PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

MAR 2 3 2004
LR-N04-0137

PSEG
Nuclear LLC

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

INSERVICE INSPECTION PROGRAM
RELIEF REQUEST HC-RR-I2-023
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSES NPF-57
DOCKET NOS. 50-354

Pursuant to 10 CFR 50.55a(a)(3), PSEG Nuclear, LLC (PSEG) requests authorization by the Nuclear Regulatory Commission (NRC) of the enclosed Hope Creek Generating Station relief request HC-RR-I2-023.

Approval for relief is requested in accordance with 10CFR50.55a(a)(3)(ii) hardship or unusual difficulty without compensating increase in the level of quality or safety. Performance of the system leakage test at the nominal operating pressure associated with 100% rated reactor power would result in potential increased personnel radiation exposure, as well as presenting unusual difficulty due to challenges specifically associated with reactor pressure control and satisfying Technical Specification requirements.

The attachment to this letter includes the proposed alternative and supporting justification for the relief. Based on the evaluation contained in the attachment, PSEG has concluded that the proposed alternative provides an acceptable level of quality and safety. Accordingly, this proposal satisfies the requirements of 10CFR50.55a(a)(3)(ii).

PSEG Nuclear requests expedited authorization of the attached relief request by March 26, 2004 in order to support the Hope Creek mini-outage currently in progress.

The NRC has previously granted similar relief to Nebraska Public Power District, Cooper Nuclear Station and Nuclear Management Corporation, Monticello Nuclear Generating Plant.

If you have any questions, please contact Mr. Howard Berrick at 856-339-1862.

Sincerely,

Steven R. Mannon
Manager – Nuclear Safety and Licensing

A047

Attachment: ISI Relief Request HC-RR-I2-023

C Mr. H. Miller, Administrator
Regional Administrator - Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

U. S. Nuclear Regulatory Commission
ATTN: Mr. John Boska
Licensing Project Manager – Hope Creek
Mail Stop 08B1
Washington, DC 20555-001

USNRC Senior Resident Inspector – Hope Creek (X24)

Mr. K. Tosch, Manager IV
Bureau of Nuclear Engineering
P. O. Box 415
Trenton, NJ 08625

ASME Code Component Affected

The following Code Class1 Components are affected by this relief request:
- Main Steam Safety Relief Valves (SRVs) "J" and "P" Assemblies
- Control Rod Drive Mechanism (CRDM) "O" Ring Replacement, planned for 7 CRDMs.

Applicable ASME Code Edition and Addenda:

ASME Section XI, 1989 Edition, is the code of record for Hope Creek Nuclear Generating Station's Second Ten-Year ISI Program Interval.

Repair/replacement activities are conducted in accordance with the 1995 Edition with 1996 Addenda of ASME Section XI, Division 1 utilizing ASME Code Case N389-1.

Applicable Code Requirement:

The 1995 Edition of American Society of Mechanical Engineers (ASME) Section XI with the 1996 Addenda, paragraph IWA-5120(a) states: "Items subjected to repair/replacement activities shall be pressure tested when required by IWA-4500."

Paragraph IWA-4540(c) states: "Mechanical joints made in installation of pressure retaining items shall be pressure tested in accordance with IWA-5211(a)."

Paragraph IWA-5211(a) states: "A system leakage test conducted during operation at nominal operating pressure, or when pressurized to nominal operating pressure and temperature."

Paragraph IWB-5210(b) states: "The system pressure tests and visual examinations shall be conducted in accordance with IWA-5000 and this Article. The contained fluid in the system shall serve as the pressurizing medium."

ASME Section XI, Table IWB-2500, Examination Category B-P, Item B15.10, requires a system leakage test of the Reactor Pressure Retaining Boundary (such as after component replacement).

IWB-5221(a) requires that the system leakage test shall be conducted at a test pressure not less than the nominal operating pressure associated with 100% rated reactor power.

Reason for Request

PSEG Nuclear, LLC (PSEG) Hope Creek Generating Station completed their eleventh refueling outage in May 2003. During the refueling outage PSEG completed the system leakage test required by American Society of Mechanical Engineers (ASME) Section XI, Table IWB-2500-1, Category B-P, Item 15.10 and 10 CFR Part 50 Appendix G, Section IV.A.2.d. Subsequent to the restart, the "J" and "P" main steam SRVs have indicated leakage, as determined by higher than normal temperatures in their respective discharge tailpipes. In addition, higher than anticipated leakage from several CRDM flanges has been noted.

