March 20, 2006 10 CFR 50.91(a)(5)

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

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

License Amendment Request: Removal of TSP from Palisades Containment

Pursuant to 10 CFR 50.91(a)(5), Nuclear Management Company, LLC (NMC) requests Nuclear Regulatory Commission (NRC) review and approval of a proposed license amendment for the Palisades Nuclear Plant (PNP). NMC proposes to remove tri-sodium phosphate (TSP) from the containment building at PNP as an interim measure until the long term resolution of GSI-191 is implemented. The requirement for TSP in the PNP containment is in Technical Specification (TS) 3.5.5, "Trisodium Phosphate." The proposed TS change would expire upon implementation of an alternate buffer program, which is currently scheduled following the fall 2007 refueling outage.

Information Notice 2005-26, "Results of Chemical Effects Head Loss Tests in a Simulated PWR Sump Pool Environment," provided the initial results of the NRC sponsored head loss testing being performed at Argonne National Laboratory (ANL). The information is relevant to plants containing phosphate and calcium sources that may dissolve within the post loss-of-coolant accident (LOCA) containment pool with sufficient concentrations to form calcium phosphate precipitate. The test results indicate that substantial head loss may occur if sufficient calcium phosphate is produced in a sump pool and transported to a pre-existing fiber bed on the containment sump screen. Consequently, the emergency core cooling system flow and containment spray system flow could be reduced due to the increased head loss across the containment sump screen while in the post-LOCA recirculation phase. PNP uses TSP as a buffering agent to increase the pH of the initially acidic post-LOCA containment water to a more neutral pH. PNP calcium sources include containment concrete and calcium silicate insulation.

As part of this license amendment request, NMC is modifying the licensing basis for control room in-leakage to use the bounding value control room envelope (CRE) in-leakage of 58 scfm, which includes the test measurement uncertainty of +9 scfm. Tracer gas testing was performed at PNP from April 6, 2005 to April 8, 2005. The current Final Safety Analysis Report (FSAR) value is 85 scfm, which does not take into account the tracer test gas results.

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Enclosure 1 provides a detailed description of the proposed change, Background and Technical Analysis, No Significant Hazards Consideration Determination, Applicable Regulatory Criteria and Environmental Review Consideration. Enclosure 2 provides the revised TS page reflecting the proposed change. Enclosure 3 provides the annotated TS page showing the changes proposed. Enclosure 4 provides the engineering analysis that evaluated the impact on the radiological consequences due to the removal of TSP from the PNP containment. Enclosure 5 provides the TS bases changes for information.

NMC requests approval of this proposed license amendment by March 17, 2007. NMC further requests a 60-day implementation period following amendment approval.

A copy of this request has been provided to the designated representative of the State of Michigan.

Summary of Commitments

This letter contains three new commitments and no revisions to existing commitments.

1. NMC will implement a potassium iodide (KI) program for control room personnel at PNP upon approval of the license amendment request. The KI program will be implemented per the guidance provided in NEI 99-03, "Control Room Habitability Assessment Guidance," as an interim compensatory measure until implementation of an alternate buffer program.

2. NMC will inject sodium hydroxide as an alternate buffer within seven days post-LOCA with recirculation at PNP.

3. NMC will submit a license amendment request to implement an alternate buffer program after the Westinghouse Owners Group (WOG) Alternate Buffer Project is concluded.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 20, 2006.

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

Enclosures (5)

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

ENCLOSURE 1
DESCRIPTION OF REQUESTED CHANGES

1.0 DESCRIPTION

Nuclear Management Company, LLC (NMC) requests to amend Operating License DPR-20 for the Palisades Nuclear Plant (PNP). The proposed change would delete Technical Specification (TS) 3.5.5, "Trisodium Phosphate."

2.0 PROPOSED CHANGE

TS 3.5.5 requires tri-sodium phosphate (TSP) baskets in the PNP containment. TS LCO 3.5.5 requires that the baskets contain ≥ 8300 pounds and ≤ 11000 pounds of active TSP. Whenever the TSP is not within limits, Condition A, Action A.1 requires restoration of the TSP within limits within 72 hours.

NMC proposes to delete TS 3.5.5 and the corresponding TS bases. The proposed TS change would expire upon implementation of an alternate buffer program, which is currently scheduled following the fall 2007 refueling outage.

3.0 BACKGROUND

Information Notice 2005-26, "Results of Chemical Effects Head Loss Tests in a Simulated PWR Sump Pool Environment," provided the initial results of the Nuclear Regulatory Commission (NRC) sponsored head loss testing being performed at Argonne National Laboratory (ANL). The information is relevant to plants containing phosphate and calcium sources that may dissolve within the post loss-of-coolant accident (LOCA) containment pool with sufficient concentrations to form calcium phosphate precipitate. The test results indicate that substantial head loss can occur if sufficient calcium phosphate is produced in a sump pool and transported to a preexisting fiber bed on the containment sump screen. Consequently, the emergency core cooling system (ECCS) flow and containment spray system (CSS) flow could be reduced by the increased head loss across the sump screen while in the post-LOCA recirculation phase.

PNP uses TSP as a buffering agent to increase the pH of the initially acidic post-LOCA containment sump water to a more neutral pH. PNP calcium sources include containment concrete and two forms of calcium silicate: pipe insulation and Marinite® fiber board.

<u>Radiological Consequences Methodology</u>

The current licensing basis methodology for radiological consequences is described in the PNP Final Safety Analysis Report (FSAR). The current licensing basis dose methodology is based on Technical Information Document (TID)-14844, with 10 CFR 100 and GDC 19 dose limits.

Current Licensing Basis for Radiological Consequences

The current licensing basis at PNP includes wind tunnel χ/Q. The use of wind tunnel χ/Q for design basis analyses is consistent with the guidance in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," provided certain conditions are met. The wind tunnel χ/Q were developed with a program materially consistent with RG 1.194 conditions. For example, detailed site-specific topology and accepted modeling similarity criteria were used to ensure representative results. Limiting measured χ/Q values were used for conservatism with respect to χ/Q corresponding to the 95th percentile confidence level. The structure of the experimental program, as well as the education and experience of Cermak Peterka Peterson, Inc. (CPP), supports the validity of the results. The wind tunnel testing and data reduction were performed under a quality assurance plan that contains the major elements expected in any acceptable quality assurance program, however, the plan makes no reference to 10 CFR 50, Appendix B.

PNP is committed to submit a methodology change to a full scope implementation of the alternate source term (AST) methodology in July 2006, which will establish a conforming radiological design basis for control room habitability and offsite doses at PNP. Implementation of the full scope AST methodology is currently expected to occur upon start-up from the fall 2007 refueling outage.

4.0 TECHNICAL ANALYSIS

TSP is intended to control post-accident sump pH between 7 and 8. There are four primary issues associated with post-accident sump pH control:

1. Retention of radioiodine in sump water
2. Rates of corrosion
3. Rates of hydrogen generation
4. Equipment Environmental Qualification (EEQ) components

Removal of TSP without the addition of alternative buffering may result in post-accident sump pH values of less than 7. The four primary issues identified above are addressed as follows:

Retention of Radioiodine in Sump Water

The technical evaluation, including assumptions, of the impact on radiological consequences by removing TSP from containment is provided as Enclosure 4.

The bulleted items below are a listing of inputs and assumptions used in Enclosure 4 that differ from the current licensing basis.

- Engineered Safeguards (ESF) Room Iodine Airborne Fraction

The current licensing basis ESF room iodine airborne fraction of 1.718% was not used in Enclosure 4, and instead a value of 10% was assumed.

- Safety Injection Refueling Water Tank (SIRWT) Iodine Volatile Fraction

The current licensing basis SIRWT iodine volatile fraction of $3x10^{-4}$, was not used in Enclosure 4, and instead a value of $4.56x10^{-2}$ was used, based on actual SIRWT water pH and iodine concentrations.

- Elemental Iodine Spray Removal Coefficient

The current licensing basis elemental iodine spray removal coefficient of 19.5/hr was not used in Enclosure 4, and instead, the elemental iodine spray removal coefficient was set to zero to remove credit for elemental iodine spray removal. However, the elemental wall deposition coefficient of 1.3/hour was retained.

- Particulate Iodine Spray Removal Coefficient

The current licensing basis particulate iodine spray removal coefficient of 2.9/hr was not used in Enclosure 4, instead, the particulate iodine spray removal coefficient was reduced to 0.1/hr to remove credit for spray removal and retain credit for natural deposition.

- Control Room Unfiltered In-leakage

The current PNP FSAR value for control room unfiltered in-leakage is 85 scfm. Actual tracer gas testing determined the limiting tracer gas test result as 49 scfm. In Enclosure 4, the control room unfiltered in-leakage was reduced from 85 scfm, to 58 scfm, bounding the results of tracer gas testing. A test measurement uncertainty of +9 scfm was used to determine 58 scfm. Since the inclusion of test uncertainty is not required for low-leakage control room envelopes (i.e. less than 100 scfm in-leakage) the uncertainty is additional margin to accommodate between surveillance degradation.

- Potassium Iodide (KI) Ingestion

Credit was taken for KI ingestion by control room personnel (i.e., a factor of ten reduction in thyroid dose).

Impact of TSP Removal on Radiological Consequences

The removal of TSP from containment is conservatively assumed to result in the loss of the ability to control post-accident sump water pH to between 7 and 8. In an actual large break LOCA event with significant fuel failure, fission products (primarily cesium in the form of cesium hydroxide, cesium borate, and cesium iodide) and core-concrete interaction products (in the event of a vessel breach), are likely to control pH to greater than 7 for time periods on the order of 24 hours in the absence of pH control additives such as TSP. Additionally, regardless of the initial pH or the presence of TSP, the pH of the sump water is expected to rise to about 7, primarily because calcium silicate contains sodium silicate as an impurity. The sodium silicate is very soluble and as it dissolves the sodium (Na), it causes the pH of the initial boric acid/LiOH solution to increase. Furthermore, PNP specific testing, NWT Report 729, "Palisades Calcium Silicate Insulation Behavior Laboratory Evaluations," dated October 2005, indicates that post-accident sump water pH would be increased by the dissolved calcium silicate. However, no credit for these phenomena is assumed in evaluating the impact of TSP removal on the radiological licensing basis. This represents a significant conservatism inherent in the evaluation of the impact of the TSP removal.

The impact on the radiological design bases due to the removal of TSP from containment will be evaluated via the impact on the maximum hypothetical accident (MHA). The impact of assuming loss of sump pH control on the current licensing basis MHA analysis dose contributors is as follows:

Resulting doses from containment leakage activity are increased since the containment atmosphere source term is increased due to the re-evolution of iodine. Resulting dose from containment leakage activity must be recalculated. Whole body dose would be essentially unaffected since whole body dose is due to noble gas releases, which are not impacted by TSP removal.

SIRWT and ESF room leakage doses are potentially increased since the volatile fraction of iodine in the leaked fluid is increased due to the assumed lower pH of the containment sump water, which increases the effective iodine airborne fraction. Resulting doses from SIRWT and ESF room leakage activity must be recalculated.

Dose from SIRWT shine is decreased since the increased volatile fraction of iodine results in less iodine reaching the tank and/or more iodine escaping from the tank, decreasing the shine source term. Dose from SIRWT shine can be conservatively assumed to remain unaffected. Other shine doses are increased, but since these dose contributors comprise only a small fraction of the total shine dose, the increase can be accommodated by the margin in the SIRWT shine dose calculation.

Based on the above physical and methodological impacts, the MHA has been re-analyzed, based on the following physical assumptions:

- Complete and instantaneous re-evolution of iodine removed from the containment atmosphere due to containment sprays.
- No credit being taken for neutral SIRWT back-leakage in determining the SIRWT iodine volatile fraction.
- Containment sump leakage to ESF rooms having iodine airborne fractions of 0.10.

Therefore, with the removal of TSP from containment, the radiological consequences are impacted as follows:

- The off-site thyroid committed does equivalent (CDE) at the exclusion area boundary increases by a factor of 4.33, from 18.05 rem to 78.17 rem. The off-site thyroid dose (CDE) at the low population zone boundary increases by a factor of 3.71, from 9.99 rem to 37.10 rem.

  The 10 CFR 100 limits for off-site thyroid dose are 300 rem, and therefore, the off-site radiological consequences at the exclusion area and low population zone boundaries remain within 10 CFR 100 limits. Whole body deep-dose equivalent (DDE) due to noble gases is unaffected since noble gas releases are not impacted by TSP removal.

- The on-site thyroid dose (CDE) at the control room increases by a factor of 10.02, from 21.26 rem to 213.1 rem, until credit for KI is taken. Once credit for KI is taken, the thyroid dose in the Control Room would be 21.31 rem (factor of 10 reduction). Due to the need to credit KI or self contained breathing apparatus (SCBA), NMC will implement the KI program for Control Room personnel at PNP upon approval of the License Amendment Request. NMC will implement KI per the guidance provided in NEI 99-03, "Control Room Habitability Assessment Guidance," dated June 2001, as endorsed by the NRC in RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," dated May 2003, Section 2.7.3, as an interim compensatory measure until implementation of an alternate buffer program.

  The GDC-19 derived limits for control room thyroid dose (CDE) are 30 rem, and therefore, the on-site radiological consequences at the control room remain within GDC 19 limits, provided compensatory measures such as KI or SCBA are implemented. In addition, a control room unfiltered in-leakage value of 58 scfm was used as an input, based on tracer gas test results. The limiting tracer gas test result was 49 scfm. An uncertainty of +9 scfm was used, resulting in an assumed value of 58 scfm. Whole body DDE due to noble gases is unaffected since noble gas releases are not impacted by TSP removal.

(Whole body equivalent doses at the exclusion area boundary increase from 0.9 to 2.7 rem TEDE; at the low population zone increase from 0.4 to 1.2 rem TEDE; and at the control room increase from 0.7 to 6.5 rem TEDE, before credit for KI is taken, and remain essentially unchanged from 0.7 to 0.7 rem TEDE after credit for KI is taken.)

Rates of Corrosion

NMC performed an evaluation of the structural impact on the susceptible materials that are submerged in sump fluid, as well as materials subject to spray flow. Because the removal of TSP will prolong the time for sump fluid in an acidic condition, a higher material wastage due to corrosion is expected for the structural materials.

Industry data for general corrosion of carbon and low alloy steels in aerated boric acid solutions comes from EPRI TR-1000975, "Boric Acid Corrosion Guidebook, Managing Boric Acid Corrosion Issues at PWR Power Stations," Revision 1, and WCAP-7099, "Absorption of Corrosion Hydrogen by 302B Steel at 70°F to 500°F," dated 1967. This data lists the corrosion rate of carbon steel at 140°F in an aerated solution, with 2500 ppm boron (pH=4.96), as 15 mpy (0.015 inch/year). Corrosion rates at 70°F are 2 mpy. Corrosion rates at 100°F are 7 mpy. The average containment sump temperature at PNP over the first 7 days after the LOCA is less than 180°F. Therefore, the corrosion of any carbon steel or low alloy steel exposed to containment spray for the first 7 days will be insignificant.

With regards to stress corrosion cracking of austenitic stainless steel, Standard Review Plan (SRP) 6.1.1, paragraph II B.1.a, states that the composition of containment spray and core cooling water should be controlled to ensure a minimum pH of 7.0, as given in Branch Technical Position MTEB 6-1, "pH for Emergency Coolant Water for PWRs," dated July 1981. MTEB 6-1 bases the 7.0 pH criteria on a report from Westinghouse, WCAP -7798-L, "Behavior of Austenitic Stainless Steel in Post Hypothetical Loss of Coolant Environment," dated November 1971, and an Oak Ridge report ORNL-TM-2412, Part X, "Design Considerations of Reactor Containment Spray Systems - Part X, The Stress Corrosion Cracking of Types 304 and 316 Stainless Steel in Boric Acid Solutions," May 1971, which is reference 2 of MTEB 6-1.

The Oak Ridge document reported the results of testing of various samples (stressed, non-stressed, sensitized, non-sensitized, etc) of 304 and 316 stainless steel in a mockup containment spray model at a pH of 4.5, and at various chloride concentrations. The test sprayed the specimens at 285°F for 1 day, then at 212°F for 7 days, then submerged the samples for 2 months at 165°F. The Oak Ridge testing had many cases of stress corrosion cracking with a pH of 4.5 and greater than 5 ppm chloride ion. The report concluded that stress corrosion cracking is more likely at higher temperatures, higher chloride ion concentrations, lower pH, higher stress levels, and higher levels of sensitization.

The Oak Ridge report stated, "Qualitatively it was found that most of the cracking occurred during the first 8 days of the test when the temperatures were highest." Although not specifically concluded from the testing, the report stated that, "Cracking of the austenitic stainless steels is always possible when the temperature of the solution is ambient or above and chloride ions and oxygen are present." However, the testing at Oak Ridge was more severe than actual expected conditions, in that the MTEB 6-1 stated that, "Considering the fact that in U-bend specimens, the material was sensitized, stressed beyond yield, and plastically deformed, we conclude that the reported test conditions were much more severe than the stress conditions likely to exist in the post accident emergency coolant systems."

PNP was designed and licensed before the NRC issued the Standard Review Plans (SRP) and Regulatory Guides to specify design methods acceptable to the NRC for compliance to the GDC. PNP also participated in the Systematic Evaluation Program (SEP) for plants that received construction permits before the final GDCs. The satisfactory participation in the SEP and the responses to the GDCs as stated in the FSAR, do not include adherence to SRP 6.1.1, as part of the licensing basis. However, NMC is considering the technical information provided in SRP 6.1.1 for PNP.

Accounting for the removal of TSP and considering the results of the Oak Ridge report, NMC has determined that adding sodium hydroxide (NaOH) to the post accident containment sump within 7 days from the initiation of a LOCA with recirculation is necessary. Performing this task within 7 days allows containment pressure to decrease to a value such that the NaOH can be injected through the containment spray system, and gives the plant operators sufficient time to align the required systems. The seven days is also a reasonable time interval relative to the Oak Ridge testing for developing cracks associated with stress corrosion cracking of stainless components because the average containment sump temperature at PNP for the first seven days is less than 180°F. Therefore, NMC will inject NaOH as an alternate buffer within seven days post-LOCA with recirculation at PNP. The methods and procedure to accomplish this task will be completed as part of the implementation of the license amendment.

