NMC
Committed to Nuclear Excellence

Palisades Nuclear Plant
Operated by Nuclear Management Company, LLC

September 25, 2006

10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

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

Alternative Source Term - Proposed License Amendment

Pursuant to 10 CFR 50.90, Nuclear Management Co. LLC (NMC) requests to amend Facility Operating License DPR-20 for Palisades Nuclear Plant (PNP). NMC proposes to revise the PNP licensing bases to adopt the alternative source term (AST) as described in 10 CFR 50.67 following the guidance provided in Regulatory Guide 1.183. This application includes an amendment to the Technical Specifications, Definition 1.1, Dose Equivalent I-131.

Enclosure 1 provides a description of the proposed changes, supporting justification, the No Significant Hazards Consideration and Environmental Consideration. Enclosure 2 provides marked up copies of the proposed Technical Specification changes. Enclosure 3 provides copies of the retyped TS pages. Enclosure 4 provides the licensing technical report for the alternative source term. A CD is provided as Enclosure 5 containing the supporting engineering calculations and drawings. Enclosure 6 provides the Regulatory Issues Summary RIS 2006-04 resolution matrix. Enclosure 7 provides the RG 1.183 compliance matrix.

In accordance with 10 CFR 50.91(b) (1), a copy of the proposed amendment is being forwarded to the State Designee for the State of Michigan.

Approval of this proposed license amendment is requested within one year of submittal to support NMC planned actions to address Generic Letter 2003-01, Control Room Habitability. NMC requests this amendment to be implemented within 120 days of issuance.

27780 Blue Star Memorial Highway • Covert, Michigan 49043-9530
Telephone: 269.764.2000

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Summary of Commitments

In support of the analysis assumptions that the containment sump water will be maintained with pH > 7.0, NMC previously made a commitment in letter to NRC dated July 7, 2006:

1. NMC will implement an alternate buffer program to achieve a pH of 7.0 – 8.0 post-LOCA with recirculation, during the 2007 fall refueling outage at PNP.

This letter contains two new commitments:

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

   a) Replacement of ECCS pump minimum flow recirculation isolation valves.

   b) Replacement of the control room normal air intake and purge isolation dampers.

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

3. Palisades will conduct post-modification testing, including tracer gas testing, following the implementation of the modifications described above to validate that the modified plant configuration supports the assumptions used in the dose analysis supporting this submittal.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: Sept 25, 2006

Paul A. Harden
Site Vice President, Palisades Nuclear Plant
Nuclear Management Company, LLC

Enclosures: (7)

cc: Administrator, Region III,
USNRC Project Manager, Palisades,
USNRC Resident Inspector, Palisades, USNRC

ENCLOSURE 1

LICENSE AMENDMENT REQUEST
FULL SCOPE ALTERNATE SOURCE TERM
LICENSEE'S EVALUATION

PALISADES NUCLEAR PLANT

ENCLOSURE 1

Regulatory Assessment of the Proposed Implementation of the Alternative Radiological Source Term Methodology

1.0 DESCRIPTION

INTRODUCTION

This submittal is a request to amend Operating License DPR-20 for the Palisades Nuclear Plant. Pursuant to the requirements of 10 CFR 50.90 and 10 CFR 50.67, Nuclear Management Company, LLC (NMC) hereby proposes to amend Appendix A of Facility Operating License, DPR-20, Technical Specifications. This request incorporates a revision to the licensing basis of the Palisades Nuclear Plant (PNP) that supports a full scope application of an Alternative Source Term (AST) methodology. Proposed Technical Specification (TS) changes, which are supported by the AST Design Basis Accident (DBA) radiological consequence analyses, are included in this application for a license amendment.

The current PNP licensing basis for radiological consequence analysis of accidents discussed in Chapter 14 of the Final Safety Analysis Report (FSAR) is based on methodologies and assumptions that are primarily derived from Technical Information Document (TID)-14844 and other early guidance.

The AST methodology will modify PNP's licensing bases by: 1) replacing the current accident source term with an alternative source term as described in 10 CFR 50.67 for DBA radiological consequences, and 2) establishing the 10 CFR 50.67 Total Effective Dose Equivalent (TEDE) dose limits as acceptance criteria for the radiological consequences of DBAs.

