NMC
Committed to Nuclear Excellence

Palisades Nuclear Plant
Operated by Nuclear Management Company, LLC

April 1, 2005

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: One-Time Extension to Technical Specification Action Completion Time for Restoration of a Service Water Train to Operable Status

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. NMC proposes a one-time extension to the completion time for restoration of a service water train to operable status in Technical Specification (TS) 3.7.8, "Service Water System." This proposed change is needed to support rebuilding service water pump P-7C. The proposed provision to apply the extended completion time would expire upon start-up from the spring 2006 refueling outage.

Enclosure 1 provides a detailed description of the proposed change, background and technical analysis, No Significant Hazards Consideration Determination, 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.

NMC requests approval of this proposed license amendment by September 1, 2005. NMC further requests a 30-day implementation period following amendment approval.

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

A001

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

Summary of Commitments

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

1. NMC will perform inservice testing of the remaining operable service water pumps within one week prior to pump rebuild, as a preparatory action to verify operability.

2. NMC will perform a readiness review with a multi-disciplined team prior to performing the P-7C rebuild to ensure satisfactory condition of plant equipment.

3. NMC will ensure that the structures, systems, and components (SSCs) associated with the TS LCO 3.7.8, excluding those SSCs associated with the service water pump P-7C maintenance, will be maintained operable during the extended TS action completion time. Should SSCs become inoperable, shutdown actions will be initiated in accordance with TS LCO 3.7.8 required action C.1 to enter TS LCO 3.0.3, if applicable, or to initiate plant shutdown with action equivalent to TS LCO 3.0.3.

4. NMC will ensure that the SSCs, associated with the left train (1-1 emergency diesel generator) in the following Technical Specifications will be maintained operable during the extended TS action completion time, or plant shutdown actions will be initiated that are equivalent to TS LCO 3.0.3:

TS LCO 3.6.6	Containment Cooling System
TS LCO 3.7.7	Component Cooling Water System
TS LCO 3.8.9	Distribution Systems – Operating

5. NMC will designate the following equipment as "Protected Equipment" and control the protected equipment in accordance with the applicable site procedure during the extended TS action completion time:

Left Train	Right Train
Service Water Pump P-7B	Service Water Pump P-7A
Containment Spray Pump P-54B	Emergency Diesel Generator 1-2
Containment Spray Pump P-54C	2400 Volt Bus 1D
Emergency Diesel Generator 1-1	
2400 Volt Bus 1C	
Component Cooling Water Pump, P-52A	

6. NMC will conduct tours twice per shift during the extended TS action completion time. The tours will be conducted to monitor the condition of the following components:

   - Service Water Pumps P-7A and P-7B,
   - Traveling Screens F-4B and F-4C, and
   - Screen Wash Pump P-4

7. NMC will ensure no planned maintenance is performed in the switchyard during the extended TS action completion time.

8. NMC will minimize the duration of maintenance on service water pump P-7C during the extended TS action completion time as much as practical by using a 24-hour work schedule, dedicated project management, and dedicated support for the activity.

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 1, 2005.

W. [signature] FOR DAN J MALONE

Daniel J. Malone
Site Vice-President, Palisades Nuclear Plant
Nuclear Management Company, LLC

Enclosures (3)

cc: Project Manager, Palisades, USNRC
Administrator, Region III, 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. The proposed change would be a one-time extension to the completion time for restoration of a service water train to operable status in Technical Specification (TS) 3.7.8, "Service Water System." This proposed change is needed to support rebuilding service water pump P-7C.

2.0 PROPOSED CHANGE

TS LCO 3.7.8 requires two service water system (SWS) trains to be operable to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst single active failure occurs coincident with the loss of offsite power. Whenever one or more SWS trains are inoperable (Condition A), Action A.1 requires restoration of both trains to operable status within 72 hours.

NMC proposes to change TS LCO 3.7.8 by adding a note that extends, on a one-time basis, the TS action completion time for an additional 96 hours. This would allow 168 hours to perform the service water pump P-7C rebuild. The proposed extension would expire upon start-up from the spring 2006 refueling outage. This provides a reasonable timeframe to complete the rebuild.

The proposed note would read as shown below:

> "The Completion Time for a SWS train to be restored to operable status may be extended to 168 hours for one time to allow performance of service water pump P-7C rebuild. The ability to apply the 168-hour Completion Time will expire upon start-up from the spring 2006 refueling outage."