Rates of Hydrogen Generation

During a severe core damage accident, hydrogen is generated as a result of core heat-up and oxidation of zircaloy clad material. For the severe accidents described in the PNP Individual Plant Evaluation (IPE) submittal, the typical core damage scenario results in approximately 30% of the clad material oxidized during the core heat-up phase occurring over a time period of 1 to 2 hours. This compares to the 1% clad oxidation that had been considered in the plant specific design basis MHA.

The IPE performed an extensive review of hydrogen containment challenges ranging from direct containment heating (DCH - particularized fuel dispersal in containment) to late challenges due to spray recovery. Hydrogen generated by the oxidation of zirconium and steel in the reactor vessel prior to vessel failure was considered the only early flammable gas source. Other potential early hydrogen sources, such as oxidation of aluminum and organic coatings in the containment due to the operation of the containment sprays, radiolytic decomposition of water, etc., were small in comparison to the in-vessel source predicted for severe accident sequences. This additional hydrogen from non in-vessel sources is on the order of or less than the magnitude of the uncertainty in predicting the in-vessel source for a severe accident; therefore, it was considered acceptable to neglect these additional hydrogen sources. These analyses employing the same conservative hydrogen assumptions were again repeated in the PNP License Renewal Application, dated March 28, 2005, severe accident mitigation alternative (SAMA) analysis and the results showed that containment challenges from early, intermediate and late hydrogen burns were very small.

Subsequent to the detailed analysis of hydrogen events in the IPE, a series of experiments were conducted to determine hydrogen combustion behavior under conditions of rapidly condensing steam. The experiments were conducted in the Surtsey facility under near prototypic severe accident conditions that would be expected in a Combustion Engineering (CE) System 80+ containment. Mixtures were initially nonflammable owing to steam dilution. The mixtures were ignited by glow plugs when they became flammable after sufficient steam was removed by condensation caused by water sprays. For example, in the CE design the well-mixed hydrogen concentration was approximately 13.6% (dry measurement) assuming 100% clad oxidation. If this hydrogen were to accumulate in the region above the operating deck, the average concentration would be about 19%. No detonations or accelerated flame propagation were observed. The observed combustion was characterized by multiple deflagrations with relatively small pressure rises.

In summary, the IPE performed a rigorous assessment of hydrogen as a result of severe accidents. Subsequent experimental results demonstrated that the variety of ways that hydrogen generation from a severe accident that could challenge containment as described in the submittal was conservatively overstated. Therefore, the removal of TSP from containment will have a negligible impact with respect to design basis hydrogen challenges to containment integrity. The NRC safety evaluation dated January 11, 2005, approving the removal of the hydrogen recombiners at PNP corroborates this conclusion.

EEQ Components

Environmental Qualification (EQ) of all Class 1E components for PNP have been qualified to the requirements of 10 CFR 50.49 (k), the DOR Guidelines, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," dated November 1979. The EQ components in containment needed after recirculation were identified and evaluated for the effects of an acidic containment spray. DOR Guidelines allow environmental qualification by analysis. These components were evaluated for an exposure to containment spray with a pH of 4.5 for an initial seven days. The age sensitive materials (non-metallic parts) covered under the requirements of 10 CFR 50.49, and the metallic parts of these EQ components (such as metal housing of 10 CFR 50.49 components) were evaluated for this spray condition. Evaluation was based on utilizing and analyzing available industry and technical research data. This included chemical resistance of materials, effect of aging on materials that will be exposed to spray, and the duration of highly acidic and alkaline spray. The analysis concluded that all components analyzed will be able to perform the safety function in accordance with the requirements of 10 CFR 50.49, under a chemical spray of 4.5 pH to 8.0 pH, for a period of seven days without loss of function. Therefore, NMC will inject sodium hydroxide as an alternate buffer within seven days post-LOCA with recirculation at PNP.

Alternate Buffer Injection

The corrosion and EQ evaluations justified seven days without a loss of function in the absence of a buffering agent. For this reason, within seven days, in the event of a LOCA with recirculation, NaOH will be injected as the alternative buffer agent via the CSS. The NaOH will be injected in a dose accessible location. The non-safety related system for NaOH injection will be designed to meet the necessary system requirements. Operating procedures are being developed to ensure an adequate amount of NaOH could be delivered into the containment sump within seven days after a LOCA event. The methods and procedure to accomplish this task will be completed as part of the implementation of the license amendment.

The long term resolution of finding an alternative buffer agent to replace the TSP is in progress. NMC is supporting the Westinghouse Owners Group's (WOG) program of identifying alternative sump buffering agents. NMC will submit a license amendment request to implement an alternate buffer program after the WOG Alternate Buffer Project is concluded.

Risk Evaluation

To evaluate the risk from containment sump blockage as a result of the potential for formation of calcium phosphate precipitate, failure of recirculation due to sump screen blockage was considered a failure mode and the potential consequences were assessed qualitatively. In considering that recirculation could fail due to containment sump blockage given a large break LOCA, the core damage frequency would increase as the screen failure probability would increase. Subsequent containment mitigating function analyses would show an increase in the frequency of failure as well. For example, containment spray would not be available while in the post-LOCA recirculation mode. With this information, the containment event trees which model the various phenomena that could threaten containment integrity may also exhibit greater failure frequencies. This would result in an increase in the estimated fission product releases and subsequent consequential health effects. Therefore, eliminating the potential credible 'chemical effect' sump screen failure mode would present an increase in the margin of safety for PNP.

With respect to small break LOCAs, past analyses cited time to recirculation values ranging from about 45 minutes to 130 minutes for break diameter sizes of 1-inch, 2-inch and 4-inch respectively, while addressing a variety of initial conditions including different engineered safeguards systems (ESS) pump combinations. The operators could be forced to aggressively cool down the plant in order to achieve shutdown cooling entry conditions given the low likelihood of recirculation failure due to the postulated 'chemical effect' that could occur even for a small break LOCA. The FSAR Chapter 14 limiting small break LOCA is 0.08 ft^2, which is about a 3.8 inch diameter break. Also, successful shutdown cooling operation is necessary to preclude recirculation operation given a small break LOCA. Given that any LOCA could consequentially fail the shutdown cooling system (assuming the break is close to the shutdown-primary coolant system tee connections) eliminating the potential 'chemical effect' failure mode would present an increase in the margin of safety for PNP.

Based on the above, it is concluded that the removal of TSP from the PNP containment results in a net increase in plant safety.

5.0 REGULATORY SAFETY ANALYSIS

5.1 <u>No Significant Hazards Consideration</u>

Nuclear Management Company, LLC (NMC) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed amendment does not involve a significant increase in the probability of an accident previously evaluated because tri-sodium phosphate (TSP) is not considered to be an initiator of any analyzed accident. The TSP in containment is designed to buffer the acids expected to be produced after a loss of coolant accident and is credited in the radiological analysis for iodine retention. The proposed change does not impact any failure modes that could lead to an accident.

The proposed amendment does not involve a significant increase in the consequences of an accident previously evaluated because the removal of TSP from containment results in the radiological consequences remaining under 10 CFR 100 limits and GDC-19 limits (with credit taken for potassium Iodide (KI)).

The off-site thyroid committed does equivalent (CDE) at the exclusion area boundary increases by a factor of 4.33, from 18.05 rem to 78.17 rem. The off-site thyroid dose at the low population zone boundary increases by a factor of 3.71, from 9.99 rem to 37.10 rem. The 10 CFR 100 limits for off-site thyroid dose are 300 rem, and therefore, the off-site radiological consequences at the exclusion area boundary and low population zone boundaries remain within 10 CFR 100 limits. Whole body deep-dose equivalent (DDE) due to noble gases is unaffected since noble gas releases are not impacted by TSP removal.

The on-site thyroid dose (CDE) at the control room increases by a factor of 10.02, from 21.26 rem to 213.1 rem, until credit for KI is taken. Once credit for KI is taken, the thyroid dose in the control room would be 21.31 rem (a factor of 10 reduction). Due to the need to credit KI or self-contained breathing apparatus (SCBA), Nuclear Management Company (NMC) will implement a KI program for control room personnel at Palisades Nuclear Plant (PNP) upon approval of the license amendment request. NMC will implement a KI program per the guidance provided in NEI 99-03, "Control Room Habitability Assessment Guidance," dated June 2001, as endorsed by the NRC in RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," Section 2.7.3, as an interim compensatory measure until final resolution of an alternate buffer is achieved and subsequently implemented.

The GDC-19 derived limits for control room thyroid dose are 30 rem, and therefore, the on-site radiological consequences at the control room remain within GDC 19 limits, provided credit for compensatory measures such as KI or SCBA are implemented. In addition, a control room unfiltered in-leakage value of 58 scfm was used as an input, based on actual tracer gas test results. The control room unfiltered in-leakage was reduced to 58 scfm, bounding the results of tracer gas testing. The limiting tracer gas test result was 49 scfm. A test measurement uncertainty of +9 scfm was used to determine 58 scfm. Since the inclusion of test uncertainty is not required for low-leakage control room envelopes (i.e. less than 100 scfm in-leakage) the uncertainty is additional margin to accommodate between surveillance degradation. The current

PNP FSAR value for in-leakage is 85 scfm. Whole body DDE due to noble gases are unaffected since noble gas releases are not impacted by TSP removal.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   TSP is a passive component which is used at PNP as a buffering agent to increase the pH of the initially acidic post-loss of coolant accident (LOCA) containment water to a more neutral pH. The removal of TSP from containment will mitigate the potential consequences resulting from the postulated chemical effects of TSP causing failure of accident mitigating systems to perform their function during the post-LOCA recirculation phase.

   The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. Removal of TSP from the PNP containment does not constitute an accident initiator or create a new or different kind of accident previously analyzed. The proposed amendment does not involve operation of any required systems, structures or components (SSCs) in a manner or configuration different from those previously recognized or evaluated. No new failure mechanisms will be introduced by the changes being requested. The removal of TSP and the potential chemical effects on SSCs (i.e. safety injection and containment spray) would enhance the ability of these systems to perform their post-accident mitigating functions.

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

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed amendment does not involve a significant reduction in a margin of safety. The removal of TSP from the PNP containment results in a net increase in plant safety. Containment sump blockage as a result of the potential for formation of calcium phosphate precipitate may be a credible failure mode and therefore, removing the potential 'chemical effect' sump screen failure mode given a large break LOCA is risk positive. In addition, the requirement to rapidly cool down to shutdown cooling conditions given the low likelihood of recirculation failure due to the postulated 'chemical effect' that exists even for a small break LOCA, and given the low likelihood that the LOCA break location could also fail shutdown cooling, or that the shutdown cooling system could be unavailable due to random failures, removing TSP would result in an increase in the margin of safety for PNP.

Therefore, removing TSP to reduce the risk of recirculation failure due to sump screen blockage would present an increase in the overall margin of safety for PNP.

Based on the evaluation above, NMC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 100

100.11 - Determination of exclusion area, low population zone, and population center distance.

(a) As an aid in evaluating a proposed site, an applicant should assume a fission produce release from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:

(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

As described in Section 4, following the removal of TSP from the PNP containment, post-LOCA off-site radiological consequences at the exclusion area and low population zone boundaries remain within 10 CFR 100 limits.

10 CFR 50 Appendix A GDC-19

Criterion 19 - Control room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

As described in Section 4, following the removal of TSP from the PNP containment, post-LOCA on-site radiological consequences at the control room remain within GDC 19 limits provided compensatory measures such as KI or SCBA are taken.

10 CFR 50.36

Section 182a of the Atomic Energy Act (the Act) requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The licensee provides TSs in order to maintain the operational capability of structures, systems, and components that are required to protect the health and safety of the public. The staff of the Nuclear Regulatory Commission (NRC) used the regulatory requirements for TS changes set forth in 10 CFR 50.36 for this evaluation. Specifically, 10 CFR 50.36(c)(1) specifies safety limits, limiting safety systems settings and control settings, 10 CFR 50.36(c)(2) specifies the requirements for limiting conditions for operation, 10 CFR 50.36(c)(3) specifies the surveillance requirements, 10 CFR 50.36(c)(4) specifies the design requirements, and 10 CFR 50.36(c)(5) specifies the administrative controls.

The NRC provided guidance for the specific contents of the TSs in Final Policy Statement, 58 FR 39133, "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," dated July 22, 1993. The Final Policy Statement established four criteria for determining the items required for inclusion in the TSs:

Criterion 1- Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2- A process variable, design feature, or operating restriction that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of, or presents a challenge to, the integrity of the fission product barrier.

Criterion 3- A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of, or presents a challenge to, the integrity of the fission product barrier.

Criterion 4- A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

These criteria have been codified in 10 CFR 50.36(c)(2)(ii). See Final Rule, "Technical Specifications," 60 FR 36953 (July 19, 1995). As a result, TS requirements which fall within or satisfy any of the criteria in the Final Policy Statement must be retained in the TS.

The PNP applicable Safety Analyses as described in TS Bases Section B3.5.5, specifies that the LOCA radiological consequences analysis takes credit for iodine retention in the sump solution based on the recirculation water pH being above 7.

Thus, PNP considers Technical Specifications related to post-LOCA containment sump pH control as required to be retained (TS Bases page B 3.5.5-2). As such, this request for removal of TS 3.5.5, "Trisodium Phosphate," is considered an interim compensatory measure until implementation of an alternate buffer program.

<u>10 CFR 50.46</u>

(b)(3) Maximum hydrogen generation

"The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding surrounding the plenum volume, were to react."

The PNP IPE performed a rigorous assessment of hydrogen as a result of severe accidents. Subsequent experimental results demonstrated that the variety of ways that hydrogen generation could challenge containment as described in the submittal was conservatively overstated. Therefore, the removal of TSP from containment would have a negligible impact with respect to design basis hydrogen challenges to containment integrity

In order to maintain compliance with these criteria, NMC will inject sodium hydroxide as an alternate buffer within seven days post-LOCA with recirculation at PNP.

10 CFR 50.49

As described in Section 4, environmentally qualified components were analyzed and the evaluation concluded that all components analyzed will be able to perform the safety function in accordance with the requirements of 10 CFR 50.49, under a chemical spray of 4.5 pH to

8.0 pH for a period of seven days without loss of function. Therefore, NMC will inject sodium hydroxide as an alternate buffer within seven days post-LOCA with recirculation at PNP.

In conclusion, based on the considerations described 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

NMC 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.

ENCLOSURE 2

LICENSE AMENDMENT REQUEST: REMOVAL OF TSP FROM PALISADES CONTAINMENT

REVISED TECHNICAL SPECIFICATION PAGE
3.5.5-1
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
3.5.5-1	3.5.5-1

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 (Not Used)

ENCLOSURE 3

LICENSE AMENDMENT REQUEST: REMOVAL OF TSP FROM PALISADES CONTAINMENT

MARK-UP OF TECHNICAL SPECIFICATION PAGE
3.5.5-1
(showing proposed changes)
(additions are highlighted; deletions are strikethrough)

1 Page Follows

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 ~~Trisodium Phosphate (TSP)~~ (Not Used)

~~LCO 3.5.5~~ ~~The TSP baskets shall contain ≥ 8,300 lbs and ≤ 11,000 lbs of active TSP.~~

~~APPLICABILITY:~~ ~~MODES 1, 2, and 3.~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. TSP not within limits.~~	~~A.1 Restore TSP to within limits.~~	~~72 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~	~~6 hours~~
	~~AND~~	
	~~B.2 Be in MODE 4.~~	~~30 hours~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.5.5.1~~	~~Verify the TSP baskets contain ≥ 8,300 lbs and ≤ 11,000 lbs of TSP.~~	~~18 months~~
~~SR 3.5.5.2~~	~~Verify that a sample from the TSP baskets provides adequate pH adjustment of borated water.~~	~~18 months~~

ENCLOSURE 4

LICENSE AMENDMENT REQUEST: REMOVAL OF TSP FROM PALISADES CONTAINMENT

EA-EC-976-01, "IMPACT ON RADIOLOGICAL CONSEQUENCES DUE TO REMOVAL OF TRISODIUM PHOSPHATE FROM CONTAINMENT" DATED 3/2/06

49 Pages Follow

NMC Committed to Nuclear Excellence	EA-EC-976-01	Revision: 0
	Date: 03/02/2006	
	Total Number of Pages: 49	
Title: **Impact on Radiological Consequences Due to Removal of Trisodium Phosphate from Containment**		
Approval: See signature page		

Objective

This calculation conservatively determines the impact on the Palisades radiological design bases given the removal of all trisodium phosphate (TSP) from containment. Post-accident on-site and off-site radiological consequences are re-evaluated and compared to design basis limits derived from Title 10, Code of Federal Regulations, Part 100 (10-CFR-100) and Appendix A, General Design Criterion 19 (GDC 19).

This calculation supports engineering change EC-976, which is being implemented to remove TSP from containment in order to address the potential for sump blockage due to chemical reaction products in post-accident sump water containing TSP and calcium-silicate (Cal-Sil) insulation.

Conclusions

With the removal of all TSP from containment:

- Off-site radiological consequences at the exclusion area and low population zone boundaries remain within 10-CFR-100 limits with no deviations from the current licensing basis.
- On-site radiological consequences at the control room remain within GDC 19 limits with no deviations from the current licensing basis, provided credit for compensatory measures such as potassium iodide (KI) or self-contained breathing apparatus (SCBA) is taken.

Table of Contents

1.0 PURPOSE

This calculation conservatively determines the impact on the Palisades radiological design bases given the removal of all trisodium phosphate from containment. Post-accident on-site and off-site radiological consequences are re-evaluated and compared to design basis limits derived from Title 10, Code of Federal Regulations, Part 100 (Ref. [1]) and Appendix A, General Design Criterion 19 (Ref. [2]).

This calculation supports engineering change EC-976 (Ref. [3]), which is being implemented to remove TSP from containment in order to address the potential for sump blockage due to chemical reaction products in post-accident sump water containing TSP and calcium-silicate (Cal-Sil) insulation.

