 kilowatts per foot (kW/ft) peak rod average power for burnups exceeding 54 GWD/MTU (referred to as the "54/6.3 criterion"). Gap release fractions for rods that do not meet this criterion are considered by the NRC on a case-by-case basis.

The AST LAR submittal included an assumption that no fuel rod will have a linear heat generation rate greater than 6.3 kW/ft when rod burnup exceeds 54 GWD/MTU. However, Entergy Nuclear Operations, Inc. (ENO) recently determined that this limit is predicted to be exceeded for a small number of rods at some point in the current and future fuel cycles to which the AST analysis applies.

Section 1.7.1 of the AST Licensing Technical Report for Palisades (NAI-1149-027, revision 1, Page 11 of Enclosure 4 of AST LAR) and Item 3.2 of the regulatory compliance matrix (Page 3 of Enclosure 7 of AST LAR) state that the heat generation rate is limited to 6.3 kW/ft for rod average burnups in excess of 54 GWD/MTU. Design input for the AST analysis addresses the RG 1.183, footnote 11, restriction. The design input states that the core average linear heat generation rate is approximately five kW/ft and that it is extremely unlikely that rods with greater than 54 GWD/MTU will be operated with peaking factors greater than unity. However, the assertion regarding the peaking factors is an incorrect statement, as the "54/6.3" criterion is predicted to be exceeded for a small number of rods in the current and future fuel cycles to which the AST analysis would be applied. The information below clarifies the application of margins, and demonstrates that the submitted analysis remains bounding with respect to the operation of PNP fuel cycles.

Bounding PNP core design calculations using SIMULATE-3 (Studsvik CMS Steady-State 3-D Reactor Simulator Version 5.08.13) for the current operating cycle, cycle 19, and the next operating cycle, cycle 20, have been reviewed in detail to determine the maximum number of fuel rods that may exceed the "54/6.3" criterion. The results of the review are presented in the following tables and discussed below.

Cycle 19 – 14,500 MWD/MTU[1]					
Full Core Location (1/8 Core Location)	Number of Assemblies with Rods Exceeding "54/6.3" Criterion[2] (Assembly ID)	Number of Rods Exceeding "54/6.3" Criterion per Assembly	Maximum Rod Average Burnup for rods exceeding "54/6.3" criterion (GWD/MTU)	Maximum Rod Average Power for rods exceeding "54/6.3" Criterion (kW/ft)	Number of Compensating Rods[3] / Required Number of Compensating Rods per Assembly with rods exceeding "54/6.3" Criterion
09,12 (4,1)	1 (U349)	5	~54	~6.4	211 / 20
09,05 (4,1)	1 (U350)	5	~54	~6.4	211 / 20
08,05 (4,1)	1 (U351)	5	~54	~6.4	211 / 20
08,12 (4,1)	1 (U352)	5	~54	~6.4	211 / 20
12,09 (4,1)	1 (U357)	3	~54	~6.4	213 / 12
12,08 (4,1)	1 (U358)	3	~54	~6.4	213 / 12
05,08 (4,1)	1 (U359)	3	~54	~6.4	213 / 12
05,09 (4,1)	1 (U360)	3	~54	~6.4	213 / 12
Total	8	32			

Cycle 20 – 15,400 MWD/MTU[1]					
Full Core Location (1/8 Core Location)	Number of Assemblies with Rods Exceeding "54/6.3" Criterion[2] (Assembly ID)	Number of Rods Exceeding "54/6.3" Criterion per Assembly	Maximum Rod Average Burnup for rods exceeding "54/6.3" criterion (GWD/MTU)	Maximum Rod Average Power for rods exceeding "54/6.3" Criterion (kW/ft)	Number of Compensating Rods[3] / Required Number of Compensating Rods per Assembly with rods exceeding "54/6.3" Criterion
13,09 (5,1)	1 (V105)	2	~55	~6.4	214 / 8
13,08 (5,1)	1 (V106)	2	~55	~6.4	214 / 8
04,08 (5,1)	1 (V107)	2	~55	~6.4	214 / 8
04,09 (5,1)	1 (V108)	2	~55	~6.4	214 / 8
11,11 (3,3)	1 (V109)	6	~55	~6.4	210 / 24
11,06 (3,3)	1 (V110)	6	~55	~6.4	210 / 24
06,06 (3,3)	1 (V111)	6	~55	~6.4	210 / 24
06,11 (3,3)	1 (V112)	6	~55	~6.4	210 / 24
09,13 (5,1)	1 (V113)	6	~55	~6.4	210 / 24
09,04 (5,1)	1 (V114)	6	~55	~6.4	210 / 24
08,04 (5,1)	1 (V115)	6	~55	~6.4	210 / 24
08,13 (5,1)	1 (V116)	6	~55	~6.4	210 / 24
Total	12	56			

(1) Analyzed Cycle 19 cycle length bounds expected length of 13,400 MWD/MTU.

(2) Due to inaccuracies in the SIMULATE-3 reconstruction of pin burnup, an uncertainty factor of 0.5 GWD/MTU was used in the review, i.e., fuel rods with burnup in excess of a SIMULATE-3 calculated value of 53.5 GWD/MTU were considered to exceed the 54 GWD/MTU burnup aspect of the "54/6.3" criterion.