3.0 BACKGROUND

Current inservice testing data is showing a long-term, differential pressure, decreasing trend in pump performance. If the trend continues, pump performance may fall below the acceptance criteria for differential pressure, prior to the start of the spring 2006 refueling outage. Therefore, pump rebuild is desired prior to the start of the spring 2006 refueling outage to avoid an unnecessary shutdown.

In order to perform the service water pump P-7C rebuild online, an extension of TS LCO 3.7.8 completion time to 168 hours is requested, since the current schedule shows that a rebuild can take up to 168 hours.

4.0 TECHNICAL ANALYSIS

System Description

The SWS provides a heat sink for the removal of process and operating heat from safety related components during a design basis accident (DBA) or transient. During normal operation or a normal shutdown, the SWS also provides this function for various safety related and non-safety related components.

The Palisades Nuclear Plant has three service water pumps, which are designated as P-7A, P-7B and P-7C. The service water pumps are 50-percent capacity, electric motor driven pumps, connected in parallel. The service water pumps take suction from a common intake structure supplied by Lake Michigan. The motors for P-7A and P-7C are connected to one 2.4 kV bus and the motor for P-7B is connected to a separate 2.4 kV bus. The discharge of the pumps flows into a common header before splitting into three headers, two critical headers for safety-related equipment and one non-critical header for non-safety related equipment.

There are two SWS trains, each associated with a safeguards electrical train. The SWS train associated with the left safeguards train consists of one service water pump, P-7B, associated piping, valves, and controls for the equipment to perform their safety function. The SWS train associated with the right safeguards train consists of two service water pumps, P-7A and P-7C, associated piping, valves, and controls for the equipment to perform their safety function.

Normal Operation

The SWS supplies three groups of loads. The major loads are as follows:

1. Critical loads inside containment, including containment air coolers (CACs) VHX-1, VHX-2, and VHX-3.

2. Critical loads outside containment, including diesel generator 1-1 and diesel generator 1-2, component cooling heat exchangers E-54A and E-54B, engineered safeguards room coolers VHX-27A and VHX-27B, control room heating, ventilation and air conditioning coolers, VC-10 and VC-11.

3. Non-critical loads in the turbine building.

Each of these groups of loads can be cooled by the flow from one service water pump. During normal operation, two service water pumps can provide the required flow for all three groups of loads.

Post-DBA Operation

Either one or two service water pumps are required to provide post-accident cooling, depending on the accident events. If off-site power sources are lost, all pump motors are automatically supplied with power from the emergency diesel generators with one service water pump, P-7B, supplied by diesel generator 1-1 and two service water pumps, P-7A and P-7C, supplied by diesel generator 1-2.

One hundred percent of the required SWS post accident cooling capability can be provided by any two SWS pumps if SWS flow, either to the non-critical header or to the critical loads inside the containment, is capable of being isolated. One hundred percent of the required SWS post accident cooling capability can be provided by any one SWS pump if SWS flow, both to the non-critical header and to the critical loads inside the containment, are capable of being isolated.

The capability to isolate SWS flow to the non-critical SWS header requires isolation valve, CV-1359, to be operable. The capability to isolate SWS flow to containment requires one SWS containment isolation valve, either CV-0824 or CV-0847, to be operable. The allowance to isolate SWS flow to containment requires the ability to provide post accident containment cooling without reliance on the CACs.

During a main steam line break (MSLB) or large break loss of coolant accident (LOCA), a minimum of one containment cooling train is required to maintain the containment peak pressure and temperature below the design limits. 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 CAC fan V-4A (Left Safeguards Train). 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, CACs VHX-1, VHX-2, and VHX-3, and their associated safety related fans, V-1A, V-2A, and V-3A (Right Safeguards Train).

With service water pump P-7C inoperable, the right train service water loads including containment cooling, are affected. NMC proposes to minimize risk by implementing compensatory measures that, at a minimum, provide protection for left train components. During the extended TS action completion time, left train (1-1 emergency diesel generator) SSCs in the containment cooling system, component cooling system and electrical distribution systems, can be credited for design basis accident mitigation.

With service water pump P-7C inoperable, 100% of the required post-accident SWS cooling capability remains available with the redundant train maintained operable. The SSCs associated with the remainder of the SWS will be maintained operable in order to preserve system redundancy.