2.0 METHODOLOGY

2.1 Background

This calculation evaluates the impact to the radiological design basis of the removal of all tri-sodium phosphate (TSP) baskets currently present on the 590' level of containment (Ref. [4], Section 6.4; Ref. [5], LCO 3.5.5). The purpose of TSP in containment is to control post-accident sump fluid pH to within the range 7-8. Ensuring post-accident sump pH between 7 and 8 assures that hydrogen generation, corrosion, insulation and debris dissolution, and radio-iodine retention are all acceptable and consistent with licensing basis assumptions. TSP removal is being pursued to mitigate the potential for containment sump screen blockage during the post-accident recirculation mode. By removing the TSP, reaction with dissolved calcium-silicate (Cal-Sil) insulation in the post-accident environment and subsequent precipitation is precluded. With the elimination of TSP/Cal-Sil precipitate, the predicted magnitude of sump screen blockage is reduced. This alleviates a specific NRC concern resulting from the integrated chemical effects testing performed in September 2005 (Ref. [6]). Note that since a significant amount or essentially all of the TSP is predicted to react and precipitate out of solution in the post-accident environment, the current ability to control pH requires evaluation (see Section 2.2).

2.2 Radiological Consequence Methods

The current licensing basis methodology for radiological consequences is described in the Palisades Final Safety Analysis Report (FSAR, Ref. [4]). Palisades is committed to submit a methodology change to a full-scope implementation of the alternate source term (AST) methodology in July 2006 (Ref. [7]). Given the current licensing basis and the typical review times for AST submittals, it is appropriate to utilize the current licensing basis methodology to evaluate the TSP removal.

The removal of all TSP from containment is assumed to result in the complete loss of the ability to control post-accident sump water pH to between 7-8. In an actual large break LOCA event with significant fuel failure, fission products (primarily cesium in the form of cesium hydroxide, cesium borate, and cesium iodide) and core-concrete interaction products are likely to control pH to pH>7 for time periods on the order of 24 hours in the absence of pH control additives such as TSP (Ref. [8]). Additionally, regardless of the initial pH or the presence of TSP, the pH of the sump water is expected to rise to about 7, primarily because Cal–Sil contains sodium silicate as an impurity (Ref. [9]). The sodium silicate is very soluble and as it dissolves the dissolved sodium (Na) causes the pH of the initial boric acid/LiOH solution to increase. Moreover, Palisades specific testing indicates that post-accident sump water pH would be increased by the dissolved Cal-Sil (Ref. [10]). However, no credit for these phenomenon will be assumed in evaluating the impact of TSP removal on the radiological design basis.

Potentially impacted radiological analyses are the core barrel failure (FSAR: Ref. [4], Section 14.5), steam line rupture incident (FSAR: Ref. [4], Section 14.14), control rod ejection (FSAR: Ref. [4], Section 14.16), loss of coolant accident (FSAR: Ref. [4], Section 14.17), and maximum hypothetical accident (FSAR: Ref. [4], Section 14.22). Radiological consequences for the core barrel failure are bounded by the control rod ejection. Radiological consequences for the steam line rupture incident are based on steam line break outside containment and are therefore not affected by sump pH. In-containment radiological consequences for control rod ejection include a case that does not credit containment spray removal, which bounds the impact of the loss of pH control, and secondary side releases are postulated outside containment and are therefore not affected by sump pH. Radiological consequences for the loss of coolant accident are bounded by maximum hypothetical accident (MHA). Therefore, the impact on the radiological design bases due to the removal of TSP from containment will be evaluated via the impact on the MHA.

For onsite consequences (control room habitability) of the MHA, contributions to dose from containment leakage, SIRWT leakage, ESF room leakage, SIRWT shine, and other shine (i.e., submersion shine, containment shine, cloud shine) are explicitly considered in the current licensing basis. For offsite consequences (exclusion area boundary and low population zone) of the MHA, contributions to dose from containment leakage, SIRWT leakage, and ESF room leakage are explicitly considered in the current licensing basis.

The impact of assuming a complete loss of sump pH control on the current licensing basis MHA analysis dose contributors is as follows:

Containment leakage doses are increased since containment atmosphere source term is increased due to the re-evolved iodine. Containment leakage dose must be recalculated.

SIRWT and ESF room leakage doses are potentially increased since the volatile fraction of iodine in the leaked fluid is increased due to the assumed lower pH of the containment sump water, which increases the effective iodine airborne fraction. These doses are characterized as potentially increased since increased iodine re-evolution from the sump water could result in lower iodine concentrations in the leaked fluid, depending on the timing of the re-evolution process. That is, if significant quantities of iodine are re-evolved from sump water prior to leakage, the available source term is reduced. For this analysis, the non-mechanistic assumption of 50% of the total core iodine inventory deposited instantaneously to the containment sump and available for release is retained and re-evolution is conservatively assumed to impact fluid only after it is leaked. The impact, as described below, is through the elimination of the justification for airborne fractions of less than 10% for ESF room leakage and a change in the justification for airborne fractions less than 10% for SIRWT leakage (through a change to the iodine volatile fraction in the SIRWT). Based on the conservative artifact of the methodology as described above, SIRWT and ESF room leakage doses must be recalculated.

SIRWT shine doses are decreased since the increased volatile fraction of iodine results in less iodine reaching the tank and/or more iodine escaping from the tank, decreasing the shine source term. SIRWT shine dose can be conservatively assumed to remain unaffected. Other shine doses are increased, but since these dose contributors comprise only a very small fraction of the total shine dose, the increase is clearly negligible (0.4 rem whole body total due to all non-SIRWT shine compared to 2.25 rem whole body SIRWT shine: Ref. [11], Input 3.20 and Sheet 34) and can be accommodated by the margin in the SIRWT shine dose calculation (including conservatively ignoring the decrease in SIRWT dose due to TSP removal).

Loss of sump pH control impacts the containment atmosphere source term chemical form distribution, since re-evolved iodine is preferentially elemental. However, since the non-spray related iodine removal coefficients retained in the calculation (i.e., wall deposition and natural/settling deposition) preferentially remove elemental iodine, it is conservative to assume the loss of pH control does not impact iodine chemical form in the containment atmosphere. Therefore, utilizing the current licensing basis chemical form of iodine addresses the impact of TSP removal on chemical form.

Loss of sump pH control impacts iodine scrubbing and retention in pools. Re-evolution of iodine limits the effectiveness of the containment sprays. Therefore, utilizing elemental and particulate spray removal coefficients of 0/hr addresses the impact of TSP removal on iodine scrubbing and retention in pools.

Loss of sump pH control impacts iodine airborne fraction of leaked sump water. The amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history. If sump pH control is lost and sump pH remains appreciably below 7, aqueous iodine will convert to elemental iodine and re-evolve. Re-evolution will occur in containment and could occur after leakage outside of containment. Re-evolution in containment will reduce the iodine concentration available for release via leaked sump fluid, whereas re-evolution outside of containment could potentially increase the airborne fraction of the remaining iodine in the leaked sump water. Therefore, utilizing non-mechanistic leaked sump water iodine concentration and utilizing leakage iodine airborne fractions of 10% for ESF room leakage and appropriately justified airborne fractions for SIRWT leakage addresses the impact of TSP removal on iodine airborne fraction of leaked sump water.

Based on the above physical and methodological impacts, the MHA will be re-analyzed, based on the following physical assumptions:

- Loss of sump pH control results in complete and instantaneous re-evolution of iodine removed from the containment atmosphere due to sprays.
- Loss of sump pH control results in no credit being taken for neutral SIRWT back-leakage in determining the SIRWT iodine volatile fraction.
- Loss of sump pH control results in sump leakage to ESF rooms having iodine airborne fractions of 0.10.

The assumption of loss of post-accident sump pH control can be implemented in the following input assumptions:

- Spray removal coefficients eliminated, non-spray removal reduced to 0.1/hr particulate, 1.3/hr elemental wall deposition.
- SIRWT iodine volatile fraction of 0.453.
- ESF room iodine airborne fraction of 0.10.

2.3 Current Licensing Basis Potential Non-Conformances

Following issuance of regulatory guidance on appropriate methods for radiological consequence analyses (Refs. [12]-[15]), several potential non-conformances have been identified (see below). Characterization of these items as *potential* non-conformances indicates that the underlying assumptions are either not consistent with the current regulatory requirements (Refs. [12]-[15]), not consistent with accepted industry interpretation of current regulatory requirements, or are consistent with the current regulatory requirements but have recently met with difficulty in being approved by the NRC staff for design basis analyses. Management and resolution of the issue is being tracked in the corrective action process (Ref. [16]). Submittal of a full-scope AST implementation and completion of plant modifications will establish a clearly conforming radiological design basis for control room habitability and offsite doses at Palisades and resolve any potential non-conformances.

For this evaluation, the potential non-conforming assumptions will be retained and justified or made conforming as described below.

2.3.1 Wind Tunnel Based Atmospheric Relative Concentrations (Wind Tunnel X/Q)

The current licensing basis control room normal and emergency intake atmospheric relative concentrations are based on data derived from site-specific wind tunnel testing (Ref. [17]), referred to herein as wind tunnel X/Q. The use of wind tunnel X/Q for design basis analyses is consistent with the guidance in RG 1.194 (Ref. [12]) provided certain conditions are met. The conditions require that data based on wind tunnel tests should be accompanied with an evaluation of the representativeness of the experiment results to the particular plant configuration and site meteorological regimes. Also, the conditions define an acceptable experimental program as meeting the following standards:

- The experimental program should be appropriately structured so as to provide data of appropriate quantity and quality to support data analysis and conclusions drawn from that data. The program should be developed by personnel who have educational and work experience credentials in air dispersion meteorology and modeling.
- The experimental program should encompass a sufficient range of meteorological conditions applicable to the particular site so as to ensure that the data obtained address the site-specific meteorological regimes and the site-specific release point/receptor configurations that impact the control room X/Q values. Meteorological conditions observed at the particular site with a frequency of 5 percent or greater in a year should be addressed. Parameters derived from statistical analyses on the experimental data should represent the 95th-percentile confidence level.
- The experimental program, including data reduction and analysis, should incorporate applicable quality control criteria of Appendix B to 10 CFR Part 50. The products of the experimental program should be verified and validated.

The wind tunnel X/Q were developed with a program materially consistent with these conditions. For example, detailed site-specific topology and accepted modeling similarity criteria were used to ensure representative results. Limiting measured X/Q values were used for conservatism with respect to X/Q values corresponding to the 95th percentile confidence level. The structure of the experimental program, as well as the education and experience of Cermak Peterka Peterson, Inc. (CPP), supports the validity of the results. The potential non-conformance is with respect to the quality assurance program. Although the wind tunnel testing and data reduction was performed under a quality assurance plan that contains the major elements expected in any acceptable quality assurance program, the plan makes no reference to 10-CFR-50, Appendix B. Based on the discussion above, there is a high degree of confidence that the wind tunnel X/Q are appropriately representative and conservative for design basis calculations.

Therefore, the current licensing basis wind tunnel X/Q will be utilized for this analysis.

2.3.2 Time Dependent Control Room Operator Breathing Rates

The use of the time dependent breathing rates for control room operator doses is not in compliance with RG 1.195 (Ref. [13], Section 4.2.6). However, the use of time dependent breathing rates for control room operator doses at Palisades has been communicated to the NRC in the past with no specific objections raised. In addition, the occupancy factors used in the analysis contain significant conservatism, based on staffing and shift relief expected during events requiring emergency response organization activation. Conservatisms in control room occupancy factors help mitigate the non-conformances in the control room breathing rates.

Therefore, the licensing basis time dependent control room operator breathing rates will be utilized in this analysis.

2.3.3 ESF Room Iodine Airborne Fraction

The use of ESF room leakage airborne fractions of less than 10%, based on bounding calculated flashing fractions is consistent with RG 1.195 (Ref. [13], Appendix A, Section 4.4) for design basis analyses, provided it is justified based on actual sump pH history and area ventilation rates. Since this analysis assumes complete loss of sump pH control, justification is conservatively ignored.

Therefore, the licensing basis ESF room iodine airborne fraction of 1.718% will not be utilized in this calculation, and instead a value of 10% will conservatively be assumed.

2.3.4 SIRWT Iodine Volatile Fraction

The use of SIRWT leakage airborne fractions of less than 10%, based on bounding calculated flashing fractions is consistent with RG 1.195 (Ref. [13], Appendix A, Section 4.4) for design basis analyses, provided it is justified based on actual sump pH history and area ventilation rates. The use of a very low SIRWT iodine volatilization fraction (which results in an effective SIRWT leakage iodine airborne fraction of far less than 10%) based on neutral sump water is not consistent with industry practice that considers the effect of the pH of the water remaining in the SIRWT on iodine re-volatization. Hence, the justification for the use of an iodine airborne fraction of less than 10% is potentially non-conforming. Furthermore, since this analysis assumes complete loss of sump pH control, justification must also address non-neutral SIRWT backleakage due to the TSP removal.

Therefore, the very low licensing basis SIRWT leakage iodine airborne fraction based on a SIRWT iodine volatile fraction of $3x10^{-4}$, will not be utilized in this calculation, and instead a value that is based on actual SIRWT water pH and iodine concentrations will conservatively be assumed.

2.3.5 Iodine Containment Atmosphere Release Fraction

The use of an iodine release fraction of 25% for the iodine released to the containment atmosphere in conjunction with a model for wall deposition that is time-dependent is not consistent with RG 1.195 (Ref. [13], Section 3.2, Footnote 6). The guidance notes that revision 2 to SRP 6.5.2 (Ref. [18]; which is the guidance used for the current licensing basis analysis) is in error, that a release fraction of 50% should be used if a time-dependent model for iodine wall deposition is used, and that release fractions of 25% are appropriate if no other credit for wall deposition is taken. However, a release fraction of 25% in addition to a time-dependent wall deposition model is credited in the current licensing basis analysis.

Therefore, the use of the current licensing basis iodine containment atmosphere release fraction of 25% is acceptable and will be utilized.

2.4 Software Codes

2.4.1 MHACALC

MHACALC will be used for modeling containment, ESF room, and SIRWT releases and for off-site dose calculations. MHACALC is an in-house developed software code used for the current licensing basis calculations, and has extensive benchmarking, verification and validation documentation (Ref. [19]).

2.4.2 CONDOSE

CONDOSE will be used for modeling control room doses. CONDOSE is an in-house developed software code used for the current licensing basis calculations, and has extensive benchmarking, verification and validation documentation (Ref. [20]).

3.0 ACCEPTANCE CRITERIA

Palisades current licensing basis for radiological consequence analysis is located in Chapter 14 of the FSAR. The dose consequence portion of this calculation utilizes a TID source term methodology and therefore the dose limits of 10-CFR-100.11 and 10-CFR-50-Appendix A, GDC 19 are applicable:

1) An individual located at any point on the boundary of the exclusion area for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

2) An individual located at any point on the outer boundary of the low population zone who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

3) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. SRP 6.4 (Ref. [30]) interprets the limits of GCD 19 of being 5 rem whole body, 30 rem thyroid, and 30 rem skin dose. Note that skin dose is not part of the current licensing basis and is not specifically calculated.

4.0 INPUTS

4.1 Breathing rates for offsite and onsite doses:

$3.47E-4$ m^3/sec from 0 to 8 hours

$1.75E-4$ m^3/sec from 8 to 24 hours

$2.32E-4$ m^3/sec from 1 to 30 days.

Source: Values are the current licensing basis value in the FSAR (Ref. [4], Tables 14.22-2 and 14.24-1).

4.2 Rated core thermal power and calorimetric uncertainty:

2580.6 MW_{th}

Source: FSAR (Ref. [4], Table 14.22-2). Note this is consistent with the measurement uncertainty recapture uprate power level of 2565.4 MW_{th} with 0.5925% calorimetric uncertainty.

4.3 Containment design leak rate:

0.1 % containment air weight/day

Source: FSAR and TS (Ref. [4], Section 1.2; Ref. [5], ADMIN 5.5.14)

4.4 Containment net free air volume:

1.64E+6 ft^3

Source: FSAR (Ref. [4], Section 5.8)

4.5 Site boundary atmospheric dispersion factor:

1.78E-4 sec/m^3 from 0 to 2 hours

Source: FSAR (Ref. [4], Section 2.5). Note offsite dispersion factors were calculated with the PAVAN code.

4.6 Low population zone atmospheric dispersion factors:

1.19E-5 sec/m^3 from 0 to 8 hours

7.68E-6 sec/m^3 from 8 to 24 hours

2.95E-6 sec/m^3 from 1 to 4 days

7.47E-7 sec/m^3 from 4 to 30 days

Source: FSAR (Ref. [4], Section 2.5). Note offsite dispersion factors were calculated with the PAVAN code.

4.7 Minimum sump volume after recirculation:

41,498.2 ft^3

Source: FSAR (Ref. [4], Table 14.22-2). Note the volume is based on the sump mass 2,439,636 at RAS.

4.8 Maximum ECCS leakage into ESF rooms following recirculation:

0.2 gpm

Source: FSAR and TS (Ref. [4], Table 14-22; Ref. [5], ADMIN 5.5.2).

4.9 ESF room iodine decontamination factor:

2

Source: Value is the current licensing basis value in the FSAR (Ref. [4], Table 14.22-2) and has been accepted by the NRC (Ref. [34]). The ventilation in the ESF rooms automatically isolates upon detection of high radiation. The factor is used to account for plateout of iodine onto surfaces in the safeguards rooms since the ventilation rates would be very low after isolation even assuming gross leakage of the dampers.

4.10 Total SIRWT volume:

40,151.1 ft^3

Source: FSAR (Ref. [4], Table 14.22-2).

4.11 SIRWT liquid volume following recirculation:

554.0 ft^3

Source: FSAR (Ref. [4], Table 14.22-2).

4.12 Maximum ECCS leakage into SIRWT following recirculation:

2.2 gpm

Source: FSAR and procedures (Ref. [4], Table 14.22-2; Refs. [21]-[22]).

4.13 Wall deposition removal coefficient for elemental iodine:

1.3 hr^{-1}

Source: Value is current licensing basis value for wall deposition (Ref. [27]).

4.14 Minimum time to recirculation:

19 minutes

Source: Value is current licensing basis value for minimum time to RAS (Ref. [11], Section 4.11).