Regulatory Guide (RG) 1.183 provides guidance on application of Alternative Source Terms (AST) in revising the accident source terms used in design basis radiological consequences analyses, as allowed by 10CFR50.67. Because of advances made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents, 10CFR50.67 was issued to allow holders of operating licenses to voluntarily revise the traditional accident source term used in the design basis accident (DBA) radiological consequence analyses with alternative source terms (ASTs).

EVALUATION OVERVIEW AND OBJECTIVE

As documented in NEI 99-03 and Generic Letter 2003-01, several nuclear plants performed testing on control room unfiltered air inleakage that demonstrated leakage rates in excess of amounts assumed in the current accident analyses. While the Palisades tracer gas test demonstrated inleakage less than that assumed in the FSAR analyses, the AST methodology as established in RG 1.183 and as supplemented by Regulatory Issues Summary 2006-04, is being used to calculate the offsite and control room radiological consequences for Palisades to support the control room habitability program by establishing a conforming set of radiological analyses.

The following limiting UFSAR Chapter 14 accidents are analyzed:

- Loss-of-Coolant Accident (LOCA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Small Line Break Outside Containment (SLBOC)
- Control Rod Ejection (CRE)
- Fuel Handling Accident (FHA)
- Spent Fuel Cask Drop

2.0 PROPOSED CHANGES

LICENSING BASIS AND TECHNICAL SPECIFICATION CHANGES

Nuclear Management Company, LLC (NMC) proposes to revise the Palisades licensing basis to implement the AST, described in RG 1.183, through reanalysis of the radiological consequences of the UFSAR Chapter 14 accidents listed in Section 1.0 above. As part of the full implementation of this AST, the following changes are incorporated in the analysis:

- The total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10CFR100.11.

- New onsite (Control Room) and offsite atmospheric dispersion factors are developed in accordance with the guidance of RG 1.183.

- Dose conversion factors for inhalation and submersion are from Federal Guidance Reports (FGR) Nos. 11 and 12 respectively.

- Decreased values for limiting control room unfiltered air inleakage are assumed.

Accordingly, the following changes to the Palisades Technical Specifications (TS) are proposed:

- The definition of Dose Equivalent I-131 in Section 1.10 is revised to reference Table 2.1 of Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989, as the source of effective dose conversion factors.

CREDITED PLANT MODIFICATIONS

To support specific assumptions used in the AST radiological consequence analyses, plant modifications are needed. The modifications are described below.

ENCLOSURE 1

Regulatory Assessment of the Proposed Implementation of the Alternative Radiological Source Term Methodology

Background

Palisades has unique control room habitability considerations that are due in large part to the location of the safety injection refueling water tank (SIRWT). The SIRWT is located on the roof directly above the control room and in close proximity to the control room normal intakes. Due to this tank location, backleakage of contaminated sump water to the SIRWT during post accident sump recirculation results in significant direct dose to control room operators. Significant inhalation and immersion doses also result due to the release from SIRWT vent and subsequent intake and inleakage into the control room envelope.

Palisades has identified scenarios in which the ECCS pump suction header could potentially be pressurized to nearly the containment spray pump discharge pressure with a single active failure if there is leakage through an idle Containment Spray pump. This can be caused by failure of one emergency diesel generator to start or remain operating; or the failure of one containment sump isolation valve to open or remain open after the recirculation actuation signal (RAS) is generated.

Either of the above situations leads to an idle containment spray pump train with the discharge check valve subjected to the spray pump discharge header pressure. Due to system configuration, it is not practical to measure the leakage for these check valves and since the leak resistance cannot be characterized, it cannot be credited in design basis safety analyses.

Leakage through High Pressure Safety Injection (HPSI) subcooling and suction line is another potential path to the SIRWT. This can be caused by failure of one emergency diesel generator to start or remain operating after it successfully starts and after the HPSI subcooling valves open; or the failure of one containment sump isolation valve to open or remain open after the recirculation actuation signal (RAS) is generated.

Either of the above situations leads to pressurization of the subcooling line to an idle HPSI pump with the suction check valves subjected to a pressure approaching the spray pump discharge pressure. Due to system configuration, it is not practical to measure the leakage for these check valves and since the leak resistance cannot be characterized, it cannot be credited in design basis safety analyses.