If emergency diesel generator 1-1 were to become inoperable during the extended TS action completion time, then TS 3.8.1 required action B.2 would require that the left train service water pump, P-7B, be declared inoperable within 4 hours. With P-7B inoperable, less than 100% of the required post accident SWS cooling capability would be available, requiring entry into TS LCO 3.0.3, in accordance with TS LCO 3.7.8 required action C.1.

During the requested one-time extension, additional compensatory measures will be implemented to manage risk. The complete list of proposed compensatory measures is provided below.

Compensatory Measures

NMC will implement the following compensatory measures to manage risk. These compensatory measures are to ensure the continued availability of the remaining operable service water pumps and to protect redundant equipment.

1. NMC will perform inservice testing of the remaining operable service water pumps within one week prior to pump rebuild, as a preparatory action to verify operability.

2. NMC will perform a readiness review using a multi-disciplined team prior to performing the P-7C rebuild to ensure satisfactory condition of plant equipment.

3. NMC will ensure that the structures, systems, and components (SSCs) associated with the Technical Specification (TS) LCO 3.7.8, excluding those SSCs associated with the service water pump P-7C maintenance, will be maintained operable during the extended TS action completion time. Should a SSCs become inoperable, shutdown actions will be initiated in accordance with TS LCO 3.7.8 required action C.1 to enter TS LCO 3.0.3, if applicable, or to initiate plant shutdown with action equivalent to TS LCO 3.0.3.

4. NMC will ensure that the SSCs, associated with the left train (1-1 emergency diesel generator) in the following Technical Specifications will be maintained operable during the extended TS action completion time, or plant shutdown actions will be initiated that are equivalent to TS LCO 3.0.3:

TS LCO 3.6.6	Containment Cooling System
TS LCO 3.7.7	Component Cooling Water System
TS LCO 3.8.9	Distribution Systems – Operating

5. NMC will designate the following equipment as "Protected Equipment" and control the protected equipment in accordance with the applicable site procedure during the extended TS action completion time:

Left Train	Right Train
Service Water Pump P-7B	Service Water Pump P-7A
Containment Spray Pump P-54B	Emergency Diesel Generator 1-2
Containment Spray Pump P-54C	2400 Volt Bus 1D
Emergency Diesel Generator 1-1	
2400 Volt Bus 1C	
Component Cooling Water Pump, P-52A	

6. NMC will conduct tours twice per shift during the extended TS action completion time. The tours will be conducted to monitor the condition of the following components:

   - Service Water Pumps P-7A and P-7B,
   - Traveling Screens F-4B and F-4C, and
   - Screen Wash Pump P-4

7. NMC will ensure no planned maintenance is performed in the switchyard during the extended TS action completion time.

8. NMC will minimize the duration of maintenance on service water pump P-7C during the extended TS action completion time as much as practical by using a 24-hour work schedule, dedicated project management, and dedicated support for the activity.

The above compensatory measures more than offset any increased risk associated with P-7C being unavailable for the extended TS action completion time. A quantitative assessment of risk is provided below, along with an overview of the risk management program, which helps ensure risk is minimized during the maintenance activities.

Risk Management Program

The Palisades implementation of a configuration risk management program (CRMP) is addressed in a site-specific administrative procedure. This procedure establishes the requirements for the CRMP program at Palisades. The program requires that the risk associated with on-line maintenance activities be assessed during the scheduling process to assure that high-risk configurations are not inadvertently entered and that appropriate risk management actions are in place for higher risk evolutions. The current program employs three risk categories.

Low or normal risk (green) represents the baseline (zero maintenance) configuration to configurations that allow low and intermediate risk SSCs to be removed from service.

Medium (yellow) risk represents configurations with a high-risk significant train of equipment removed from service. The Intermediate category is entered at a threshold of 2.5 times the baseline core damage frequency.

High Risk (red) category is based on a threshold of 10 times the baseline core damage frequency (CDF). Activities considered "High Risk" (risk achievement worth (RAW) greater than or equal to 10.00 and color red) require Plant Manager written approval to perform. Development of appropriate contingencies is required prior to activities being performed. Activities with a RAW score greater than 10 require approval of the Plant Review Committee prior to entering the condition. Unplanned (emergent) entry into high-risk (red) condition requires an immediate notification to the Plant Manager. In addition, a determination of the most expeditious means to exit the high-risk condition is required.