4.15 Normalized activity source terms:

See Table 4.1

Source: Values listed in Table 4.1 for each of the radio-nuclides of interest are the current licensing basis source terms from Reference [11]. The values are derived from the ORIGEN data in NEDO-24782 (Ref. [23]).

4.16 Radioactive decay constants:

See Table 4.1

Source: Values listed in Table 4.1 for each of the radio-nuclides of interest are those used in the current licensing basis from Reference [11]. The values are derived from the data in NUREG/CR-1413 (Ref. [24]).

4.17 Dose and dose rate conversion factors:

See Table 4.2

Source: Values listed in Table 4.2 for each of the radio-nuclides of interest are those used in the current licensing basis from Reference [11]. The values are derived from the data in ICRP-30 (Ref. [25]). Conversion factors for noble gas isotopes are listed in Table 4.1 for a semi-infinite cloud and in Table 4.2 corrected for a 1000 m^3 room. The use of the noble gas dose rate conversion factors corrected for a 1000 m^3 room is justified by the calculated control room air volume (see Input 4.19).

Table 4.1: Isotope Parameters

Nuclide	S_i (Ci/MW_t)	λ_i (min^{-1})	DRCF (semi-infinite cloud) (Rem/sec)/(Ci/m^3)	
			Thyroid	Whole Body
Kr-83m	2.998E+03	6.313E-03	0.000E-00	3.649E-06
Kr-85m	6.498E+03	2.579E-03	3.083E-02	3.031E-02
Kr-85	2.999E+02	1.230E-07	0.000E-00	4.738E-04
Kr-87	1.155E+04	9.084E-03	1.439E-01	1.447E-01
Kr-88	1.690E+04	4.068E-03	3.803E-01	3.690E-01
Kr-89	1.993E+04	2.194E-01	0.000E-00	0.000E-00
Xe131m	1.760E+02	4.065E-05	0.000E-00	1.324E-03
Xe133m	1.954E+03	2.198E-04	0.000E-00	5.375E-03
Xe-133	5.648E+04	9.177E-05	7.297E-03	6.259E-03
Xe135m	1.698E+04	4.513E-02	0.000E-00	7.647E-02
Xe-135	9.781E+03	1.268E-03	0.000E-00	4.676E-02
Xe-137	4.705E+04	1.810E-01	0.000E-00	0.000E-00
Xe-138	4.433E+04	4.906E-02	1.953E-01	1.969E-01
I-131	2.938E+04	5.987E-05	N/A	N/A
I-132	4.160E+04	5.023E-03	N/A	N/A
I-133	4.808E+04	5.554E-04	N/A	N/A
I-134	6.218E+04	1.318E-02	N/A	N/A
I-135	4.022E+04	1.748E-03	N/A	N/A

Table 4.2: Dose and Dose Rate Conversion Factors

Nuclide	Thyroid	Lungs	B. Surface	B. Marrow	Skin	Eye Lens	Wh. Bdy.
	Submersion Dose Rate Conversion Factors Corrected for a 1000 m^3 room (Rem/sec)/(Ci/m^3)						
Kr-83m	0.000E-00	0.000E-00	6.475E-06	5.653E-6	1.747E-04	1.747E-04	3.649E-06
Kr-85m	1.233E-03	1.131E-03	1.953E-03	1.850E-03	5.139E-02	1.542E-03	1.269E-03
Kr-85	0.000E-00	2.056E-05	3.083E-05	2.878E-05	4.728E-02	3.906E-05	2.314E-5
Kr-87	5.550E-03	5.756E-03	6.886E-03	6.269E-03	3.392E-01	1.007E-01	5.684E-03
Kr-88	1.439E-02	1.336E-02	1.542E-02	1.336E-02	9.456E-02	2.878E-02	1.402E-02
Kr-89	0.000E-00	0.000E-00	0.000E-00	0.000E-00	0.000E-00	0.000E-00	0.000E-00
Xe-131m	0.000E-00	1.233E-04	3.597E-04	3.289E-04	1.542E-02	5.344E-04	1.915E-04
Xe-133m	0.000E-00	2.775E-04	6.167E-04	5.653E-04	3.083E-02	7.503E-04	3.823E-04
Xe-133	4.317E-04	2.672E-04	6.886E-04	6.269E-04	1.131E-02	6.989E-04	3.361E-04
Xe-135m	0.000E-00	3.392E-03	4.522E-03	4.214E-03	2.672E-02	4.522E-03	3.618E-03
Xe-135	0.000E-00	1.850E-03	2.981E-03	2.775E-03	6.578E-02	2.467E-03	2.086E-03
Xe-137	0.000E-00	0.000E-00	0.000E-00	0.000E-00	0.000E-00	0.000E-00	0.000E-00
Xe-138	7.811E-03	8.119E-03	9.558E-03	8.730E-03	1.644E-01	3.186E-02	7.801E-03
	Inhalation Dose Conversion Factors (Rem/Ci-inhaled)						
I-131	1.073E+06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.256E+04
I-132	6.290E+03	9.990E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.367E+02
I-133	1.813E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	5.550E+03
I-134	1.073E+03	5.180E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.106E+02
I-135	3.145E+04	1.628E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.121E+03

4.18 Chemical form of iodine released:

91% elemental

4% organic

5% particulate

Source: FSAR (Ref. [4], Table 14.22-2).

4.19 Net free air volume of the control room:

35,923 ft3 (1017 m^3)

Source: Value is the current licensing basis volume from Reference [11]. This volume conservatively ignores the volume of the air space in the drop ceiling for the control room area and the volume of the mechanical equipment room (MER) which is also maintained at a positive pressure by the control room HVAC system. A lower volume is conservative since it increases the concentrations of radio-nuclides in the control room following an accident.

4.20 Total delay time for emergency HVAC operation:

1.5 minutes

Source: Value is the current licensing basis volume from Reference [26]. The time represents total time for pressurizing the control room to greater than 0.125" water gauge, including time to reach CHP/CHR setpoint (2.5 seconds), signal delay (1 second), delay time for start of diesel generator (10 seconds), delays associated with the DBA sequencer (55.3 seconds), fan motor acceleration time (1.4 seconds), and time for re-pressurization of the control room envelope (7.5 seconds).

4.21 Efficiency of the control room HVAC HEPA/charcoal filters:

99% for iodines

0% for noble gases

Source: FSAR (Ref. [4], Table 14.24-1).

4.22 Control room occupancy factors:

1.0 from 0 to 24 hours

0.6 from 24 to 96 hours

0.4 from 96 to 720 hours

Source: FSAR (Ref. [4], Table 14.24-1). The values are consistent with RG 1.195 (Ref. [13], Section 4.2.6).

4.23 Control room atmospheric dispersion factors

See Table 4.3

Source: FSAR (Ref. [4], Table 14.24-1).

Table 4.3: Control Room Intake Atmospheric Dispersion Factors

Time Interval	Containment/ESF Room Releases		SIRWT Releases	
	Normal Intake	Emergency Intake	Normal Intake	Emergency Intake
0 - 8 Hrs	7.72E-04 s/m3	2.56E-04 s/m3	1.32E-02 s/m3	6.35E-04 s/m3
8 - 24 Hrs	4.55E-04 s/m3	1.51E-04 s/m3	7.78E-03 s/m3	3.74E-04 s/m3
1 - 4 Days	2.90E-04 s/m3	9.60E-05 s/m3	4.95E-03 s/m3	2.38E-04 s/m3
4 - 30 Days	1.27E-04 s/m3	4.22E-05 s/m3	2.18E-03 s/m3	1.05E-04 s/m3

4.24 Control room ventilation rates:

Emergency Mode Filtered Make-up Flow	= 1413.6 cfm
Emergency Mode Filtered Recirculation Flow	= 1413.6 cfm
Emergency Mode Unfiltered In-leakage Flow	= 58.0 cfm
Normal Mode Fresh Air Make-up Flow	= 660.0 cfm
Base Infiltration Leak Rate (Depressurized)	= 384.2 cfm

Source: FSAR and tracer gas test results (Ref. [4], Table 14.24-1; Ref. [28]). Note that the unfiltered in-leakage value bounds the results of the most recent tracer gas test and includes the test measurements uncertainty of 9 scfm. Since the inclusion of test uncertainty is not required for low-leakage control room envelopes (i.e., less than 100 scfm in-leakage, Ref. [15], Section 1.4), the uncertainty can be construed as additional margin to accommodate between surveillance degradation.

4.25 Whole body dose from containment shine:

400 mrem whole body

Source: Value is the current licensing basis value (Ref. [11], Input 3.20).

4.26 Whole body dose from SIRWT shine

2.25 rem whole body

Source: Value is the current licensing basis value (Ref. [11], Sheet 34).

5.0 REFERENCES

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[2] Title 10, Code of Federal Regulations, Part 50 – Domestic Licensing of Production and Utilization Facilities, Appendix A – General Design Criteria for Nuclear Power Plants, Criterion 19 – Control Room.

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[5] Technical Specifications for the Palisades Nuclear Power Plant, Amendment #221.

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[18] NUREG-0800, USNRC Standard Review Plan, Section 6.5.2 Rev 2, Containment Spray as a Fission Product Cleanup System, December 1988.

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[20] EA-A-NL-92-012-01, Revision 0, Benchmarking of the CONDOSE Code for Control Room Habitability Calculations, March 1992. (Cart/Frame: F059/1625)

[21] RO-119, Revision 9, Inservice Testing of Engineered Safeguards Valves CV-3027 and CV-3056, Palisades Nuclear Plant Technical Specification Surveillance Procedure, December 2002.

[22] RT-71L, Revision 12, Technical Specification ADMIN 5.5.2 Pressure Test of ESS Pump Suction Piping, Palisades Nuclear Plant Technical Specification Surveillance Procedure, September 2005.

[23] NEDO-24782, BWR Owner's Group NUREG-0578 Implementation: Analysis and Positions for Plant Unique Submittals, General Electric, August 1980.

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[32] ADMIN 1.16, Revision 1, Respiratory Protection Program, Palisades Nuclear Plant Administrative Procedure, March 2005.

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6.0 ASSUMPTIONS

6.1 The removal of TSP results in the complete loss of post-accident sump pH control.

Basis: This is a conservative assumption since complete loss of sump pH control results in worst-case radiological consequences and sump pH is not expected to be completely uncontrolled even in the absence of TSP. See Section 2.2.

6.2 The complete loss of post-accident sump pH control results in complete, instantaneous re-evolution of the iodine scrubbed from the atmosphere by containment sprays. The instantaneous re-evolution can be modeled by setting all spray removal coefficients to zero, eliminating radiological credit for containment sprays.

Basis: This is a conservative assumption since 100% re-evolution is not expected to occur and a time delay would exist between scrubbing and re-evolution that would reduce the initial atmosphere iodine concentrations.

6.3 The pH of the SIRWT is assumed to be 4.5.

Basis: The design basis low pH of the SIRWT is 4.5 (Ref. [33], Attachment 1). Sump pH is expected to approach 7 (See Section 2.2). The assumption of a constant SIRWT pH of 4.5 conservatively ignores any beneficial increase in pH due to the backleakage.

6.4 Control room operators will take appropriate and timely compensatory measures, such as ingesting potassium iodide (KI) or donning self-contained breathing apparatus (SCBA) such that inhalation (thyroid) doses are reduced by a factor of 10.

Basis: Part of the engineering change modification package (Ref. [3]) will be to implement an appropriate compensatory measures program for control room personnel in compliance with NEI 99-03 guidance (Ref. [31]). With a program in place, a factor of 10 reduction in thyroid dose is consistent with the guidance for KI protection (Ref. [31], Appendix F, Section 3) and a factor of 100-10,000 reduction in inhalation dose is consistent with the guidance for SCBA protection (Ref. [31], Appendix F, Section 2). Note: Palisades maintains positive-pressure SCBA which have a protection factor of 10,000 (Ref. [32]).

6.5 Fraction of total core inventory released:

25% iodines to containment atmosphere

50% iodines to containment sump

100% noble gases to containment atmosphere

Basis: These release fractions are the current licensing basis assumptions (Ref. [4], Table 14.22-2). The noble gas to atmosphere and iodine to sump release fractions are consistent with RG 1.195 (Ref. [13], Table 1 and Appendix A, Section 4.1). The iodine to atmosphere release fraction of 25% for the iodine released to the containment atmosphere, if used in conjunction with a model for wall deposition by containment sprays that is time-dependent, is not consistent with RG 1.195 (Ref. [13], Section 3.2, Footnote 6). However, a release fraction of 25% in addition to a time-dependent wall deposition model is credited in the current licensing basis analysis. Therefore, the use of an iodine containment atmosphere release fraction of 25% is acceptable and will be assumed.

6.6 Sedimentation removal coefficient for particulate iodine:

$0.1\ hr^{-1}$

Basis: This value is assumed based on an IDCORE report (Industry Degraded Core Rulemaking Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," Atomic Industrial Forum, December 1983), which supports higher coefficient values. This analysis conservatively assumes a coefficient of only 0.1/hr, consistent with several prior approved submittals. Specifically, the Safety Evaluation for Kewaunee Nuclear Power Plant Amendment No. 166 to DPR-43 dated March 17, 2003 (ADAMS Accession No. ML030210062) concludes that NRC staff finds 0.1 per hour aerosol removal rate to be reasonable based on NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor containment." The Palisades analyzed thermal power (2580.6 MWt) is greater than the Kewaunee Nuclear Power Plant analyzed thermal power (1851.3 MWt), from the above Kewaunee SER. Since the value of the coefficient generally increases with thermal power, the rate of 0.1/hr is more conservative for Palisades than for Kewaunee.

6.7 Iodine airborne fractions:

0.10 for ESF room leakage

0.0456 for SIRWT leakage

Basis: Flashing fractions for ESF leakage are expected to be well below 10% and SIRWT leakage is not expected to flash given the long transit leakage for this path must take (Ref. [11]). Based on this, lower fractions can be justified based on sump pH history and area ventilation rates (Ref. [13], Appendix A, Section 4.4).

ESF room leakage will be conservatively assumed to have an iodine airborne fraction of 10%. Much lower values are likely justifiable based on past and recent analyses (1.718% from Ref. [11], Sheet 22; 2.953% from Ref. [29], Page 24).

The SIRWT leakage iodine airborne fraction is based on equations {12} and {15} of NUREG/CR-5950 (Ref. [8]). Note that the partition coefficient and the volatile fraction are combined to give the airborne fraction and in MHACALC this result is implemented by setting the SIRWT partition coefficient to unity and the volatile fraction to the airborne fraction.

Based on a conservative value of 180°F for the SIRWT temperature (Ref. [11], Sheet 22), a partition coefficient of 9.94 is obtained (FSAR value is 10: Ref. [4],

Table 14.22-2). Note that more detailed analyses indicate SIRWT temperature does not exceed 120°F (Ref. [29], Table 6-4). Based on a conservative SIRWT sump iodine concentration of 6.72E-5 g-atoms/L (Ref. [11], Sheet 24) and a pH of 4.5 (Assumption 6.3), a volatile fraction of 0.453 is obtained. This results in a an airborne fraction of 0.0456 (See Attachment 1).

6.8 All ECCS leakage will be increased by a factor of 2.

Basis: This conservative assumption is consistent with RG 1.195 (Ref. [13], Appendix A, Section 4.2).

6.9 All assumptions inherent in the methodology of the MHACALC code as documented in Reference [19] apply to this analysis, except for specific inputs as indicated above.

Basis: The current licensing basis software is acceptable for this application.

6.10 All assumptions inherent in the methodology of the CONDOSE code as documented in Reference [20] apply to this analysis, except for specific inputs as indicated above.

Basis: The current licensing basis software is acceptable for this application.

6.11 All assumptions inherent in the current licensing basis MHA calculations (Refs. [11] and [26]) apply to this analysis, except for specific inputs as indicated above.

Basis: The current licensing basis analysis is the starting point for this analysis.

6.12 The core is assumed to operate at 100.5925% of rated power, or 2580.6 MWth.

Basis: This is the current licensed rate thermal power and calorimetric uncertainty.

6.13 The event begins with instantaneous release of the source term to the containment building atmosphere and sump.

Basis: This conservative assumption is consistent with RG 1.195 (Ref. [13], Appendix A, Section 2.1).

6.14 Loss of offsite power occurs coincident with the automatic switch to control room emergency HVAC operating mode.

Basis: This conservative assumption is more limiting than the design basis assumption of loss of offsite power (LOOP) occurring coincident with the event (Ref. [11], Assumption 4.8) and is consistent with RG 1.195 (Ref. [13], Section 5.1.2).

The timing of the LOOP does not impact off-site doses since the offsite release scenario bounds and does not depend on the operation of mitigating systems that are impacted by LOOP timing. However, LOOP timing impacts the delay in achieving control emergency HVAC mode and impacts non-emergency intake since the base infiltration rate (384.2 scfm) that occurs during LOOP prior to EDG start and loading is different from the assumed normal mode intake (660 scfm).

Since early conditions impact dose results more significantly, two LOOP scenarios were investigated: (1) LOOP concurrent with initiating of emergency CR HVAC, and (2) LOOP coincident with RAS (start of ESF room and SIRWT leakage). Of these, scenario (1) has been determined to be limiting through sensitivity studies.

6.15 Time dependent control room ventilation parameters:

Table 6.1: Time Dependent Control Room ventilation Parameters

Time Interval [minutes]	Unfiltered Intake [scfm]	Filtered Intake [scfm]	Filtered Recirculation [scfm]
0.0 – 1.5	660	0	0
1.5 – 3.0	384.2	0	0
3.0 – 43200.0	58	1413.6	1413.6

Basis: Assumption 6.14 discusses the limiting LOOP scenario. Based on Inputs 4.20 and 4.24, and on the current licensing basis assumption that without a LOOP, a delay of 1.5 minutes is appropriate for control room emergency HVAC mode operation (Ref. [26], Section 6.4, Subsection 4, Sheet 24), the above table can be constructed.

6.16 No additional multiplication factor due to thermal cycling is applied to the iodine release from the SIRWT.

Basis: Recent detailed calculations of sump SIRWT back leakage to the SIRWT demonstrate that diurnal thermal cycling does not result in increased total iodine released from the SIRWT (Ref. [29] – See review comment 15).