Therefore, the MHA/LOCA with a single active failure as described above could result in the potential for backleakage through the SIRWT discharge isolation valves.

To address potential backleakage to the SIRWT, Palisades will implement three distinct plant modifications:

1. Replacement of ECCS pump minimum flow recirculation isolation valves.

2. Replacement of the control room normal air intake and purge isolation dampers.

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

Replacement of the recirculation isolation valves (1) reduces the backleakage to the SIRWT through the minimum flow recirculation lines that are isolated upon recirculation actuation signal. Replacement of the control room normal air intake and purge isolation dampers (2) reduces the unfiltered inleakage of airborne contamination into the control room envelope. Providing an alternate power source to allow the cross-tie of the ECCS suction trains (3) depressurizes a potentially faulted suction line and eliminates potential backleakage to the SIRWT through the main discharge isolation valves.

Modification 3 involves an operator action to effect the cross-tie and is discussed in detail below.

<u>Credited Manual Operator Action</u>

The Maximum Hypothetical Accident / Loss of Coolant Accident analysis credits a post-accident manual operator action to open two low pressure safety injection (LPSI) suction line valves. The action is needed in the event that one train of emergency core cooling system suction piping becomes pressurized. The action of opening both valves serves to equalize pressure and eliminate the driving head that could potentially force contaminated sump water to the safety injection and refueling water storage tank through the SIRWT main discharge isolation valves.

The power supplies for the valve operators are located directly beneath the control room in the cable spreading room. Access to the cable spreading room is via control room envelope vestibule doors, stairs and other human-access doors and does not traverse any harsh post-accident environments. The cable spreading room itself is not a harsh post-accident environment with respect to temperature, pressure, relative humidity, chemical spray, radiation or submergence. The operator action is assumed to be taken if needed no later than 1 hour and 41 minutes after recirculation mode is entered (2 hours into the event). The action is being incorporated into emergency operating procedure "Loss of Coolant Accident Recovery" as a time-verified/time-critical operator action.

The LPSI cross-tie valves are normally closed during power operation. In the event of an MHA/LOCA with a concurrent loss of offsite power, the emergency diesel generators (EDG) would power both LPSI cross-tie valves motor control centers (MCC) via their associated power trains. If one of the EDGs fails to start or run under these conditions, its associated power train would be deenergized, and only one engineered safeguards features (ESF) train would be available – allowing for the potential pressurization conditions described above. Opening both LPSI cross-tie valves cross-ties the ECCS suction headers and eliminates the driving head for SIRWT backleakage. However, with one train de-energized, operators would be unable to open both valves. Operator action to open both valves post-accident using manual operators is not viable since engineered safeguards feature room entry would be required. Therefore, a modification is being implemented to provide an alternate source of power and controls from the operable train to the inoperable train isolation valve allowing remote opening of valves from the cable spreading room.

The action would only be implemented following a LOCA with recirculation in conjunction with a single active failure as discussed above. Therefore, no changes to the normal plant valve line-up are required.

As per RG 1.183, Section 1.1.2, proposed uses of an alternative source term and the associated proposed facility modifications and changes to procedures are evaluated to determine whether the proposed changes are consistent with the principle that adequate defense in depth is maintained and system redundancy, independence, and diversity are preserved. The manual operator action does not replace an automatic function. The implementation of the AST and the modification/operator action discussed above does not create a need for compensatory reliance on manual operator actions. Rather, the pre-existing need for the termination of a potential post-accident radiological release path in a timely manner is addressed by the modification/operator action. Independent of the implementation of the AST, the modification/operator action reduces potential onsite and offsite radiological consequences.

3.0. BACKGROUND

In the past, power reactor licensees have typically used U.S. Atomic Energy Commission Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962, as the basis for DBA analysis source terms. The power reactor siting regulation, 10 CFR Part 100, Section 11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," which contains offsite dose limits in terms of whole body and thyroid dose, makes reference to TID-14844.

On December 23, 1999, the NRC published 10 CFR 50.67, "Accident source term," in the Federal Register. This regulation provides a mechanism for licensed power reactors to replace the current accident source term used in DBA analyses with an alternative source term. The direction provided in 10 CFR 50.67 is that licensees who seek to revise their current accident source term in design basis radiological consequence analyses shall apply for a license amendment under 10 CFR 50.90.