Entry into unplanned or rescheduled maintenance outages that will result in a system, structure or component being removed from service and has not been previously evaluated, or will occur outside the time frame in which it was previously evaluated, requires a risk assessment to be performed by a Senior Reactor Operator.

In addition, the site-specific administrative procedure provides additional requirements regarding actions to prevent planned entry into high-risk conditions.

The CRMP, as implemented, provides additional assurance that a high-risk configuration will not occur during the proposed extension of the completion time.

Probabilistic Risk Assessment Discussion

Regulatory Guide (RG) 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications, " describes acceptable methods for assessing the nature and impact of proposed TS changes by considering engineering issues and applying risk insights. RG 1.177 describes acceptance guidelines for incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP). When the calculated increase in ICCDP is very small, which is taken as being less than 5.0E-07, the change will be considered. When the calculated increase in ICLERP is very small, which is taken as being less than 5.0E-08, the change will be considered. Assuming P-7C is out-of-service for 168 hours, the bounding risk for this activity results in an ICCDP of 1.85E-08 and an ICLERP of 3.53E-10.

Both the ICCDP and the ICLERP values are considered very small changes per RG 1.177. The risk evaluation concludes that the one-time, 168 hour completion time proposed in this license amendment request results in an ICCDP and an ICLERP that are reasonable based on the guidelines in RG 1.177.

The discussion of risks from external events is based on qualitative assessments using the individual plant examination of external events (IPEEE) results and insights from the internal events probabilistic risk assessment (PRA).

The analysis of risks due to internal fires completed for the IPEEE was based on a blended approach utilizing the fire-induced vulnerability evaluation (FIVE) method developed by the Electric Power Research Institute (EPRI) for qualitative screening followed by the quantification of a simplified PRA model. The Palisades fire analysis used an approach that combined the deterministic evaluation techniques from the EPRI FIVE methodology with classical PRA techniques. The FIVE methodology was used to establish fire boundaries and to evaluate the probability and the timing of damage to components located in a fire area/zone involved in a fire. Based on the results from implementing the FIVE methodology, PRA techniques were then employed to determine the probability of core damage associated with fires within the identified fire areas/zones. Fire areas identified by the fire protection program were used as the basis of the fire areas evaluated by the fire risk analysis. These fire areas were evaluated for further division based on combustible loading and fire-spread potential to identify fire zones within fire areas. The fire areas/zones identified were evaluated and quantified using the fault trees and transient event tree from the individual plant examination (IPE). The fault and event trees were modified to accurately reflect the fire analysis.

The core damage frequency contribution from internal fires for Palisades is 3.31E-05/yr. The dominant contribution to the fire CDF (>89%) is related to five fire areas: cable spreading room (33.5%); main control room (24.4%); 1D switchgear room (14.7%); turbine building (9.3%); and 1C switchgear room (7.6%).

The principle finding of the fire analysis was that there is no area in the plant in which a fire would lead directly to the inability to cool the core. Without additional random equipment failures (unrelated to damage caused by the fire) or human errors, core damage will not occur. As a result, the study concluded that there are no major vulnerabilities due to fire events at the Palisades Nuclear Power Plant. This is primarily because the damage in the important fire areas was to support systems (e.g. AC Power or DC Power) that resulted in the loss of one train of equipment with adequate equipment unaffected on the other train. During the extended TS action completion time for the service water pump P-7C rebuild an operable service water pump will remain available on each train.

A seismic risk assessment was used to assess risks due to seismic events. The risk assessment was a hybrid of the conventional PRA and seismic margins analysis. The seismic analysis has not been updated since originally developed for the IPEEE submittal. A review of the results of the IPEEE submittal indicates that the core damage frequency was 8.88E-06, with a high confidence low probability of failure (HCLPF) of 0.217g peak ground acceleration. There were no specific seismic events identified as dominant contributors to the core damage frequency. Important seismic induced failures identified were; the fire protection system (FPS), main steam isolation valves, diesel generator fuel oil supply, and an undervoltage relay for Bus 1D. Several important random failures were identified in the report as important because of their contribution in combination with seismically induced failures. The random failures identified were: Diesel Generator 1-2, auxiliary feedwater (AFW) pump P-8C, and atmospheric dump valves. The FPS provides an alternate suction source to AFW pumps P-8A and P-8B. As noted, the FPS is an important contributor due to the probability of seismically induced failures of fire protection system components and the condensate storage tank. The increased probability of failure of AFW pumps, P-8A and P-8B, increase the contribution of the service water pumps to the CDF in conjunction with the FPS failures. The importance of the service water pumps is due to their function as alternate sources of water for AFW pump P-8C.