6.17 The source of radionuclide ingress into the control room when the control room is depressurized is through the normal air intakes.

Basis: The normal intake location are the most likely source of unfiltered inleakage and is the most conservative point on the control room envelope with respect to SIRWT leakage.

6.18 Following SRP 6.4 (Ref. [30], Page 6.4-8), the base infiltration rate of air into the control room when depressurized is assumed to be one-half the leakage from the control room when pressurized to 1/8" water gauge. A contribution from opening and closing doors does not need to be accounted for since vestibules have been installed on the entrances (Ref. [30], Page 6.4-9).

Basis: This is a current licensing basis assumption (Ref. [11]) and is consistent with regulatory guidance.

6.19 Shine doses are assumed not to be impacted by the removal of TSP.

Basis: Shine dose from the SIRWT will decrease for two reasons: (1) re-volatization in containment reduces the radionuclide concentration in the sump water leaked to SIRWT, and (2) re-volatization in SIRWT reduces the radionuclide concentration in the SIRWT water itself. The current licensing basis calculation (Ref. [11]) does not credit the reduction in sump concentration, but does credit reduction in SIRWT concentration. However, the additional reduction in SIRWT concentration due to TSP removal is not credited in the current licensing basis. Therefore, SIRWT shine dose can be conservatively assumed not be impacted by the removal of the TSP.

Other shine doses are increased, but since these dose contributors comprise only a very small fraction of the total shine dose, the increase is clearly negligible (0.4 rem whole body total due to all non-SIRWT shine compared to 2.25 rem whole body SIRWT shine: Ref. [11], Input 3.20 and Sheet 34). Furthermore, this small absolute increase can be accommodated by the margin in the SIRWT shine dose result, including the margin inherent in ignoring the decrease in SIRWT dose due to TSP removal as described above.

7.0 ANALYSIS

The above inputs and assumptions are used to generate the MHACALC and CONDOSE input files (See Attachments 2 and 3). For convenience, key parameters of the analysis are given below.

7.1 MHACALC Key Analysis Parameters

Table 7.1: MHACALC Key Parameters	
Reactor Thermal Power	2580.6 MW_{th}
Source Term	
Core Activity	See Table 4.1
Nuclide Parameters	See Table 4.1
Activity Release Timing	
Gap Release	Instantaneous
Fuel Release	Instantaneous
Activity Release from the Fuel	
Noble Gases	100%
Iodines	75% total (25% atmosphere, 50% sump)
Iodine Chemical Form in Containment	91% elem., 4% organic and 5% part.
Iodine Chemical Form Released from ECCS Leakage	97% elem., 3% organic[1]
Spray Operation	No credit for spray removal
Removal Coefficients	
Elemental Iodine Spray Removal	0.0 hr^{-1}
Particulate Iodine Spray Removal	0.0 hr^{-1}
Elemental Iodine Wall Deposition Removal	1.3 hr^{-1}
Particulate Iodine Sedimentation Removal	0.1 hr^{-1}
Containment Net-Free Volume	1.64E+6 ft^3
Containment Sump Volume	41,498.2 ft^3
Total SIRWT Volume	40,151.1 ft^3
SIRWT Water Volume	554.0 ft^3
Containment Leakage Rates	
0 – 24 hours	0.10 weight %/day
> 24 hours	0.05 weight %/day
Time to RAS (ECCS Leakage Begins)	19 minutes
Leakage to ESF Room	0.2 gpm x 2

Table 7.1: MHACALC Key Parameters

Leakage to SIRWT	2.2 gpm x 2
Iodine Airborne Fraction for ESF Room Leakage	10%
Iodine Airborne Fraction for SIRWT Leakage	4.56%
Plate-out Fraction for ESF Room Leakage	2
Offsite Breathing Rates	
0 - 8 hours	3.47E-4 m^3/sec
8 - 24 hours	1.75E-4 m^3/sec
1 - 4 days	2.23E-4 m^3/sec
4 - 30 days	2.32E-4 m^3/sec
Atmospheric Relative Concentrations	
Exclusion Area Boundary	
0 – 2 hours	1.78E-4 sec/m^3
Low Population Zone	
0 – 8 hours	1.19E-5 sec/m^3
8 - 24 hours	7.68E-6 sec/m^3
1 – 4 days	2.95E-6 sec/m^3
4 - 30 days	7.47E-7 sec/m^3
Dose and Dose Rate Conversion Factors	See Table 4.2

[1] Note: chemical for ECCS releases do not affect results since transport and filtration are the same for all forms after ECCS release.

7.2 CONDOSE Key Analysis Parameters

Table 7.2: CONDOSE Key Parameters

Control Room Volume	35,923 ft^3
Control Room Breathing Rates	
0 - 8 hours	3.47E-4 m^3/sec
8 - 24 hours	1.75E-4 m^3/sec
1 - 4 days	2.23E-4 m^3/sec
4 - 30 days	2.32E-4 m^3/sec
Control Room Occupancy Factors	
0 - 24 hours	1.0
1 - 4 days	0.6
4 - 30 days	0.4
Atmospheric Relative Concentrations	
Containment/ESF Rooms to Normal Intake	
0 – 8 hours	7.72E-4 sec/m^3
8 - 24 hours	4.55E-4 sec/m^3
1 – 4 days	2.90E-4 sec/m^3
4 - 30 days	1.27E-4 sec/m^3
Containment/ESF Rooms to Emergency Intake	
0 – 8 hours	2.56E-4 sec/m^3
8 - 24 hours	1.51E-4 sec/m^3
1 – 4 days	9.60E-5 sec/m^3
4 - 30 days	4.22E-5 sec/m^3
SIRWT to Normal Intake	
0 – 8 hours	1.32E-2 sec/m^3
8 - 24 hours	7.78E-3 sec/m^3
1 – 4 days	4.95E-3 sec/m^3
4 - 30 days	2.18E-3 sec/m^3
SIRWT to Emergency Intake	
0 – 8 hours	6.35E-4 sec/m^3
8 - 24 hours	3.74E-4 sec/m^3
1 – 4 days	2.38E-4 sec/m^3
4 - 30 days	1.05E-4 sec/m^3
Control Room HVAC Ventilation System	
Normal Mode Ventilation System Flow Rates	

Table 7.2: CONDOSE Key Parameters

Unfiltered Intake Flow Rate	660 scfm
Loss of Offsite Power Flow Rates	
Base Unfiltered Infiltration Flow Rate	384.2 scfm
Emergency Mode Ventilation System Flow Rates	
Filtered Makeup Air Flow Rate	1413.6 scfm
Filtered Recirculation Flow Rate	1413.6 scfm
Unfiltered In-leakage	58 scfm
Filter Efficiencies	
Elemental	99%
Organic	99%
Particulate	99%
Delay time to Initiate Switchover of HVAC from Normal Operation to Emergency Operation with Offsite Power	1.5 minutes
Delay time to Re-Initiate HVAC Emergency Operation following Loss of Offsite Power	1.5 minutes
Nuclide Release Rates	From MHACALC
Dose and Dose Rate Conversion Factors	See Table 4.2

7.3 Results

Table 7.3 presents the results of the analysis. Results with TSP removed are given in bold below current licensing basis results for each dose contributor. All limits are met, provided a factor of 10 reduction in the control room thyroid dose is taken (See Assumption 6.4).

Table 7.3: Results

	Offsite Dose				Onsite Dose		
Dose Contributor	Limits [rem]		EAB[1] [rem]	LPZ[2] [rem]	Limits [rem]		CR [rem]
Containment Leakage	Thyroid	300	14.37 **56.64**	6.27 **8.17**			
	Whole Body	25	0.330 **0.330**	0.059 **0.059**			
ESF Leakage	Thyroid	300	3.685 **21.45**	3.697 **21.52**			
	Whole Body	25	N/A **N/A**	N/A **N/A**			
SIRWT Leakage	Thyroid	300	0.000 **0.080**	0.026 **7.41**			
	Whole Body	25	N/A **N/A**	N/A **N/A**			
Total Dose	Thyroid	300	18.05 **78.17**	9.99 **37.10**	Thyroid	30	21.26 **21.31[3]**
	Whole Body	25	0.330 **0.330**	0.059 **0.059**	Whole Body	5	2.71 **2.71[4]**

[1] 0 – 2 hour dose; [2] 30 day dose

[3] Includes credit for factor of 10 reduction due to compensatory measure (KI or SCBA) (Assumption 6.4)

[4] Sum of external shine doses (Inputs 4.25 and 4.26) and in control room whole body dose (Attachment 3).

8.0 CONCLUSIONS

With the removal of all TSP from containment:

- Off-site radiological consequences at the exclusion area and low population zone boundaries remain within 10-CFR-100 limits with no deviations from the current licensing basis.
- On-site radiological consequences at the control room remain within GDC 19 limits with no deviations from the current licensing basis, provided credit for compensatory measures such as potassium iodide (KI) or self-contained breathing apparatus (SCBA) is taken.

9.0 ATTACHMENTS

Attachment 1: SIRWT Iodine Airborne Fraction Calculation (1 page)

Attachment 2: MHACALC Output (10 pages)

Attachment 3: CONDOSE Output (8 pages)

SIRWT Iodine Partition Coefficient

Maximum SIRWT Water Temperature (F)	Maximum SIRWT Water Temperature (K)	Minimum SIRWT Partition Coefficient (unitless)
180.0	355.2	9.94

Ref. [8] - NUREG/CR-5950, Section 3, Equation (15)

SIRWT Iodine Volatile Fraction

[I]aq (g-atom/L)	pH	Term 1	Term 2	Term 3	Term 4	Term 5	$[I_2]$aq (g-atom/L)
6.72E-05	4.50E+00	3.36E-05	1.11E-05	3.95E+03	7.87E-18	4.77E-17	1.52E-05
		D=	4.22E-14				
		E=	1.47E-09				

Maximum Volatile Fraction
4.53E-01

Ref. [11] - EA-JLV-05-08, Attachment 2, Page 35
Ref. [8] - NUREG/CR-5950, Section 3, Equation (12)

SIRWT Iodine Airborne Fraction

Minimum SIRWT Partition Coefficient (unitless)	Maximum Volatile Fraction	Maximum Airborne Fraction
9.94E+00	4.53E-01	4.56E-02

MHACALC Output:

```
MHACALC VERSION-01
 LAST MODIFIED FEBRUARY 21, 1992

  EA-EC-976-01, No TSP: no spray, nat & wall dep, 10% ESF af, 4.56% SIRWT af
   1
     43200
    2580.6       0.10   41498.2
     100.0       25.0      50.0
      91.0        5.0       4.0
       2.0       10.0       1.0       2.0    40151.1      554.0  0.46E-01
  0.35E-03   0.17E-03  0.23E-03
  0.18E-03
  0.12E-04   0.77E-05  0.29E-05  0.75E-06
      0.10      0.100
         0      1.300
         0      0.000
         0      0.000
         0      0.000
         0      0.000
1000000.00          0
        19        0.2        19     2.200
         0        0.0         0     0.000
         0        0.0         0     0.000
         0        0.0         0     0.000
 0.300E+04 0.631E-02 0.000E+00 0.365E-05
 0.650E+04 0.258E-02 0.308E-01 0.303E-01
 0.300E+03 0.123E-06 0.000E+00 0.474E-03
 0.116E+05 0.908E-02 0.144E+00 0.145E+00
 0.169E+05 0.407E-02 0.380E+00 0.369E+00
 0.199E+05 0.219E+00 0.000E+00 0.000E+00
 0.176E+03 0.407E-04 0.000E+00 0.132E-02
 0.195E+04 0.220E-03 0.000E+00 0.538E-02
 0.565E+05 0.918E-04 0.730E-02 0.626E-02
 0.170E+05 0.451E-01 0.000E+00 0.765E-01
 0.978E+04 0.127E-02 0.000E+00 0.468E-01
 0.471E+05 0.181E+00 0.000E+00 0.000E+00
 0.443E+05 0.491E-01 0.195E+00 0.197E+00
 0.294E+05 0.599E-04 0.107E+07 0.326E+05
 0.416E+05 0.502E-02 0.629E+04 0.337E+03
 0.481E+05 0.555E-03 0.181E+06 0.555E+04
 0.622E+05 0.132E-01 0.107E+04 0.111E+03
 0.492E+05 0.175E-02 0.315E+05 0.112E+04
        11
      0.00       1.00      1.26     12.00      19.00     480.00     600.00
1440.00
   5760.00   14400.00  28800.00  43200.00
        15
        19         60       120       480        720       1440       2880
4320
      5760       7200     14400     21600      28800      36000      43200
1  EA EC 976 01, No TSP: no spray, nat & wall dep, 10% ESF af, 4.56% SIRWT af

                     INITIAL ACTIVITIES IN CONTAINMENT
```

TIME = 0 MIN ACTIVITY IN CONTAINMENT

ISOTOPE	CTMT. ATM. ACTIVITY (Ci)	SUMP ACTIVITY (Ci)
Kr-83m	0.7737E+07	
Kr-85m	0.1677E+08	
Kr-85	0.7739E+06	
Kr-87	0.2981E+08	
Kr-88	0.4361E+08	
Kr-89	0.5143E+08	
Xe-131m	0.4542E+06	
Xe-133m	0.5042E+07	
Xe-133	0.1458E+09	
Xe-135m	0.4382E+08	
Xe-135	0.2524E+08	
Xe-137	0.1214E+09	
Xe-138	0.1144E+09	
I-131	0.1895E+08	0.3791E+08
I-132	0.2684E+08	0.5368E+08
I-133	0.3102E+08	0.6204E+08
I-134	0.4012E+08	0.8023E+08
I-135	0.3175E+08	0.6351E+08

TIME = 19 MIN ACTIVITY IN CONTAINMENT AND SIRW TANK

ISOTOPE	CTMT. ATM. ACTIVITY (Ci)	SUMP ACTIVITY (Ci)	SIRW TANK ACTIVITY (CI)
Kr-83m	0.6862E+07		
Kr-85m	0.1597E+08		
Kr-85	0.7739E+06		
Kr-87	0.2508E+08		
Kr-88	0.4037E+08		
Kr-89	0.7958E+06		
Xe-131m	0.4538E+06		
Xe-133m	0.5021E+07		
Xe-133	0.1455E+09		
Xe-135m	0.1859E+08		
Xe-135	0.2464E+08		
Xe-137	0.3897E+07		
Xe-138	0.4504E+08		
I-131	0.1309E+08	0.3787E+08	0.0000E+00
I-132	0.1687E+08	0.4879E+08	0.0000E+00
I-133	0.2122E+08	0.6139E+08	0.0000E+00
I-134	0.2159E+08	0.6246E+08	0.0000E+00
I-135	0.2124E+08	0.6143E+08	0.0000E+00

1

TIME = 60 MIN ACTIVITY IN CONTAINMENT AND SIRW TANK

ISOTOPE	CTMT. ATM. ACTIVITY (Ci)	SUMP ACTIVITY (Ci)	SIRW TANK ACTIVITY (CI)
Kr-83m	0.5297E+07		
Kr-85m	0.1436E+08		
Kr-85	0.7739E+06		
Kr-87	0.1728E+08		
Kr-88	0.3417E+08		
Kr-89	0.9866E+02		
Xe-131m	0.4531E+06		
Xe-133m	0.4976E+07		
Xe-133	0.1449E+09		
Xe-135m	0.2922E+07		
Xe-135	0.2339E+08		
Xe-137	0.2333E+04		
Xe-138	0.6026E+07		
I-131	0.6294E+07	0.3776E+08	0.1097E+05
I-132	0.6616E+07	0.3970E+08	0.1157E+05
I-133	0.9998E+07	0.5998E+08	0.1744E+05
I-134	0.6062E+07	0.3637E+08	0.1064E+05
I-135	0.9528E+07	0.5717E+08	0.1663E+05

TIME = 120 MIN ACTIVITY IN CONTAINMENT AND SIRW TANK

ISOTOPE	CTMT. ATM. ACTIVITY (Ci)	SUMP ACTIVITY (Ci)	SIRW TANK ACTIVITY (CI)
Kr-83m	0.3627E+07		
Kr-85m	0.1230E+08		
Kr-85	0.7738E+06		
Kr-87	0.1002E+08		
Kr-88	0.2676E+08		
Kr-89	0.1893E-03		
Xe-131m	0.4519E+06		
Xe-133m	0.4911E+07		
Xe-133	0.1441E+09		
Xe-135m	0.1948E+06		
Xe-135	0.2168E+08		
Xe-137	0.4481E-01		
Xe-138	0.3174E+06		
I-131	0.2795E+07	0.3761E+08	0.2693E+05
I-132	0.2181E+07	0.2935E+08	0.2108E+05
I-133	0.4310E+07	0.5799E+08	0.4155E+05
I-134	0.1225E+07	0.1649E+08	0.1189E+05
I-135	0.3824E+07	0.5145E+08	0.3689E+05

1

TIME = 480 MIN ACTIVITY IN CONTAINMENT AND SIRW TANK

CTMT. ATM. SUMP SIRW TANK

ISOTOPE	ACTIVITY (Ci)	ACTIVITY (Ci)	ACTIVITY (CI)
Kr-83m	0.3736E+06		
Kr-85m	0.4861E+07		
Kr-85	0.7736E+06		
Kr-87	0.3806E+06		
Kr-88	0.6186E+07		
Kr-89	0.0000E+00		
Xe-131m	0.4453E+06		
Xe-133m	0.4536E+07		
Xe-133	0.1394E+09		
Xe-135m	0.1713E-01		
Xe-135	0.1373E+08		
Xe-137	0.0000E+00		
Xe-138	0.6779E-02		
I-131	0.1151E+07	0.3669E+08	0.1203E+06
I-132	0.1504E+06	0.4797E+07	0.1577E+05
I-133	0.1484E+07	0.4734E+08	0.1553E+06
I-134	0.4482E+04	0.1429E+06	0.4718E+03
I-135	0.8572E+06	0.2734E+08	0.8973E+05