The guidance of RG 1.183 was used by NMC in preparing AST DBA radiological consequences analyses for PNP. The NRC staff prepared RG 1.183 to provide guidance to licensees of operating power reactors on acceptable applications of alternative source terms; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals.

The criteria of 10 CFR 50.67 were used to evaluate the Current Licensing Basis (CLB) DBAs for radiological consequences and analysis assumptions are consistent with RG 1.183 as presented in Enclosure 7.

4.0 TECHNICAL ANALYSIS

The Technical Analysis supporting this submittal is provided in Enclosure 4, NAI-1149-027, AST Licensing Technical Report.

RADIOLOGICAL EVALUATION METHODOLOGY

Acceptance Criteria

Offsite and Control Room doses must meet the guidelines of RG 1.183 and requirements of 10 CFR 50.67. The acceptance criteria for specific postulated accidents are provided in Table 6 of RG 1.183. For analyzed events not addressed in RG 1.183, the basis used to establish the acceptance criteria for the radiological consequences is provided in the discussion of the event in Enclosure 4. For Palisades, the events not specifically addressed in RG 1.183 are the Small Line Break Outside Containment and the Spent Fuel Cask Drop.

Analysis Input Assumptions

Common analysis input assumptions include those for the control room ventilation system and dose calculation model, direct shine dose, radiation source terms, and atmospheric dispersion factors. Event-specific assumptions are discussed in the event analyses in Enclosure 4. Major items of interest are highlighted below.

Control Room Envelope Unfiltered Air Inleakage

Each accident analyzed, along with the specific inputs and assumptions, is described in Enclosure 4. These analyses provide for a bounding allowable control room unfiltered air inleakage of 10 cfm. The use of 10 cfm as a design basis value will be established to be above the unfiltered inleakage value determined through modification, testing and analysis consistent with the resolution of issues identified in NEI 99-03 and Generic Letter 2003-01. It is noted that two assumptions for the value of control room envelope unfiltered inleakage have been utilized. See Section 1.6.3.1 of Enclosure 4 for additional details.

All unfiltered inleakage is assumed to enter the control room envelope via the normal intakes. Potential locations for control room envelope unfiltered inleakage include normal intake and purge exhaust isolation damper leakage, air handling unit drain leakage, and switchgear and cable spreading room emergency exhaust fan duct leakage. The potential for inleakage is due to the potential for portions of the control room envelope to be at a negative differential pressure with respect to outside air in these locations and conditions. See Section 1.6.3.1 of Enclosure 4 for additional details.

Palisades Design Basis Locked Rotor Event

The limiting Palisades locked rotor event is described in FSAR Section 14.7.2. As per

RG 1.183 (Appendix G, Item 2), since no fuel damage is postulated for the limiting locked rotor event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break outside containment.

The limiting locked rotor event, locked rotor without loss of offsite power (LR without LOOP), was presented for information purposes in the original Palisades FSAR (Consumers Power Company, Palisades Power Plant, Final Safety Analysis Report, Volume 3, Section 14.7). Revision 0 of the Palisades FSAR included analyses of the locked rotor event with concurrent loss of offsite power, but identified the results as part of the systematic evaluation program and as being presented for informational purposes (Letter from R. B. DeWitt (CPCo) to H. Denton (NRC), titled "Docket 50-255 – License DPR-20 – Palisades Plant Final Safety Analysis Report (FSAR) Update Chapter 14, "Safety Analysis," Revision 0", dated December 21, 1984). Revision 8 of the Palisades FSAR eliminated the informational cases and only included LR without LOOP analyses (Letter from D. P. Hoffman (CPCo) to Document Control Desk (NRC), titled "Docket 50-255 – License DPR-20 – Palisades Plant – Revision 8 to the Final Safety Analysis Report (FSAR) Update", dated June 30, 1989).

The licensing basis for only analyzing LR without LOOP is seen to be established during the request for an interim power uprate (Letter from D. P. Hoffman (CPCo) to A. Schwencer (NRC), titled "Docket 50-255, License DPR-20, Palisades Plant – Power Increase Application", dated July 28, 1977). In support of the license amendment request a report (XN-NF-77-18, titled "Plant Transient Analysis of the Palisades Reactor for Operation at 2530 MWt", issued July 18, 1977) was submitted to the NRC that included analysis of the LR without LOOP. The Safety Evaluation (Letter from A. Schwencer (NRC) to D. Bixel (CPCo), titled "Docket No. 50-255", dated November 1, 1977) accepted the LR without LOOP analysis and the uprate was approved.