Other external events (high winds, external floods, transportation, etc.) were evaluated consistent with the approach described in NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," dated June 1991. There were no other external events (other than seismic and fire) identified that have an impact on core damage frequency at Palisades.

Based on the above evaluation, the proposed one-time change will have no adverse effect on plant safety.

<u>Quality of Palisades Nuclear Plant PRA Model</u>

The PRA model addresses internal events at full power, including anticipated transients without scram, interfacing system LOCAs outside the containment. NMC has not developed a shutdown PRA model for Palisades. It is an evolution of the original IPE for Severe Accident Vulnerabilities and IPEEE for Severe Accident Vulnerabilities submitted in response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," dated November 23, 1988. The IPE was submitted on January 29, 1993, and the NRC safety evaluation report (SER) was issued February 7, 1996. The IPEEE was submitted on June 30, 1995, and the NRC SER was issued November 29, 1999.

Revisions to the models have been made to maintain the models consistent with plant design changes and operational changes. The changes were made by individuals knowledgeable in risk assessment techniques and methods and familiar with the plant design and operation. The current PRA model of record and the risk assessment performed for this proposed amendment have been documented as calculations.

Current administrative controls include written procedures and review of all model changes, data updates, and risk assessments performed using PRA methods and models. Risk assessments are performed as calculations per the plant administrative procedure by a PRA engineer, reviewed by another PRA engineer, and approved by the programs and analysis supervisor or designee. Procedures, PRA model documentation, and associated records for applications of the PRA models are maintained as controlled documents.

Since the submittal of the original PRA study, the PRA models have been maintained consistent with the current plant configuration such that they are considered "living" models, which reflect the as-built, as-operated plant. The PRA models are updated for different reasons, including plant changes and modifications, procedure changes, accrual of new plant data, discovery of modeling errors, and advances in PRA technology. The update process ensures that the applicable changes are implemented and documented so that risk analyses performed in support of plant operations reflect the current plant configuration, operating philosophy, and transient and component failure history.

The PRA maintenance and update process is described in a risk informed engineering guideline. Model updates are performed at a frequency dependent on the estimated impact of the accumulated changes and as described in the guideline. Guidance to determine the need for a model update is provided in a NMC procedure.

An update to the PRA model is in progress that includes several recent changes to the plant such as the replacement of two half-capacity instrument air compressors, with two redundant full-capacity instrument air compressors, and model enhancements including a reactor coolant pump seal LOCA model. The revision to the model also includes an update to the original IPE plant damage state and containment event tree models that will provide a mechanism to quantify large early release frequency (LERF) based on the current as-built plant. This model was used to provide an estimate of the changes in LERF resulting from the proposed extension to the TS action completion time for service water pump P-7C.

In order to help ensure the quality and validity of the PRA, the Combustion Engineering Owners Group (CEOG) conducted an independent assessment in May 2000. The assessment report, CE-NPSD-1194-P, "Probabilistic Safety Assessment Peer Review Report," was issued September 2000. Items that were identified in the assessment were labeled as level A and level B. The level

A items were considered extremely important and necessary to address. All of the items resulting from the assessment designated as level A have been dispositioned. The level B items were considered important and necessary to address, but may be deferred until the next PRA update. Approximately 80% of the category B items have been resolved with the remainder scheduled for completion by August 2005. The issues identified by the peer review involving model documentation are being addressed as each individual PRA document is reviewed and approved under Palisades administrative procedures. Other changes involving guidance documents and administrative processes used for model updates are in the process of being updated. The issues identified by the peer review in these areas have been reviewed and determined not to have any impact on this submittal, and so deferral of completion of these items is acceptable for this application of the PRA model. All other peer review items which impact the PRA model have been addressed and are reflected in this proposed amendment.