TIME = 720 MIN ACTIVITY IN CONTAINMENT AND SIRW TANK

ISOTOPE	CTMT. ATM. ACTIVITY (Ci)	SUMP ACTIVITY (Ci)	SIRW TANK ACTIVITY (CI)
Kr-83m	0.8209E+05		
Kr-85m	0.2617E+07		
Kr-85	0.7735E+06		
Kr-87	0.4301E+05		
Kr-88	0.2330E+07		
Kr-89	0.0000E+00		
Xe-131m	0.4409E+06		
Xe-133m	0.4302E+07		
Xe-133	0.1364E+09		
Xe-135m	0.3386E-06		
Xe-135	0.1012E+08		
Xe-137	0.0000E+00		
Xe-138	0.5218E-07		
I-131	0.9991E+06	0.3610E+08	0.1803E+06
I-132	0.3970E+05	0.1434E+07	0.7183E+04
I-133	0.1144E+07	0.4135E+08	0.2066E+06
I-134	0.1670E+03	0.6033E+04	0.3034E+02
I-135	0.4964E+06	0.1793E+08	0.8968E+05

1

TIME = 1440 MIN ACTIVITY IN CONTAINMENT AND SIRW TANK

ISOTOPE	CTMT. ATM. ACTIVITY (Ci)	SUMP ACTIVITY (Ci)	SIRW TANK ACTIVITY (CI)

Kr-83m	0.8711E+03		
Kr-85m	0.4085E+06		
Kr-85	0.7730E+06		
Kr-87	0.6207E+02		
Kr-88	0.1245E+06		
Kr-89	0.0000E+00		
Xe-131m	0.4279E+06		
Xe-133m	0.3671E+07		
Xe-133	0.1276E+09		
Xe-135m	0.0000E+00		
Xe-135	0.4061E+07		
Xe-137	0.0000E+00		
Xe-138	0.0000E+00		
I-131	0.7737E+06	0.3436E+08	0.3499E+06
I-132	0.8625E+03	0.3831E+05	0.3912E+03
I-133	0.6203E+06	0.2755E+08	0.2807E+06
I-134	0.1021E-01	0.4535E+00	0.4650E-02
I-135	0.1140E+06	0.5064E+07	0.5163E+05

TIME = 2880 MIN ACTIVITY IN CONTAINMENT AND SIRW TANK

ISOTOPE	CTMT. ATM. ACTIVITY (Ci)	SUMP ACTIVITY (Ci)	SIRW TANK ACTIVITY (Ci)
Kr-83m	0.9812E-01		
Kr-85m	0.9957E+04		
Kr-85	0.7725E+06		
Kr-87	0.1293E-03		
Kr-88	0.3555E+03		
Kr-89	0.0000E+00		
Xe-131m	0.4034E+06		
Xe-133m	0.2673E+07		
Xe-133	0.1117E+09		
Xe-135m	0.0000E+00		
Xe-135	0.6538E+06		
Xe-137	0.0000E+00		
Xe-138	0.0000E+00		
I-131	0.6138E+06	0.3114E+08	0.6461E+06
I-132	0.5650E+00	0.2733E+02	0.5686E+00
I-133	0.2528E+06	0.1223E+08	0.2539E+06
I-134	0.5297E-10	0.2563E-08	0.5354E-10
I-135	0.8343E+04	0.4036E+06	0.8384E+04

1

TIME = 4320 MIN ACTIVITY IN CONTAINMENT AND SIRW TANK

ISOTOPE	CTMT. ATM. ACTIVITY (Ci)	SUMP ACTIVITY (Ci)	SIRW TANK ACTIVITY (CI)
Kr-83m	0.1105E-04		
Kr-85m	0.2427E+03		
Kr-85	0.7719E+06		

Kr-87	0.2694E-09		
Kr-88	0.1015E+01		
Kr-89	0.0000E+00		
Xe-131m	0.3803E+06		
Xe-133m	0.1947E+07		
Xe-133	0.9786E+08		
Xe-135m	0.0000E+00		
Xe-135	0.1053E+06		
Xe-137	0.0000E+00		
Xe-138	0.0000E+00		
I-131	0.5848E+06	0.2822E+08	0.8907E+06
I-132	0.4041E-03	0.1949E-01	0.6171E-03
I-133	0.1125E+06	0.5429E+07	0.1715E+06
I-134	0.2998E-18	0.1448E-16	0.4602E-18
I-135	0.6666E+03	0.3216E+05	0.1017E+04

TIME = 5760 MIN ACTIVITY IN CONTAINMENT AND SIRW TANK

ISOTOPE	CTMT. ATM. ACTIVITY (Ci)	SUMP ACTIVITY (Ci)	SIRW TANK ACTIVITY (CI)
Kr-83m	0.1245E-08		
Kr-85m	0.5914E+01		
Kr-85	0.7714E+06		
Kr-87	0.5614E-15		
Kr-88	0.2899E-02		
Kr-89	0.0000E+00		
Xe-131m	0.3584E+06		
Xe-133m	0.1418E+07		
Xe-133	0.8571E+08		
Xe-135m	0.0000E+00		
Xe-135	0.1694E+05		
Xe-137	0.0000E+00		
Xe-138	0.0000E+00		
I-131	0.5358E+06	0.2556E+08	0.1090E+07
I-132	0.2915E-06	0.1390E-04	0.5946E-06
I-133	0.5051E+05	0.2410E+07	0.1028E+06
I-134	0.0000E+00	0.0000E+00	0.0000E+00
I-135	0.5372E+02	0.2563E+04	0.1094E+03

1

TIME = 7200 MIN ACTIVITY IN CONTAINMENT AND SIRW TANK

ISOTOPE	CTMT. ATM. ACTIVITY (Ci)	SUMP ACTIVITY (Ci)	SIRW TANK ACTIVITY (CI)
Kr-83m	0.1403E-12		
Kr-85m	0.1441E+00		
Kr-85	0.7709E+06		
Kr-87	0.0000E+00		
Kr-88	0.8279E-05		
Kr-89	0.0000E+00		

Xe-131m	0.3379E+06		
Xe-133m	0.1033E+07		
Xe-133	0.7506E+08		
Xe-135m	0.0000E+00		
Xe-135	0.2728E+04		
Xe-137	0.0000E+00		
Xe-138	0.0000E+00		
I-131	0.4913E+06	0.2316E+08	0.1250E+07
I-132	0.2104E-09	0.9914E-08	0.5369E-09
I-133	0.2269E+05	0.1069E+07	0.5778E+05
I-134	0.0000E+00	0.0000E+00	0.0000E+00
I-135	0.4332E+01	0.2042E+03	0.1104E+02

TIME = 14400 MIN ACTIVITY IN CONTAINMENT AND SIRW TANK

ISOTOPE	CTMT. ATM. ACTIVITY (Ci)	SUMP ACTIVITY (Ci)	SIRW TANK ACTIVITY (CI)
Kr-83m	0.0000E+00		
Kr-85m	0.1240E-08		
Kr-85	0.7682E+06		
Kr-87	0.0000E+00		
Kr-88	0.1572E-17		
Kr-89	0.0000E+00		
Xe-131m	0.2515E+06		
Xe-133m	0.2117E+06		
Xe-133	0.3868E+08		
Xe-135m	0.0000E+00		
Xe-135	0.2950E+00		
Xe-137	0.0000E+00		
Xe-138	0.0000E+00		
I-131	0.3185E+06	0.1408E+08	0.1624E+07
I-132	0.0000E+00	0.0000E+00	0.0000E+00
I-133	0.4150E+03	0.1835E+05	0.2118E+04
I-134	0.0000E+00	0.0000E+00	0.0000E+00
I-135	0.1478E-04	0.6537E-03	0.7547E-04

1

TIME = 21600 MIN ACTIVITY IN CONTAINMENT AND SIRW TANK

ISOTOPE	CTMT. ATM. ACTIVITY (Ci)	SUMP ACTIVITY (Ci)	SIRW TANK ACTIVITY (CI)
Kr-83m	0.0000E+00		
Kr-85m	0.1066E-16		
Kr-85	0.7655E+06		
Kr-87	0.0000E+00		
Kr-88	0.0000E+00		
Kr-89	0.0000E+00		
Xe-131m	0.1872E+06		
Xe-133m	0.4338E+05		
Xe-133	0.1993E+08		

Xe-135m	0.0000E+00		
Xe-135	0.3190E-04		
Xe-137	0.0000E+00		
Xe-138	0.0000E+00		
I-131	0.2065E+06	0.8526E+07	0.1581E+07
I-132	0.0000E+00	0.0000E+00	0.0000E+00
I-133	0.7591E+01	0.3135E+03	0.5816E+02
I-134	0.0000E+00	0.0000E+00	0.0000E+00
I-135	0.5043E-10	0.2083E-08	0.3866E-09

TIME - 28800 MIN ACTIVITY IN CONTAINMENT AND SIRW TANK

ISOTOPE	CTMT. ATM. ACTIVITY (Ci)	SUMP ACTIVITY (Ci)	SIRW TANK ACTIVITY (CI)
Kr-83m	0.0000E+00		
Kr-85m	0.0000E+00		
Kr-85	0.7628E+06		
Kr-87	0.0000E+00		
Kr-88	0.0000E+00		
Kr-89	0.0000E+00		
Xe-131m	0.1393E+06		
Xe-133m	0.8891E+04		
Xe-133	0.1027E+08		
Xe-135m	0.0000E+00		
Xe-135	0.3449E-08		
Xe-137	0.0000E+00		
Xe-138	0.0000E+00		
I-131	0.1338E+06	0.5133E+07	0.1368E+07
I-132	0.0000E+00	0.0000E+00	0.0000E+00
I-133	0.1389E+00	0.5326E+01	0.1420E+01
I-134	0.0000E+00	0.0000E+00	0.0000E+00
I-135	0.1721E-15	0.6600E-14	0.1761E-14

1

TIME = 36000 MIN ACTIVITY IN CONTAINMENT AND SIRW TANK

ISOTOPE	CTMT. ATM. ACTIVITY (Ci)	SUMP ACTIVITY (Ci)	SIRW TANK ACTIVITY (CI)
Kr-83m	0.0000E+00		
Kr-85m	0.0000E+00		
Kr-85	0.7601E+06		
Kr-87	0.0000E+00		
Kr-88	0.0000E+00		
Kr-89	0.0000E+00		
Xe-131m	0.1037E+06		
Xe-133m	0.1822E+04		
Xe-133	0.5291E+07		
Xe-135m	0.0000E+00		
Xe-135	0.3729E-12		
Xe-137	0.0000E+00		

Xe-138	0.0000E+00		
I-131	0.8676E+05	0.3071E+07	0.1109E+07
I-132	0.0000E+00	0.0000E+00	0.0000E+00
I-133	0.2540E 02	0.8992E-01	0.3250E-01
I-134	0.0000E+00	0.0000E+00	0.0000E+00
I-135	0.0000E+00	0.2078E-19	0.0000E+00

TIME = 43200 MIN ACTIVITY IN CONTAINMENT AND SIRW TANK

ISOTOPE	CTMT. ATM. ACTIVITY (Ci)	SUMP ACTIVITY (Ci)	SIRW TANK ACTIVITY (CI)
Kr-83m	0.0000E+00		
Kr-85m	0.0000E+00		
Kr 85	0.7574E+06		
Kr-87	0.0000E+00		
Kr-88	0.0000E+00		
Kr-89	0.0000E+00		
Xe-131m	0.7720E+05		
Xe-133m	0.3734E+03		
Xe-133	0.2726E+07		
Xe-135m	0.0000E+00		
Xe-135	0.4033E-16		
Xe-137	0.0000E+00		
Xe-138	0.0000E+00		
I-131	0.5625E+05	0.1824E+07	0.8637E+06
I-132	0.0000E+00	0.0000E+00	0.0000E+00
I-133	0.4647E-04	0.1507E-02	0.7141E-03
I-134	0.0000E+00	0.0000E+00	0.0000E+00
I-135	0.0000E+00	0.0000E+00	0.0000E+00

1

1 TOTAL ACTIVITY OF EACH RADIONUCLIDE RELEASED (Ci)

ISOTOPE	CTMT ATM	ESF ROOMS	SIRW TANK
Kr-83m	0.8509E+03		
Kr-85m	0.4459E+04		
Kr-85	0.1204E+05		
Kr-87	0.2278E+04		
Kr-88	0.7433E+04		
Kr-89	0.1628E+03		
Xe-131m	0.3412E+04		
Xe-133m	0.1011E+05		
Xe-133	0.6071E+06		
Xe-135m	0.6743E+03		
Xe-135	0.1270E+05		
Xe-137	0.4658E+03		
Xe-138	0.1619E+04		
I-131	0.5325E+04	0.3767E+05	0.3135E+05
I-132	0.9231E+03	0.6258E+03	0.9151E+01
I-133	0.2558E+04	0.7121E+04	0.9182E+03
I-134	0.9058E+03	0.3053E+03	0.1712E+01
I-135	0.1663E+04	0.2264E+04	0.9448E+02

RESULTANT OFFSITE DOSES FROM THE EVENT (Rem)

0-2 Hr SB	CTMT ATM	ESF LEAKAGE	SIRWT LEAKAGE	TOTAL
Thyroid (inhalation)	56.635	21.447	0.080	78.162
Thyroid (submersion)	0.312	N/A	N/A	0.312
Total Thyroid Dose =	56.947	21.447	0.080	78.474
CEDE (inhalation)	1.743	0.659	0.002	2.405
Whole Body Dose	0.330	N/A	N/A	0.330
TEDE (whole body eq)	2.073	0.659	0.002	2.735
0-30 Day LPZ				
Thyroid (inhalation)	8.166	21.518	7.403	37.088
Thyroid (submersion)	0.055	N/A	N/A	0.055
Total Thyroid Dose =	8.221	21.518	7.403	37.143
CEDE (inhalation)	0.250	0.655	0.225	1.130
Whole Body Dose	0.059	N/A	N/A	0.059
TEDE (whole body eq)	0.309	0.655	0.225	1.189

TIME AT WHICH DFmax OR THE SPRAY STOP TIME WAS REACHED = 592 MINUTES

FILE TRAIL DATA

MHACALC PARAMETERS INPUT DECK = FORT5 = mha-no_tsp.mha
MHACALC DOSE OUTPUT DECK = FORT6 = mha-no_tsp.mout
STACK RELEASE RATES = FORT7 = mha-no_tspstk.rr
SIRWT RELEASE RATES = FORT13 = mha-no_tspsrw.rr

DATE AND TIME OF RUN =

Wed Mar 1 13:45:07 EST 2006

C01

CONDOSE SIRWT Release Output:

```
CONDOSE VERSION-00
 LAST MODIFIED FEBRUARY 10, 1992

 EA-EC-976-01, SIRWT release, No TSP, 58 scfm inleakage
 0.359E+05
         3
      0.00  0.35E-03
    480.00  0.17E-03
   1440.00  0.23E-03
         3
      0.00       1.0
   1440.00       0.6
   5760.00       0.4
         4
      0.00  0.13E-01  0.64E-03  0.00E+00
    480.00  0.78E-02  0.37E-03  0.00E+00
   1440.00  0.49E-02  0.24E-03  0.00E+00
   5760.00  0.22E-02  0.10E-03  0.00E+00
         3
      0.00     660.0       0.0       0.0       0.0
      1.50     384.2       0.0       0.0       0.0
      3.00      58.0    1413.6    1413.6       0.0
         1
      0.00     0.000     0.000     0.000     0.000     0.990     0.990
         0
        11
      0.00      1.00      1.26     12.00     19.00    480.00    600.00
1440.00
   5760.00  14400.00  28800.00  43200.00
        18         1
        18
 Kr-83m              1
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.000E+00 0.000E+00 0.648E-05 0.565E-05 0.175E-03 0.175E-03 0.365E-05
         1 0.631E-02         0     0.000
 Kr-85m              2
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.123E-02 0.113E-02 0.195E-02 0.185E-02 0.514E-01 0.154E-02 0.127E-02
         1 0.258E-02         0     0.000
 Kr-85               3
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.000E+00 0.206E-04 0.308E-04 0.288E-04 0.473E-01 0.391E-04 0.231E-04
         1 0.123E-06         0     0.000
 Kr-87               4
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.555E-02 0.576E-02 0.689E-02 0.627E-02 0.339E+00 0.101E+00 0.568E-02
         1 0.908E-02         0     0.000
 Kr-88               5
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.144E-01 0.134E-01 0.154E-01 0.134E-01 0.946E-01 0.288E-01 0.140E-01
         1 0.407E-02         0     0.000
 Kr-89               6
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
```

```
        1 0.219E+00          0       0.000
Xe-131m           7
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.123E-03 0.360E-03 0.329E-03 0.154E-01 0.534E-03 0.192E-03
        1 0.407E-04          0       0.000
Xe-133m           8
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.277E-03 0.617E-03 0.565E-03 0.308E-01 0.750E-03 0.382E-03
        1 0.220E-03          0       0.000
Xe-133            9
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.432E-03 0.267E-03 0.689E-03 0.627E-03 0.113E-01 0.699E-03 0.336E-03
        1 0.910E-04          0       0.000
Xe-135m          10
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.339E-02 0.452E-02 0.421E-02 0.267E-01 0.452E-02 0.362E-02
        1 0.451E-01          0       0.000
Xe-135           11
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.185E-02 0.298E-02 0.277E-02 0.658E-01 0.247E-02 0.209E-02
        1 0.127E-02          0       0.000
Xe-137           12
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
        1 0.181E+00          0       0.000
Xe-138           13
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.781E-02 0.812E-02 0.956E-02 0.874E-02 0.164E+00 0.319E-01 0.780E-02
        1 0.491E-01          0       0.000
I-131            14
0.107E+07 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.326E+05
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
        2 0.599E-04          0       0.000
I-132            15
0.629E+04 0.999E+03 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.337E+03
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
        2 0.502E-02          0       0.000
I-133            16
0.181E+06 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.555E+04
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
        2 0.555E-03          0       0.000
I-134            17
0.107E+04 0.518E+03 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.111E+03
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
        2 0.132E-01          0       0.000
I-135            18
0.315E+05 0.163E+04 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.112E+04
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
        2 0.175E-02          0       0.000
 ECHO OF RELEASE RATES IN THE FOLLOWING ORDER:
 Kr-83m    Kr-85m    Kr-85     Kr-87     Kr-88     Kr-89
 Xe-131m   Xe-133m   Xe-133    Xe-135m   Xe-135    Xe-137
 Xe-138    I-131     I-132     I-133     I-134     I-135
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
```

0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.243E+07	0.803E+06	0.339E+07	0.219E+06	0.240E+07
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.541E+07	0.528E+06	0.677E+07	0.103E+05	0.364E+07
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.100E+08	0.133E+06	0.970E+07	0.495E+03	0.297E+07
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.295E+08	0.813E+03	0.818E+07	0.339E-02	0.378E+06
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.536E+08	0.544E-06	0.105E+07	0.000E+00	0.305E+03
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.538E+08	0.000E+00	0.107E+05	0.000E+00	0.113E-03
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.000E+00	0.351E+08	0.000E+00	0.614E+01	0.000E+00	0.234E-14

1 EA-EC-976-01, SIRWT release, No TSP, 58 scfm inleakage

SUMMARY OF CONTROL ROOM OPERATOR DOSES

ORGAN	INHALATION DOSE (Rem)	SUBMERSION DOSE (Rem)	TOTAL DOSE (Rem)
Thyroid	170.0527	0.0000	170.0527
Lung	0.0062	0.0000	0.0062
Bone Surface	0.0000	0.0000	0.0000
Bone Marrow	0.0000	0.0000	0.0000
Beta Skin	N/A	0.0000	0.0000
Eye Lens	N/A	0.0000	0.0000
Whole Body	5.1615 *	0.0000	5.1615 **

* - The whole body inhalation dose is the Committed Effective Dose Equivalent.
**- The total value for whole body dose is the Total Effective Dose Equivalent.