Palisades was licensed prior to the formalization of the General Design Criteria (GDC) and Standard Review Plan (SRP), to which more recently licensed plants are held. The Systematic Evaluation Program (SEP) reviewed the Palisades then-current licensing basis with respect to updated regulations, guides, SRP (Section 15.3.3), etc. NRC review of SEP Topic XV-7 identified that Palisades licensing basis for LR did not include LOOP (Letter from D. M. Crutchfield (NRC) to D. P. Hoffman (CPCo), titled "RE: SEP

Design Basis Events Request for Additional Information", dated July 1, 1980).

The NRC requested that LR with LOOP be analyzed for the SEP process to assess the adequacy of the Palisades licensing basis with respect to the new standards, i.e., SRP 15.3.3. Results of LR with LOOP were found to be somewhat worse than the licensing basis LR without LOOP event (Letter from R. A. Vincent (CPCo) to D. M. Crutchfield (NRC), titled "Docket 50-255 – License DPR-20 – Palisades Plant – SEP Topics – Design Basis Events", dated September 24, 1981; attaches XN-NF-81-25, "Systematic Evaluation Program Design Basis Events Transient Reanalysis for Palisades Reactor", Revision 1, dated June 1981). As indicated in the final safety assessment report

(NUREG-0820, "Integrated Plant Safety Assessment, Systematic Evaluation Program, Palisades Plant, Final Report", dated October 1982), the final NRC disposition for the deviation in Topic XV-7 (i.e., that LR is analyzed without LOOP) was that plant design meets current criteria or was acceptable on another defined basis. Therefore, the LR without LOOP remains Palisades's current licensing basis.

Fission Product Inventory and Source Terms

The source term data to be used in performing alternative source term (AST) analyses for Palisades are:

- Primary Coolant Source Term
- Secondary Side Source Term (non-LOCA)
- LOCA Source Term
- Fuel Handling Accident Source Term
- Spent Fuel Cask Drop Source Terms

See Section 1.7 of Enclosure 4 for additional details.

Atmospheric Relative Concentrations

New atmospheric relative concentrations (X/Q) have been calculated for the AST analyses. The methods for both onsite and offsite X/Q are purely analytical, with no reliance on site-specific wind tunnel testing. Meteorological data collected by the PNP meteorological monitoring program described in the UFSAR is used in generating the accident atmospheric dispersion (χ/Q) factors. See Section 1.8 of Enclosure 4 for additional details.

Dose Calculations

Analyses consider the radionuclides listed in Table 5 of RG 1.183 and assume that fission products are released to containment in particulate form, except for elemental iodine, organic iodine, and noble gases. Radioiodine fractions released to containment in a postulated accident are assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodine, including both gap releases and fuel pellet releases. In specific instances, transport models may affect radioiodine fractions. See the discussion of each event in Section 2 of Enclosure 4.

Methodologies

The AST analyses performed for PNP use assumptions and models defined in RG 1.183 to provide appropriate and prudent safety margins. Except as otherwise stated, credit is taken for engineered safety features (ESF) and other appropriately qualified, safety-related, accident mitigation features. In some cases, PNP has opted to not take credit for a qualified accident mitigation feature in order to provide an additional measure of conservatism. Enclosure 4 describes these exceptions. Selected numeric input values are conservative to assure a conservative calculated dose. Except as noted or as otherwise required by regulatory guidance, analyses use current licensing basis values.

Analysis Conservatisms

In order to maximize the resulting doses and to provide margin allowance, conservatisms have been used while preparing this submittal. A discussion of these conservatisms is provided for each event in the corresponding section in Enclosure 4.

The areas where conservatisms have been applied are:

- Fission Product Inventory
- Primary Coolant Source Term
- Secondary Side Source Term
- LOCA Source Term
- Fuel Handling Accident Source Term
- Spent Fuel Cask Drop Source Terms
- Onsite and Offsite Atmospheric Relative Concentrations
- All Accidents/Events

  The apparent limited margin for the Main Steam Line Break (MSLB) control room operator dose is mitigated by noting the conservatisms used in the calculation of the main steam line break results.