A 10 CFR 50.65(a)(4) implementation peer assessment was performed during November 2003. The peer review was performed in accordance with CEOG Task-2024, "CEOG Assessment of Palisades Plant Implementation of 10 CFR 50.65(a)(4)," dated December 23, 2003. Based upon this assessment, it is concluded that Palisades Nuclear Plant was in general compliance with the requirements of maintenance rule 10 CFR 50.65 (a)(4) for all modes of power operations and shutdown activities, as presented in NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated July 2000. The strengths and weaknesses were entered into the plant corrective action process and have been evaluated and resolved.

The following computer programs, which are used to process PRA model inputs, are verified, validated and classified in accordance with NMC procedures. Software quality assurance plans were developed in accordance with a NMC procedure. These plans provide for software verification and validation to ensure the software meets the software requirement specifications and functional requirements, and may include a comparison of results generated to the results generated from previously approved software.

SQAP-PL-0070, Rev 2.0, CAFTA & SAPHIRE
SQAP-PL-0062, Rev 2.0, EOOS
SQAP-PL-0285, Rev 2.0, MAAP
SQAP-PL-0290, 11/8/04, SHIP

Therefore, based on the information above, the scope, level of detail and quality of the Palisades PRA is commensurate with the proposed license amendment for which it is intended and the role the PRA results play in the integrated decision process.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

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 the extended Technical Specification action completion time is not an accident initiator. Therefore the probability is not increased significantly.

   The proposed amendment does not involve a significant increase in the consequences of an accident previously evaluated. With service water pump P-7C inoperable, 100% of the required post-accident SWS cooling capability remains available with the redundant train maintained operable. A risk analysis was performed to show that the consequences are not significantly increased. The compensatory measures provide additional assurance that there is no significant increase in the consequences of an accident associated with extending the Technical Specification action completion time for the service water system for an additional 96 hours.

   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.

   The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment only extends the Technical Specification action completion time and does not involve a physical alteration of any system, structure or component (SSC), or change in the way any SSC is operated. The proposed amendment does not involve operation of any required 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.

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. With service water pump P-7C inoperable, 100% of the required post-accident service water system cooling capability remains available with the redundant train maintained operable. Therefore, there is no significant reduction in the margin of safety.

   Based on the availability of redundant systems, the compensatory measures that will be taken, and the low probability of an accident that could not be mitigated by the available systems, the proposed amendment would not involve a significant reduction in a margin of safety.

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

The proposed change is a one-time extension to the Technical Specification action completion time. It does not affect the design basis of the plant.

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.

7.0 REFERENCES

By letter dated January 6, 2005, South Texas Project Electric Generating Station submitted a similar license amendment request for South Texas Unit 1. By letter dated January 10, 2005, the NRC approved the license amendment request.

ENCLOSURE 2

LICENSE AMENDMENT REQUEST: ONE-TIME EXTENSION TO TECHNICAL SPECIFICATION ACTION COMPLETION TIME FOR RESTORATION OF A SERVICE WATER TRAIN TO OPERABLE STATUS

REVISED TECHNICAL SPECIFICATION PAGE
3.7.8-1
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
3.7.8-1	3.7.8-1

3.7 PLANT SYSTEMS

3.7.8 Service Water System (SWS)

LCO 3.7.8 Two SWS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SWS trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72* hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours
C. Less than 100% of the required post accident SWS cooling capability available.	C.1 Enter LCO 3.0.3.	Immediately

*The Completion Time for a SWS train to be restored to operable status may be extended to 168 hours for one time to allow performance of service water pump P-7C rebuild. The ability to apply the 168-hour Completion Time will expire upon start-up from the spring 2006 refueling outage.

ENCLOSURE 3

LICENSE AMENDMENT REQUEST: ONE-TIME EXTENSION TO TECHNICAL SPECIFICATION ACTION COMPLETION TIME FOR RESTORATION OF A SERVICE WATER TRAIN TO OPERABLE STATUS

MARK-UP OF TECHNICAL SPECIFICATION PAGE
3.7.8-1
(showing proposed changes)
(additions are highlighted; deletions are strikethrough)

1 Page Follows

3.7 PLANT SYSTEMS

3.7.8 Service Water System (SWS)

LCO 3.7.8 Two SWS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SWS trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72* hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Less than 100% of the required post accident SWS cooling capability available.	C.1 Enter LCO 3.0.3.	Immediately

*The Completion Time for a SWS train to be restored to operable status may be extended to 168 hours for one time to allow performance of service water pump P-7C rebuild. The ability to apply the 168-hour Completion Time will expire upon start-up from the spring 2006 refueling outage.