FILE TRAIL DATA

CONTROL ROOM PARAMETERS INPUT DECK = FORT4 = mha-no_tspsrw.con
RELEASE RATE INPUT DECK = FORT4 = mha-no_tspsrw.rr
CONDOSE DOSE OUTPUT DECK = FORT7 = mha-no_tspsrw.cout

DATE AND TIME OF RUN =

Wed Mar 1 13:45:07 EST 2006

C02

CONDOSE Containment and ESF Room Release Output:

```
CONDOSE VERSION-00
 LAST MODIFIED FEBRUARY 10, 1992

 EA-EC-976-01, STACK release, No TSP, 58 scfm inleakage
 0.359E+05
         3
      0.00  0.35E-03
    480.00  0.17E-03
   1440.00  0.23E-03
         3
      0.00       1.0
   1440.00       0.6
   5760.00       0.4
         4
      0.00  0.77E-03  0.26E-03  0.00E+00
    480.00  0.46E-03  0.15E-03  0.00E+00
   1440.00  0.29E-03  0.96E-04  0.00E+00
   5760.00  0.13E-03  0.42E-04  0.00E+00
         3
      0.00     660.0       0.0       0.0       0.0
      1.50     384.2       0.0       0.0       0.0
      3.00      58.0    1413.6    1413.6       0.0
         1
      0.00     0.000     0.000     0.000     0.000     0.990     0.990
         0
        11
      0.00      1.00      1.26     12.00     19.00    480.00    600.00
1440.00
   5760.00  14400.00  28800.00  43200.00
        18         1
        18
 Kr-83m              1
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.000E+00 0.000E+00 0.648E-05 0.565E-05 0.175E-03 0.175E-03 0.365E-05
         1 0.631E-02         0     0.000
 Kr-85m              2
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.123E-02 0.113E-02 0.195E-02 0.185E-02 0.514E-01 0.154E-02 0.127E-02
         1 0.258E-02         0     0.000
 Kr-85               3
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.000E+00 0.206E-04 0.308E-04 0.288E-04 0.473E-01 0.391E-04 0.231E-04
         1 0.123E-06         0     0.000
 Kr-87               4
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.555E-02 0.576E-02 0.689E-02 0.627E-02 0.339E+00 0.101E+00 0.568E-02
         1 0.908E-02         0     0.000
 Kr-88               5
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.144E-01 0.134E-01 0.154E-01 0.134E-01 0.946E-01 0.288E-01 0.140E-01
         1 0.407E-02         0     0.000
 Kr-89               6
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
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```
         1 0.219E+00         0        0.000
Xe-131m              7
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.123E-03 0.360E-03 0.329E-03 0.154E-01 0.534E-03 0.192E-03
         1 0.407E-04         0        0.000
Xe-133m              8
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.277E-03 0.617E-03 0.565E-03 0.308E-01 0.750E-03 0.382E-03
         1 0.220E-03         0        0.000
Xe-133               9
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.432E-03 0.267E-03 0.689E-03 0.627E-03 0.113E-01 0.699E-03 0.336E-03
         1 0.918E-04         0        0.000
Xe-135m             10
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.339E-02 0.452E-02 0.421E-02 0.267E-01 0.452E-02 0.362E-02
         1 0.451E-01         0        0.000
Xe-135              11
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.185E-02 0.298E-02 0.277E-02 0.658E-01 0.247E-02 0.209E-02
         1 0.127E-02         0        0.000
Xe-137              12
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
         1 0.181E+00         0        0.000
Xe-138              13
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.781E-02 0.812E-02 0.956E-02 0.874E-02 0.164E+00 0.319E-01 0.780E-02
         1 0.491E-01         0        0.000
I-131               14
0.107E+07 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.326E+05
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
         2 0.599E-04         0        0.000
I-132               15
0.629E+04 0.999E+03 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.337E+03
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
         2 0.502E-02         0        0.000
I-133               16
0.181E+06 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.555E+04
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
         2 0.555E-03         0        0.000
I-134               17
0.107E+04 0.518E+03 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.111E+03
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
         2 0.132E-01         0        0.000
I-135               18
0.315E+05 0.163E+04 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.112E+04
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
         2 0.175E-02         0        0.000
 ECHO OF RELEASE RATES IN THE FOLLOWING ORDER:
 Kr-83m    Kr-85m    Kr-85     Kr-87     Kr-88     Kr-89
 Xe-131m   Xe-133m   Xe-133    Xe-135m   Xe-135    Xe-137
 Xe-138    I-131     I-132     I-133     I-134     I-135
0.321E+09 0.698E+09 0.329E+08 0.124E+10 0.181E+10 0.192E+10
0.189E+08 0.210E+09 0.607E+10 0.179E+10 0.105E+10 0.463E+10
0.465E+10 0.782E+09 0.110E+10 0.128E+10 0.164E+10 0.131E+10
0.319E+09 0.696E+09 0.329E+08 0.123E+10 0.181E+10 0.155E+10
0.189E+08 0.210E+09 0.607E+10 0.171E+10 0.105E+10 0.386E+10
```

0.443E+10	0.767E+09	0.108E+10	0.125E+10	0.159E+10	0.128E+10
0.309E+09	0.687E+09	0.329E+08	0.117E+10	0.177E+10	0.628E+09
0.189E+08	0.210E+09	0.607E+10	0.137E+10	0.104E+10	0.178E+10
0.348E+10	0.694E+09	0.953E+09	0.113E+10	0.135E+10	0.115E+10
0.292E+09	0.671E+09	0.329E+08	0.108E+10	0.171E+10	0.787E+08
0.189E+08	0.209E+09	0.606E+10	0.911E+09	0.103E+10	0.327E+09
0.224E+10	0.584E+09	0.766E+09	0.948E+09	0.101E+10	0.953E+09
0.929E+08	0.389E+09	0.329E+08	0.246E+09	0.759E+09	0.328E+06
0.187E+08	0.199E+09	0.593E+10	0.372E+08	0.777E+09	0.195E+07
0.830E+08	0.249E+09	0.154E+09	0.368E+09	0.106E+09	0.301E+09
0.109E+08	0.174E+09	0.329E+08	0.966E+07	0.204E+09	0.000E+00
0.185E+08	0.187E+09	0.578E+10	0.131E+00	0.531E+09	0.000E+00
0.478E-01	0.188E+09	0.185E+08	0.236E+09	0.370E+06	0.127E+09
0.137E+07	0.607E+08	0.329E+08	0.698E+06	0.448E+08	0.000E+00
0.182E+08	0.168E+09	0.553E+10	0.837E-04	0.302E+09	0.000E+00
0.190E-04	0.175E+09	0.307E+07	0.175E+09	0.134E+05	0.576E+08
0.665E+03	0.764E+06	0.163E+08	0.329E+02	0.146E+06	0.000E+00
0.817E+07	0.493E+08	0.219E+10	0.000E+00	0.154E+08	0.000E+00
0.000E+00	0.131E+09	0.771E+04	0.454E+08	0.349E-01	0.292E+07
0.476E-09	0.553E+01	0.163E+08	0.149E-15	0.172E-02	0.000E+00
0.629E+07	0.132E+08	0.123E+10	0.000E+00	0.322E+05	0.000E+00
0.000E+00	0.897E+08	0.144E-05	0.224E+07	0.000E+00	0.763E+03
0.000E+00	0.695E-09	0.162E+08	0.000E+00	0.556E-18	0.000E+00
0.396E+07	0.133E+07	0.446E+09	0.000E+00	0.336E+00	0.000E+00
0.000E+00	0.459E+08	0.000E+00	0.112E+05	0.000E+00	0.126E-03
0.000E+00	0.000E+00	0.161E+08	0.000E+00	0.000E+00	0.000E+00
0.219E+07	0.560E+05	0.118E+09	0.000E+00	0.393E-08	0.000E+00
0.000E+00	0.194E+08	0.000E+00	0.375E+01	0.000E+00	0.148E-14

1 EA-EC-976-01, STACK release, No TSP, 58 scfm inleakage

SUMMARY OF CONTROL ROOM OPERATOR DOSES

ORGAN	INHALATION DOSE (Rem)	SUBMERSION DOSE (Rem)	TOTAL DOSE (Rem)
Thyroid	42.9928	0.0496	43.0424
Lung	0.0476	0.0463	0.0939
Bone Surface	0.0000	0.0729	0.0729
Bone Marrow	0.0000	0.0656	0.0656
Beta Skin	N/A	1.1418	1.1418
Eye Lens	N/A	0.1486	0.1486
Whole Body	1.3138 *	0.0511	1.3649 **

* - The whole body inhalation dose is the Committed Effective Dose Equivalent.
**- The total value for whole body dose is the Total Effective Dose Equivalent.

FILE TRAIL DATA

CONTROL ROOM PARAMETERS INPUT DECK = FORT4 = mha-no_tspstk.con
RELEASE RATE INPUT DECK = FORT4 = mha-no_tspstk.rr
CONDOSE DOSE OUTPUT DECK = FORT7 = mha-no_tspstk.cout

DATE AND TIME OF RUN =

Wed Mar 1 13:45:07 EST 2006

C03

ENCLOSURE 5

LICENSE AMENDMENT REQUEST: REMOVAL OF TSP FROM PALISADES CONTAINMENT

TECHNICAL SPECIFICATION BASES CHANGES

TS BASES 3.6, "Containment Systems"

TS BASES 3.5, "Emergency Core Cooling Systems"

17 Pages Follow

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Cooling Systems

BASES

BACKGROUND

The Containment Spray and Containment Air Cooler systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a Main Steam Line Break (MSLB) or a large break Loss of Coolant Accident (LOCA). The Containment Spray and Containment Air Cooler systems are designed to the requirements of the Palisades Nuclear Plant design criteria (Ref. 1).

The Containment Air Cooler System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The systems are arranged with two spray pumps and one air cooler fan powered from one diesel generator, and with one spray pump and three air cooler fans powered from the other diesel generator. The Containment Spray System was originally designed to be redundant to the Containment Air Coolers (CACs) and fans. These systems were originally designed such that either two containment spray pumps or three CACs could limit containment pressure to less than design. However, the current safety analyses take credit for one containment spray pump when evaluating cases with three CACs, and for one air cooler fan in cases with two spray pumps and both Main Steam Isolation Valve (MSIV) bypass valves closed. If an MSIV bypass valve is open, 2 service water pumps and 2 CACs are also required to be OPERABLE in addition to the 2 spray pumps for containment heat removal.

To address this dependency between the Containment Spray System and the Containment Air Cooler System the title of this Specification is "Containment Cooling Systems," and includes both systems. The LCO is written in terms of trains of containment cooling. One train of containment cooling is associated with Diesel Generator 1-1 and includes Containment Spray Pumps P-54B and P-54C, Containment Spray Valve CV-3001 and the associated spray header, and Air Cooler Fan V-4A. The other train of containment cooling is associated with Diesel Generator 1-2 and includes Containment Spray Pump P-54A, Containment Spray Valve CV-3002 and the associated spray header, and CACs VHX-1, VHX-2, and VHX-3 and their associated safety related fans, V-1A, V-2A, and V-3A.

BASES

BACKGROUND (continued)

If reliance is placed solely on one spray pump and three CACs, at least two service water pumps must be OPERABLE to provide the necessary service water flow to assure OPERABILITY of the CACs. Additional details of the required equipment and its operation is discussed with the containment cooling system with which it is associated.

Containment Spray System

The Containment Spray System consists of three half-capacity (50%) motor driven pumps, two shutdown cooling heat exchangers, two spray headers, two full sets of full capacity (100%) nozzles, valves, and piping, two full capacity (100%) pump suction lines from the Safety Injection and Refueling Water Tank (SIRWT) and the containment sump with the associated piping, valves, power sources, instruments, and controls. The heat exchangers are shared with the Shutdown Cooling System. SIRWT supplies borated water to the containment spray during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the SIRWT to the containment sump.

Normally, both Shutdown Cooling Heat Exchangers must be available to provide cooling of the containment spray flow in the event of a Loss of Coolant Accident. If the Containment Spray side (tube side) of one SDC Heat Exchanger is out of service, 100% of the required post accident cooling capability can be provided, if other equipment outages are limited (refer to Bases for Required Action C.1).

The Containment Spray System provides a spray of cold borated water into the upper regions of containment to reduce the containment pressure and temperature during a MSLB or large break LOCA event. In addition, the Containment Spray System ~~in conjunction with the use of trisodium phosphate (LCO 3.5.5, "Trisodium Phosphate,") serve~~serves to remove iodine which may be released following an accident. The SIRWT solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase.

BASES

BACKGROUND

Containment Spray System (continued)

In the recirculation mode of operation, heat is removed from the containment sump water by the shutdown cooling heat exchangers.

The Containment Spray System is actuated either automatically by a Containment High Pressure (CHP) signal or manually. An automatic actuation opens the containment spray header isolation valves, starts the three containment spray pumps, and begins the injection phase. Individual component controls may be used to manually initiate Containment Spray. The injection phase continues until an SIRWT Level Low signal is received. The Low Level signal for the SIRWT generates a Recirculation Actuation Signal (RAS) that aligns valves from the containment spray pump suction to the containment sump. RAS opens the HPSI subcooling valve CV-3071, if the associated HPSI pump is operating. After the containment sump valve CV-3030 opens from RAS, HPSI subcooling valve CV-3070 will open, if the associated HPSI pump is operating. RAS will close containment spray valve CV-3001, if containment sump valve CV-3030 does not open. The Containment Spray System in recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operating procedures.

The containment spray pumps also provide a required support function for the High Pressure Safety Injection pumps as described in the Bases for specification 3.5.2. The High Pressure Safety Injection pumps alone may not have adequate NPSH after a postulated accident and the realignment of their suctions from the SIRWT to the containment sump. Flow is automatically provided from the discharge of the containment spray pumps to the suction of the High Pressure Safety Injection (HPSI) pumps after the change to recirculation mode has occurred, if the HPSI pump is operating. The additional suction pressure ensures that adequate NPSH is available for the High Pressure Safety Injection pumps.

BASES

BACKGROUND (continued)

Containment Air Cooler System

The Containment Air Cooler System includes four air handling and cooling units, referred to as the Containment Air Coolers (CACs), which are located entirely within the containment building. Three of the CACs (VHX-1, VHX-2, and VHX-3) are safety related coolers and are cooled by the critical service water. The fourth CAC (VHX-4) is not taken credit for in maintaining containment temperature within limit (the service water inlet valve for VHX-4 is closed by an SIS signal to conserve service water flow), but is used during normal operation along with the three CACs to maintain containment temperature below the design limits. The fan associated with VHX-4, V-4A, is assumed in the safety analysis as assisting in the containment atmosphere mixing function.

The DG which powers the fans associated with VHX-1, VHX-2, and VHX-3 (V-1A, V-2A and V-3A) also powers two service water pumps. This is necessary because if reliance is placed solely on the train with one spray pump and three CACs, at least two service water pumps must be OPERABLE to provide the necessary service water flow to assure OPERABILITY of the CACs.

Each CAC has two vaneaxial fans with direct connected motors which draw air through the cooling coils. Both of these fans are normally in operation, but only one fan and motor for each CAC is rated for post accident conditions. The post accident rated "safety related" fan units, V-1A, V-2A, V-3A, and V-4A, serve not only to provide forced flow for the associated cooler, but also provide mixing of the containment atmosphere. A single operating safety related fan unit will provide enough air flow to assure that there is adequate mixing of unsprayed containment areas to assure the assumed iodine removal by the containment spray. The fan units also support the functioning of the hydrogen recombiners, as discussed in the Bases for LCO 3.6.7, "Hydrogen Recombiners." In post accident operation following a SIS, all four Containment air coolers are designed to change automatically to the emergency mode.

The CACs are automatically changed to the emergency mode by a Safety Injection Signal (SIS). This signal will trip the normal rated fan motor in each unit, open the high-capacity service water discharge valve from VHX-1, VHX-2, and VHX-3, and close the high-capacity service water supply valve to VHX-4. The test to verify the service water valves actuate to their correct position upon receipt of an SIS signal is included in the surveillance test performed as part of Specification 3.7.8, "Service Water System." The safety related fans are normally in operation and only receive an actuation signal through the DBA sequencers following an SIS in conjunction with a loss of offsite power. This actuation is tested by the surveillance which verifies the energizing of loads from the DBA sequencers in Specification 3.8.1, "AC Sources-Operating."

BASES

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Air Cooler System limit the temperature and pressure that could be experienced following either a Loss of Coolant Accident (LOCA) or a Main Steam Line Break (MSLB). The large break LOCA and MSLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients.