  The assumption of 2% fuel failures is very conservative considering the thermal-hydraulic analysis of the MSLB demonstrates that no fuel failures would occur. In addition, no credit for the high velocity, buoyant release from the main steam safety valves or atmospheric dump valves is applied to the atmospheric dispersion factors for these releases. This is also a significant conservatism given the close proximity of the Main Steam Safety Valves (MSSV) and Atmospheric Dump Valves (ADV) to the control room normal intakes (source of the control room envelope unfiltered inleakage) and resulting

  high probability that a significant fraction of the release bypasses the normal intakes.

CONCLUSION

Full implementation of the Alternative Source Term methodology, as defined in Regulatory Guide 1.183 and Regulatory Issue Summary 2006-04, into the design basis accident analysis has been made to support control room habitability. Analysis of the dose consequences of the Loss-of-Coolant Accident (LOCA), Fuel Handling Accident (FHA), Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR), Small Line Break Outside Containment (SLBOC), Control Rod Ejection (CRE) Ejection, and Spent Fuel Cask Drop have been made using the RG 1.183 methodology. The analyses used assumptions consistent with proposed changes in the Palisades licensing basis and the calculated doses do not exceed the defined acceptance criteria.

SUMMARY OF RESULTS

Results of the PNP radiological consequence analyses using the AST methodology and the corresponding allowable control room unfiltered air inleakage are summarized in Table 1. Enclosure 4, NAI-1149-027, AST Licensing Technical Report, explains these results and acceptance criteria in more detail. This report supports a maximum allowable control room unfiltered air inleakage of 10 cfm.

TABLE 1
Palisades Summary of Alternative Source Term Analysis Results

Case	Unfiltered CR Inleakage (cfm)	EAB Dose[1] (rem TEDE)	LPZ Dose[2] (rem TEDE)	Control Room Dose[2] (rem TEDE)
LOCA	10	12.76	3.27	4.01
MSLB	10	2.46	0.77	4.98
SGTR Pre-accident Iodine Spike	100	0.99	0.22	3.79
Acceptance Criteria		**≤ 25[3]**	**≤ 25[3]**	**≤ 5[4]**
SGTR Concurrent Iodine Spike	100	1.17	0.21	3.48
Small Line Break Outside of Containment	100	0.41	0.05	0.53
Acceptance Criteria		**≤ 2.5 [3]**	**≤ 2.5 [3]**	**≤ 5 [4]**
FHA in Containment	100	2.20	0.28	4.04
FHA in FHB 10% Release Filtration	100	2.02	0.25	3.68
FHA in FHB 34% Release Filtration	100	1.60	0.20	2.81
FHA in FHB 50% Release Filtration	100	1.31	0.17	2.22
Control Rod Ejection – Containment Release	10	2.70	0.43	1.14
Control Rod Ejection – Secondary Release	10	2.61	0.68	1.14
Spent Fuel Cask Drop 30 days Decay 90% of Release via FHB filtration system	100	2.04	0.25	1.37
Spent Fuel Cask Drop 30 days Decay 82.5% of Release via FHB filtration system	100	2.78	0.35	1.99
Spent Fuel Cask Drop 90 days Decay No Control Room Isolation	100	0.08	0.01	1.67
Acceptance Criteria		**≤ 6.3[3]**	**≤ 6.3[3]**	**≤ 5[4]**

(1) Worst 2-hour dose
(2) Integrated 30-day dose
(3) RG 1.183, Table 6
(4) 10CFR50.67

5.0 REGULATORY SAFETY ANALYSIS

NMC proposes to revise the PNP licensing basis to implement the alternative source term (AST), described in Regulatory Guide (RG) 1.183, through reanalysis of the radiological consequences of the following limiting Updated Final Safety Analysis Report (UFSAR) Chapter 14 accidents:

- Loss-of-Coolant Accident (LOCA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Small Line Break Outside Containment (SLBOC)
- Control Rod Ejection (CRE)
- Fuel Handling Accident (FHA)
- Spent Fuel Cask Drop

As part of the full implementation of this AST, the following changes are assumed in the analysis:

- The total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11.
- New onsite (Control Room) and offsite atmospheric dispersion factors are developed.
- Dose conversion factors for inhalation and submersion are from Federal Guidance Reports (FGR) Nos. 11 and 12 respectively.
- New values for control room unfiltered air inleakage are assumed. The AST methodology as established in RG 1.183 is being used to calculate the offsite and control room radiological consequences for PNP to support the control room habitability program.