(3) Fewer rods would be available for compensation if an assembly were a shield assembly, an armored assembly, or a reconstituted assembly.

The data above indicate that between eight and twelve of the 204 fuel assemblies in cycles 19 and 20 may contain fuel rods that exceed the “54/6.3” criterion at some point during these cycles. Of the 216 rods in each of those assemblies, only two to six rods would exceed the “54/6.3” criterion during the cycle. Individual rods that exceed the criterion for an intermediate burnup step and subsequently fall below the criterion, are included in the reported number of rods exceeding the “54/6.3” criterion. The SIMULATE-3 runs indicate rod average burnups would not exceed 56 GWD/MTU and rod average powers do not exceed 6.5 kW/ft for these rods.

Given the small number of rods that exceed the “54/6.3” criterion, doubling the gap release fractions for the rods that do not meet the criterion can be accommodated within the margins of the affected analyses submitted in the AST LAR without a significant impact on overall margin. The basis for this conclusion is as follows:

A peaking factor of 2.04 has been applied conservatively to the source terms for all rods assumed to fail due to cladding failure for the main steam line break, fuel handling accident, spent fuel cask drop event, and the control rod ejection event. The peaking factor value of 2.04 is the total radial peaking factor limit (F_r^{TS}) as defined in Section 2.4 of the Palisades Core Operating Limits Report. The source term for these events is directly proportional to the applied peaking factor and the gap release fraction. The excess margin between the actual maximum peaking factors for fuel rods that meet the “54/6.3” criterion in assemblies that also contain fuel rods that do not meet the “54/6.3” criterion, can be applied to the source terms used in these analyses to offset an increase in release fraction by a factor of two for the rods that do not meet the “54/6.3” criterion.

The last column in the tables above indicates the number of rods that are available to compensate for the rods that do not meet the "54/6.3" criterion. Compensating rods meet the “54/6.3” criterion and have sufficient peaking factor margin. Peaking factors for the compensating rods are required to be less than ¾ of the conservatively applied peaking factor of 2.04. This would provide four compensating rods to completely offset the doubling of the gap release fraction of each of the rods that do not meet the “54/6.3” criterion. The tables also indicate the required number of rods to just balance out the increased gap release fractions. It can be seen that in each assembly there is significant excess margin to accommodate the factor of two increase in gap release fraction for the rods not meeting the criterion.

This demonstrates that the margin in the submitted analyses can be allocated to address rods that do not meet the “54/6.3” criterion for cycles 19 and 20. To ensure appropriate margin is maintained in future cycles, the procedure used to check the adequacy of a core design will be revised to include an evaluation on the pin power/burnup of the design core against the following criteria:

- Fewer than 21 rods in any one assembly violate the "54/6.3" criterion.
- Fewer than 20 assemblies in any core design contain at least one rod that violates the "54/6.3" criterion.
- All rods that violate the "54/6.3" criterion have a rod average linear heat generation rate of less than 6.7 kW/ft.
- All rods that violate the "54/6.3" criterion have a rod burnup of less than 58.5 GWD/MTU.
- In any assembly containing any rods that violate the "54/6.3" criterion there are at least four times as many rods that have total radial peaking factor of less than ¾ of the total radial peaking factor limit of 2.04.

Reactor cores are not operated along the standard analyzed letdown curve. Some deviation from the analyzed 100% power, all rods out analysis is expected to occur. However, existing restrictions on axial offset, load follow, reduced power operations, and operation at power dependent insertion limits maintain the operation of the core reasonably close to the analyzed letdown that forms the basis of this analysis. The use of long window end letdown analyses is considered sufficient to ensure the margins determined in this analysis are adequate.

For the MSLB, fuel handling accident, spent fuel cask drop event, and the control rod ejection event, the guidance of RG 1.183 was followed. The analyses assumed a bounding core isotopic inventory based on a power level of 2650 MWt. This bounds the power level of 2580.6 MWt, which represents the PNP current licensed power level of 2565.4 MWt and 0.5925% measurement uncertainty. For these analyses, the fuel/clad gap fission product release fractions were those of Table 3 of RG 1.183, or as specified in the applicable event-specific Appendix of RG 1.183. For rods that exceed the fuel rod power/burnup criterion in footnote 11 to Table 3 in RG 1.183, the gap release fractions can be increased by a factor of two. The application of a conservative peaking factor of 2.04 to all rods assumed to fail due to cladding failure more than compensates for the factor of two, with margin as ensured by the core design restrictions. This is acceptable since the number of rods that exceed the "54/6.3" criterion in any one assembly would be less than approximately 10% of the rods in any one assembly, and the total number of rods that do not meet the criterion in the entire core would be less than 0.1%. For those rods, the maximum rod average burnup and maximum rod average power are only marginally over the criterion limits as ensured by the core design restrictions. Finally, there would be a sufficiently large number of rods available to compensate for the few rods that do not meet the "54/6.3" criterion as ensured by the core design restrictions. Therefore, the allocation of margins, as discussed above, demonstrates that the submitted analysis remains bounding with respect to the operation of current and future PNP fuel cycles.

The 19 Drawings specifically referenced in the transmittal letter have been processed into ADAMS.

These drawings can be accessed by NRC Staff members within the ADAMS package or by performing a search on the Document/Report Number.