The Containment Cooling Systems have been analyzed for three accident cases (Ref. 2). All accidents analyses account for the most limiting single active failure.

1. A Large Break LOCA,

2. An MSLB occurring at various power levels with both MSIV bypass valves closed, and

3. An MSLB occurring at 0% RTP with both MSIV bypass valves open.

The postulated large break LOCA is analyzed, in regard to containment ESF systems, assuming the loss of offsite power and the loss of one ESF bus, which is the worst case single active failure, resulting in one train of Containment Cooling being rendered inoperable (Ref. 6).

The postulated MSLB is analyzed, in regard to containment ESF systems, assuming the worst case single active failure.

The MSLB event is analyzed at various power levels with both MSIV bypass valves closed, and at 0% RTP with both MSIV bypass valves open. Having any MSIV bypass valve open allows additional blowdown from the intact steam generator.

The analysis and evaluation show that under the worst-case scenario, the highest peak containment pressure and the peak containment vapor temperature are within the intent of the design basis. (See the Bases for Specifications 3.6.4, "Containment Pressure," and 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations considered a range of power levels and equipment configurations as described in Reference 2. The peak containment pressure case is the 0% power MSLB with initial (pre-accident) conditions of 140°F and 16.2 psia. The peak temperature case is the 102% power MSLB with initial (pre-accident) conditions of 140°F and 15.7 psia. The analyses also assume a response time delayed initiation in order to provide conservative peak calculated containment pressure and temperature responses.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The external design pressure of the containment shell is 3 psig. This value is approximately 0.5 psig greater than the maximum external pressure that could be developed if the containment were sealed during a period of low barometric pressure and high temperature and, subsequently, the containment atmosphere was cooled with a concurrent major rise in barometric pressure.

The modeled Containment Cooling System actuation from the containment analysis is based on a response time associated with exceeding the Containment High Pressure setpoint to achieve full flow through the CACs and containment spray nozzles. The spray lines within containment are maintained filled to the 735 ft elevation to provide for rapid spray initiation. The Containment Cooling System total response time of < 60 seconds includes diesel generator startup (for loss of offsite power), loading of equipment, CAC and containment spray pump startup, and spray line filling.

The performance of the Containment Spray System for post accident conditions is given in Reference 3. The performance of the Containment Air Coolers is given in Reference 4.

The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

During an MSLB or large break LOCA event, a minimum of one containment cooling train is required to maintain the containment peak pressure and temperature below the design limits (Ref. 2). One train of containment cooling is associated with Diesel Generator 1-1 and includes Containment Spray Pumps P-54B and P-54C, Containment Spray Valve CV-3001 and the associated spray header, and air cooler fan V-4A. This train must be supplemented with 2 service water pumps and 2 containment air coolers if an MSIV bypass valve is open. The other train of containment cooling is associated with Diesel Generator 1-2 and includes Containment Spray Pump P-54A, Containment Spray Valve CV-3002 and the associated spray header, and CACs VHX-1, VHX-2, and VHX-3 and their associated safety related fans, V-1A, V-2A, and V-3A. To ensure that these requirements are met, two trains of containment cooling must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst-case single active failure occurs.

BASES

LCO (continued)

The Containment Spray System portion of the containment cooling trains includes three spray pumps, two spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the SIRWT upon an ESF actuation signal and automatically transferring suction to the containment sump.

The Containment Air Cooler System portion of the containment cooling train which must be OPERABLE includes the three safety related air coolers which each consist of four cooling coil banks, the safety related fan which must be in operation to be OPERABLE, gravity-operated fan discharge dampers, instruments, and controls to ensure an OPERABLE flow path.

CAC fans V-1A, V-2A, V-3A, and V-4A must be in operation to be considered OPERABLE. These fans only receive a start signal from the DBA sequencer; they are assumed to be in operation, and are not started by either a CHP or an SIS signal.

APPLICABILITY

In MODES 1, 2, and 3, a large break LOCA event could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains and containment cooling trains.

In MODES 4, 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray and Containment Cooling systems are not required to be OPERABLE in MODES 4, 5 and 6.

ACTIONS

A.1

Condition A is applicable whenever one or more containment cooling trains is inoperable. Action A.1 requires restoration of both trains to OPERABLE status within 72 hours. The 72-hour Completion Time for Condition A is based on the assumption that at least 100% of the required post accident containment cooling capability (that assumed in the safety analyses) is available. If less than 100% of the required post containment accident cooling is available, Condition C must also be entered.

Mechanical system LCOs typically provide a 72 hour Completion Time under conditions when a required system can perform its required safety function, but may not be able to do so assuming an additional failure. When operating in accordance with the Required Actions of an LCO Condition, it is not necessary to be able to cope with an additional single failure.

BASES

ACTIONS

A.1 (continued)

The Containment Cooling systems can provide one hundred percent of the required post accident cooling capability following the occurrence of any single active failure. Therefore, the containment cooling function can be met during conditions when those components which could be deactivated by a single active failure are known to be inoperable. Under that condition, however, the ability to provide the function after the occurrence of an additional failure cannot be guaranteed. Therefore, continued operation with one or more trains inoperable is allowed only for a limited time.

B.1 and B.2

Condition B is applicable when the Required Actions of Condition A cannot be completed within the required Completion Time. Condition A is applicable whenever one or more trains is inoperable. Therefore, when Condition B is applicable, Condition A is also applicable. (If less than 100% of the post accident containment cooling capability is available, Condition C must be entered as well.) Being in Conditions A and B concurrently maintains both Completion Time clocks for instances where equipment repair allows exit from Condition B while the plant is still within the applicable conditions of the LCO.

If the inoperable containment cooling trains cannot be restored to OPERABLE status within the required Completion Time of Condition A, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

Condition C is applicable with one or more trains inoperable when there is less than 100% of the required post accident containment cooling capability available. Condition A is applicable whenever one or more trains is inoperable. Therefore, when this Condition is applicable, Condition A is also applicable. Being in Conditions A and C concurrently maintains both Completion Time clocks for instances where equipment repair restores 100% of the required post accident containment cooling capability while the LCO is still applicable, allowing exit from Condition C (and LCO 3.0.3).

BASES

ACTIONS

C.1 (continued)

Several specific cases have been analyzed in the safety analysis to provide operating flexibility for equipment outages and testing. These analyses show that action A.1 can be entered under certain circumstances, because 100% of the post accident cooling capability is maintained. These specific cases are discussed below.

One hundred percent of the required post accident cooling capability can be provided with both MSIV bypass valves closed if either;

1. Two containment spray pumps, two spray headers, and one CAC fan are OPERABLE, or

2. One containment spray pump, two spray headers, and three safety related CACs, are OPERABLE (at least two service water pumps must be OPERABLE if CACs are to be relied upon).

One hundred percent of the required post accident cooling capability can be provided for operation with a MSIV bypass valve open or closed if either;

1. Two containment spray pumps, two spray headers, and two safety related CACs, are OPERABLE (at least two service water pumps must be OPERABLE if CACs are to be relied upon), or

2. One containment spray pump, one spray header, and three safety related CACs are OPERABLE (at least three service water pumps must be OPERABLE to provide the necessary service water flow to assure OPERABILITY of the CACs).

If the Containment Spray side (tube side) of SDC Heat Exchanger E-60B is out of service, 100% of the required post accident cooling capability can be provided, if other equipment outages are limited. One hundred percent of the post accident cooling can be provided with the Containment Spray side of SDC Heat Exchanger E-60B out of service if the following equipment is OPERABLE: three safety related Containment Air Coolers, two Containment Spray Pumps, two spray headers, CCW pumps P-52A and P-52B, two SWS pumps, and both CCW Heat Exchangers, and if

1. One CCW Containment Isolation Valve, CV-0910, CV-0911, or CV-0940, is OPERABLE, and

2. Two CCW isolation valves for the non-safety related loads outside the containment, CV-0944A and CV-0944 (or CV-0977B), are OPERABLE.

BASES

ACTIONS

C.1 (continued)

With less than 100% of the required post accident containment cooling capability available, the plant is in a condition outside the assumptions of the safety analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment for manual, power operated, and automatic valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path exists for Containment Spray System operation. This SR also does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct positions prior to being secured. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verification~~, through a system walkdown,~~ that those valves outside containment and capable of potentially being mispositioned, are in the correct position.

SR 3.6.6.2

Operating each safety related Containment Air Cooler fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and are functioning properly. The 31-day Frequency was developed considering the known reliability of the fan units, the two train redundancy available, and the low probability of a significant degradation of the containment cooling train occurring between surveillances.

SR 3.6.6.3

Verifying the containment spray header is full of water to the 735 ft elevation minimizes the time required to fill the header. This ensures that spray flow will be admitted to the containment atmosphere within the time frame assumed in the containment analysis. The 31-day Frequency is based on the static nature of the fill header and the low probability of a significant degradation of the water level in the piping occurring between surveillances.

SR 3.6.6.4

Verifying a total service water flow rate of ≥ 4800 gpm to CACs VHX-1, VHX-2, and VHX-3, when aligned for accident conditions, provides assurance the design flow rate assumed in the safety analyses will be achieved (Ref. 8). Also considered in selecting this Frequency were the

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.4 (continued)

known reliability of the cooling water system, the two train redundancy, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6.5

Verifying that each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 5).

Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.6 and SR 3.6.6.7

SR 3.6.6.6 verifies each automatic containment spray valve actuates to its correct position upon receipt of an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. SR 3.6.6.7 verifies each containment spray pump starts automatically on an actual or simulated actuation signal. The 18-month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power.

Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Where the surveillance of containment sump isolation valves is also required by SR 3.5.2.5, a single surveillance may be used to satisfy both requirements.

SR 3.6.6.8

This SR verifies each containment cooling fan actuates upon receipt of an actual or simulated actuation signal. The 18-month Frequency is

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.8 (continued)

based on engineering judgement and has been shown to be acceptable through operating experience. See SR 3.6.6.6 and SR 3.6.6.7, above, for further discussion of the basis for the 18 month Frequency.

SR 3.6.6.9

With the containment spray inlet valves closed and the spray header drained of any solution, an inspection of spray nozzles, or a test that blows low-pressure air or smoke through test connections can be completed. Performance of this SR demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Verification following maintenance which could result in nozzle blockage is appropriate because this is the only activity that could lead to nozzle blockage.

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Section 14.18
3. FSAR, Sections 6.2
4. FSAR, Section 6.3
5. ASME, Boiler and Pressure Vessel Code, Section XI
6. FSAR, Table 14.18.1-3
7. FSAR, Table 14.18.2-1
8. FSAR, Table 9-1
9. EA-MSLB-2001-01 Rev. 1, Containment Response to a MSLB Using CONTEMPT-LT/28, January 2002.
10. EA-LOCA-2001-01 Rev. 1, Containment Response to a LOCA Using CONTEMPT-LT/28, January 2002.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 ~~Trisodium Phosphate (TSP)~~ (PAGE LEFT BLANK INTENTIONALLY)

~~BASES~~

~~BACKGROUND~~

~~TSP baskets are is placed on the floor (590 ft elevation) in the containment building to ensure that iodine, which may be dissolved in the recirculated primary cooling water following a Loss of Coolant Accident (LOCA), remains in solution (Ref. 1). Recirculation of the water for core cooling and containment spray provides mixing to achieve a uniform neutral pH. TSP also helps inhibit Stress Corrosion Cracking (SCC) of austenitic stainless steel components in containment during the recirculation phase following an accident.~~

~~Fuel that is damaged during a LOCA will release iodine in several chemical forms to the reactor coolant and to the containment atmosphere. A portion of the iodine in the containment atmosphere is washed to the sump by containment sprays. The Safety Injection Refueling Water Tank water is borated for reactivity control. This borated water, if left untreated, would cause the sump solution to be acidic. In a low pH (acidic) solution, dissolved iodine will be converted to a volatile form. The volatile iodine will evolve out of solution into the containment atmosphere, significantly increasing the levels of airborne iodine. The increased levels of airborne iodine in containment contribute to the radiological releases and increase the consequences from the accident due to containment atmosphere leakage.~~

~~After a LOCA, the components of the safety injection and containment spray systems will be exposed to high temperature borated water. Prolonged exposure to hot untreated sump water combined with stresses imposed on the components can cause SCC. The rate of SCC is a function of stress, oxygen and chloride concentrations, pH, temperature, and alloy composition of the components. High temperatures and low pH, which would be present after a LOCA, tend to promote SCC. This can lead to the failure of necessary safety systems or components.~~

BASES

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~~BACKGROUND (continued)~~

~~Adjusting the pH of the recirculation solution to levels above 7.0 prevents a significant fraction of the dissolved iodine from converting to a volatile form. The higher pH thus decreases the level of airborne iodine in containment and reduces the radiological consequences from containment atmosphere leakage following a LOCA. Maintaining the solution pH above 7.0 also reduces the occurrence of SCC of austenitic stainless steel components in containment. Reducing SCC reduces the probability of failure of components.~~

~~The hydrated form (45-57% moisture) of TSP is used because of the high humidity in the containment building during normal operation. Since the TSP is hydrated, it is less likely to absorb large amounts of water from the humid atmosphere and will undergo less physical and chemical change than the anhydrous form of TSP.~~

~~APPLICABLE SAFETY ANALYSES~~

~~The LOCA radiological consequences analysis takes credit for iodine retention in the sump solution based on the recirculation water pH being ≥ 7.0. The radionuclide releases from the containment atmosphere and the consequences of a LOCA would be increased if the pH of the recirculation water were not adjusted to 7.0 or above.~~

~~The containment hydrogen concentration analysis used in the evaluation of the Maximum Hypothetical Accident (MHA) assumes the pH of the containment sump water is between 7.0 and 8.0. The acceptance criteria of the MHA includes a containment lower flammability limit of 4 volume percent for hydrogen. Containment sump water with a pH greater than 8.0 could result in excess hydrogen generation in containment and invalidate the conclusions of the MHA.~~

~~TSP satisfies Criterion 3 of 10 CFR 50.36(c)(2).~~

~~LCO~~

~~The quantity of TSP placed in containment is designed to adjust the pH of the sump water to be between 7.0 and 8.0 after a LOCA. A pH > 7.0 is necessary to prevent significant amounts of iodine released from fuel failures and dissolved in the recirculation water from converting to a volatile form and evolving into the containment atmosphere. Higher levels of airborne iodine in containment may increase the release of radionuclides and the consequences of the accident. A pH > 7.0 is also necessary to prevent SCC of austenitic stainless steel components in containment. SCC increases the probability of failure of components.~~

BASES

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~~LCO (continued)~~

~~The pH needs to remain < 8.0 to remain within the assumptions of the analysis for post-LOCA Hydrogen concentration in the containment.~~

~~The minimum acceptable amount of TSP is that weight which will ensure a sump solution pH ≥ 7.0 after a LOCA, with the maximum amount of water at the minimum initial pH possible in the containment sump; a maximum acceptable amount of TSP is that weight which will ensure a sump solution pH of ≤ 8.0 with a minimum amount of water at a maximum initial pH.~~

~~The TSP is stored in wire mesh baskets placed inside the containment at the 590 ft elevation. Any quantity of TSP between 8,300 and 11,000 lb. will result in a pH in the desired range.~~

~~APPLICABILITY~~

~~In MODES 1, 2, and 3, the PCS is at elevated temperature and pressure, providing an energy potential for a LOCA. The potential for a LOCA results in a need for the ability to control the pH of the recirculated coolant.~~

~~In MODES 4, 5, and 6, the potential for a LOCA is reduced or nonexistent, and TSP is not required.~~

~~ACTIONS~~

~~A.1~~

~~If it is discovered that the TSP in the containment building is not within limits, action must be taken to restore the TSP to within limits.~~

~~The Completion Time of 72 hours is allowed for restoring the TSP within limits, where possible, because 72 hours is the same time allowed for restoration of other ECCS components.~~

~~B.1 and B.2~~

~~If the TSP cannot be restored within limits within the Completion Time of Required Action A.1, the plant must be brought to a MODE in which the LCO does not apply. The specified Completion Times for reaching MODES 3 and 4 are those used throughout the Technical Specifications; they were chosen to allow reaching the specified conditions from full power in an orderly manner and without challenging plant systems.~~

BASES

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~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.5.5.1~~

~~Periodic determination of the mass of TSP in containment must be performed due to the possibility of leaking valves and components in the containment building that could cause dissolution of the TSP during normal operation. A Frequency of 18 months is required to determine that ≥ 8,300 lbs and ≤ 11,000 lbs are contained in the TSP baskets. This requirement ensures that there is an adequate mass of TSP to adjust the pH of the post LOCA sump solution to a value ≥ 7.0 and ≤ 8.0.~~

~~The periodic verification is required every 18 months, since determining the mass of the TSP baskets is only feasible during outages, and normal fuel cycles are scheduled for 18 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the mass of TSP placed in the containment building.~~

~~SR 3.5.5.2~~

~~Periodic testing is performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. Satisfactory completion of this test assures that the TSP in the baskets is "active."~~

~~Adequate solubility is verified by submerging a representative sample of TSP from one of the baskets in containment in un-agitated borated water heated to a temperature representing post-LOCA conditions; the TSP must completely dissolve within a 4 hour period. The test time of 4 hours is to allow time for the dissolved TSP to naturally diffuse through the un-agitated test solution. Agitation of the test solution during the solubility verification is prohibited, since an adequate standard for the agitation intensity (other than no agitation) cannot be specified. The agitation due to flow and turbulence in the containment sump during recirculation would significantly decrease the time required for the TSP to dissolve.~~

~~Adequate buffering capability is verified by a measured pH of the sample solution, following the solubility verification, between 7.0 and 8.0. The sample is cooled and thoroughly mixed prior to measuring pH.~~

BASES

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~~SURVEILLANCE REQUIREMENTS~~ ~~SR 3.5.5.2 (continued)~~

~~The quantity of the TSP sample, and quantity and boron concentration of the water are chosen to be representative of post-LOCA conditions.~~

~~A sampling Frequency of every 18 months is specified. Operating experience has shown this Surveillance Frequency to be acceptable.~~

~~REFERENCES 1. FSAR, Section 6.4~~