The full implementation of the AST is supported by the following Technical Specification changes:

- The definition of Dose Equivalent 1-131 in Section 1.10 is revised to reference Federal Guidance Report 11 (FGR 11), Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, 1989, as the source of thyroid dose conversion factors.

5.1 NO SIGNIFICANT HAZARDS CONSIDERATION

The standards used to arrive at a determination that a request for amendment involves a no significant hazards consideration are included in the Commission's regulation, 10 CFR 50.92, which states that no significant hazards considerations are involved if the operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows:

This change does not involve a significant hazards consideration for the following reasons:

1. The proposed amendment does not involve a significant Increase in the probability or consequences of an accident previously evaluated.

   Response: No

   Alternative source term calculations have been performed for PNP that demonstrate the dose consequences remain below limits specified in NRC Regulatory Guide 1.183 and 10 CFR 50.67. The proposed change does not modify the design or operation of the plant. The use of an AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents and has no direct effect on the probability of any accident. The AST has been utilized in the analysis of the limiting design basis accidents listed above. The results of the analyses, which include the proposed changes to the Technical Specifications, demonstrate that the dose consequences of these limiting events are all within the regulatory limits.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   Response: No

   The proposed change does not affect any plant structures, systems, or components. The proposed operation of plant systems and equipment affected by this change does not create the possibility of a new or different kind of accident previously evaluated. The proposed modifications and post-modification testing are intended to enhance the capability of the plant to comply with the revised post accident dose results presented in this submittal. Since the alternative source term is a revised methodology used to estimate resulting accident doses, it is not an accident initiator.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in the margin of safety.

   Response: No

The proposed implementation of the alternative source term methodology is consistent with NRC Regulatory Guide 1.183. Conservative methodologies, per the guidance of RG 1.183, have been used in performing the accident analyses. The radiological consequences of these accidents are all within the regulatory acceptance criteria associated with use of the alternative source term methodology.

The proposed changes continue to ensure that the doses at the exclusion area and low population zone boundaries and in the control room are within the corresponding regulatory limits of RG 1.183 and 10 CFR 50.67. The margin of safety for the radiological consequences of these accidents is considered to be that provided by meeting the applicable regulatory limits, which are set at or below the 10 CFR 50.67 limits. An acceptable margin of safety is inherent in these limits.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above discussion, NMC has determined that the proposed change does not involve a significant hazards consideration.

5.2 APPLICABLE REGULATORY CRITERIA

RG 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

Enclosure 7 provides a compliance matrix describing how the applicable criteria of RG 1.183 are met for this submittal. There are no exceptions, to any applicable criteria, being proposed.

FSAR Accident Analysis Compliance

The revised Palisades accident analyses addressed in this report follow the guidance provided in RG 1.183. Assumptions and methods utilized in this analysis for which no specific guidance is provided in RG 1.183, but for which a regulatory precedent has been established, are as follows:

- Selection of the Small Line Break Outside Containment dose consequences methodology and acceptance criteria are based on Standard Review Plan Section 15.6.2 and Regulatory Guide 1.183.
- Selection of the Spent Fuel Cask Drop dose consequences methodology and acceptance criteria are based on the Fuel Handling Accident from Regulatory Guide 1.183.

- Use of the MicroShield code to develop direct shine doses to the Control Room. MicroShield is a point kernel integration code used for general purpose gamma shielding analysis. It is qualified for this application and has been used to support licensing submittals that have been accepted by

  the NRC. Precedent for this use of MicroShield is established in the Duane Arnold Energy Center submittal dated October 19, 2000 and associated NRC Safety Evaluation dated July 31, 2001.

- Use of the QADMOD-GP code to develop direct shine dose to the Control Room from the SIRWT. QAD is recommended for determining shielded dose in Standard Review Plan Section 12.3.

- Enclosure 6 contains the Regulatory Issues Summary (RIS) 2006-04 resolution matrix.

Environmental Qualification (EQ)

Section 2.8 of Enclosure 4 discusses the RG 1.183 position on performance of required EQ analyses with respect to AST and TID-14844 source term assumptions. The generic issue identified (Generic Issue 187) has since been resolved. The staff concluded that there is no clear basis for a requirement to modify the design basis for equipment qualification to adopt the AST since there would be no discernible risk reduction associated with such a requirement.

Therefore, NMC is not proposing to modify the equipment qualification design basis to adopt AST. The Palisades EQ analysis will continue to be based on TID-14844 assumptions at this time.

Emergency Plan

An effectiveness review was conducted in accordance with the requirements of 10 CFR 50.54 (q) to assess if the proposed changes represent a decrease in the effectiveness of the Palisades Emergency Plan. This evaluation concluded that the effectiveness of the Emergency Plan is not affected by these changes.

NUREG-0737 Post-Accident Access Shielding and Sampling Capabilities

PNP NUREG-0737 post-accident access basis calculations consider 30-day durations for vital area dose rates. Post-accident mission doses associated with actions defined in the PNP emergency operating procedures are based on estimates of required mission times and area dose rates from the shielding review study. In the resolution of Generic Issue 187, the NRC staff indicates that for exposure to containment atmosphere, the TID-14844 source term and the gap and in-vessel releases in the AST produced similar integrated doses and that for exposure to sump water, the integrated doses calculated with the AST only exceeded those calculated with TID-14844 after 42 days for a PWR. Also, the requirements to have and maintain post-accident sampling systems were eliminated at Palisades by Amendment No. 193. Given the conservative nature of the original NUREG-0737 analyses, there is likely no discernible risk reduction associated with re-constitution of the post-accident access doses to adopt AST.

Therefore, NMC will not modify the Palisades basis documentation for NUREG-0737 Post Accident Shielding responses to adopt AST.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites, 1962.

2. USNRC, Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants, 2000.

3. NEI 99-03, Control Room Habitability Guidance, Nuclear Energy Institute, Revision 0 dated June 2001 and Revision 1 dated March 2003.

4. NAI-1149-027, AST Licensing Technical Report for PNP, Revision 1, Numerical Applications, Inc., September 15, 2006.

5. Federal Guidance Report No. 11 (FGR 11), Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and ingestion, 1989.

6. Federal Guidance Report No. 12 (FGR 12), External Exposure to Radionuclides in Air, Water, and Soil, 1993.

7. NRC Regulatory Issue Summary 2006-04 Experience with Implementation of Alternative Source Terms, 2006.

8. NUREG 0737, Clarification of TMI Action Plan Requirements, 1980.

9. 10 CFR 50.67, Accident Source Term, 1999.

10. NRC Generic Letter 2003-01, Control Room Habitability, 2003.

11. Generic Issue 187 Resolution, 2001.

12. USNRC Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence. Assessments at Nuclear Power Plants," 1982.

ENCLOSURE 2

LICENSE AMENDMENT REQUEST
FULL SCOPE AST

REVISED TECHNICAL SPECIFICATION PAGE 1.1-3
AND
OPERATING LICENSE PAGE CHANGE
INSTRUCTIONS

2 Pages Follow

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Remove the following page of Appendix A Technical Specifications and replace with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
1.1-3	1.1-3

1.1 Definitions

CHANNEL FUNCTIONAL TEST

b. Digital channels - the use of diagnostic programs to test digital hardware and the injection of simulated process data into the channel to verify OPERABILITY, of all devices in the channel required for channel OPERABILITY.

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or control rods within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the plant specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. **The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1989; (Table 2.1, Exposure-to-Dose Conversion Factors for Inhalation).**

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except Primary Coolant Pump seal water leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

ENCLOSURE 3

LICENSE AMENDMENT REQUEST

FULL SCOPE AST

REVISED TECHNICAL SPECIFICATION PAGES

TS Pages

1.1-3

The attached retype reflects the currently issued version of the Technical Specifications. Pending Technical Specification changes or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed retype. The enclosed retype should be checked for continuity with Technical Specifications prior to issuance.

1.1 Definitions

CHANNEL FUNCTIONAL TEST	b. Digital channels - the use of diagnostic programs to test digital hardware and the injection of simulated process data into the channel to verify OPERABILITY, of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
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LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except Primary Coolant Pump seal water leakoff), that is captured and conducted to collection systems or a sump or collecting tank;