PSEG has decided to conduct a planned unit shutdown and enter a maintenance outage to replace the affected SRV assemblies. The SRV assemblies are connected to the main steam piping with a bolted, mechanical joint. In addition several CRDM O-rings will be replaced. Replacing SRV assemblies is considered a Repair-Replacement activity under the rules of ASME Section XI, 1995 Edition with the 1996 Addenda. Following repair-replacement, a system leakage test is required by IWA-4540(c). O-ring replacement on CRDMs is considered maintenance and, of itself, is exempt from the ASME Section XI pressure testing requirements, but since it will involve the opening and closing of the Reactor Coolant Pressure Boundary (RCPB), a system leakage test would nonetheless be performed to assure leakage integrity. The system leakage test at the nominal pressure associated with the reactor at 100% power would be approximately 1005 psig.

PSEG has identified three methods for performing the system leakage test on the mechanical joints associated with the repair-replacement activity that meet the requirements identified above. Several conditions associated with such testing represent an imposition on personnel safety and challenges to the normal mode and manner of equipment operation.

Method No. 1 would perform the pressure test and VT-2 exam during normal startup procedures. During normal startup with normal power ascension, nominal operating pressure of 1005 psig is reached at a reactor power level of approximately 100%. If access to containment were permitted at this power level, personnel would be exposed to excessive radiation levels, including significant exposure to neutron radiation fields, which is contrary to current station ALARA practices.

Establishing the 1005 psig test condition at a more moderate power level (e.g. during plant startup at approximately 7% reactor power) and in the manner needed to address radiation concerns would require altering the normal operational mode of the steam pressure control system.

During the performance of plant startup procedures, the Electro-Hydraulic Control (EHC) pressure regulator set point is established within normal operational ranges (approximately 920 psig). The primary function is to regulate the main steam system pressures as sensed near the inlet of the high-pressure turbine. Reactor pressure control at the nominal 1005 psig is achieved at higher reactor power levels as a function of the pressure control system and the induced differential pressure across the main steam isolation valves and main steam piping.

While it is technically feasible to manipulate these controls to establish the nominal system pressure of 1005 psig at lower power levels, this process may introduce new operational challenges and may require additional analyses. Although reactor pressure during low-power operation is sometimes raised from 920 psig to 950 psig to perform scram-time testing, it has not been previously raised to 1005 psig under these conditions. The lack of experience and predictability of setting pressure regulators outside the normal range of operation could adversely impact personnel and reactor safety.

Method No. 2 implements the use of the reactor pressure boundary leakage test which meets the requirements of Table IWB-2500-1, Category B-P, Item 15.10: the reactor pressure vessel (RPV) is filled with coolant and the steam lines are flooded to provide a water-solid condition. Use of this method would result in multiple operational challenges.

During a maintenance outage, decay heat and the reactor recirculation pumps would provide pressurization for the test. To support the pressurization evolution, the normal decay heat removal system, residual-heat removal (RHR) shutdown cooling, would be required to be removed from service and isolated from the vessel to be pressurized. This system automatically isolates at 82 psig. Thus, the remaining system available for decay heat removal is the reactor water cleanup system (RWCU).

The application of the ANSI /ANS-1994 decay heat code results in a significant level of decay heat load. The ratio of decay heat input versus the heat removal capacity provided by RWCU is approximately 4:1. Therefore, the decay heat generated by the reactor core will surpass the capacity of RWCU. The heat up rate of the vessel water will cause the temperatures to surpass 212°F prior to the initiation of the inspections. This would violate Hope Creek Generating Station Technical Specifications.

Method No. 2 would present several operational challenges. The pressure increase would be obtained by balancing the flow into the vessel, which is provided by the control rod drive (CRD) system, with the flow out of the vessel provided by the RWCU system via the drain flow control valve and flow controller. This is the method used during refueling outages to complete the

RPV system leakage test. A failure of a non-safety related component, such as the drain valve or flow controller, would cause the interruption of drain flow and would cause the RPV pressure to increase. The RPV pressure would increase until operator action would require the operating CRD pump to be tripped.

Due to the amount of decay heat being generated and the RWCU systems heat removal capacity, it is questionable whether the RPV would depressurize and may in fact continue to pressurize until further operator action would be required to depressurize the RPV. Operator actions may include one or more of the following: reestablishing RWCU drain flow if the failure mechanism was no longer present; opening the main steam line drain valves, SRVs, or head vent line. Any of these actions could cause a rapid depressurization transient on the RPV.

Extensive valve manipulations, system lineups, and procedural controls are required in order to heat up and pressurize the primary system to establish the necessary test pressure, during plant outage conditions, without the withdrawal of control rods. This test is expected to take greater than 24 hours of outage time, and the additional valve lineups and system reconfigurations necessary to support this test impose an additional challenge to the affected systems. A normal plant startup then occurs, after completion and subsequent recovery from the test procedure.

Method No. 3 would maintain the RPV at its normal level of ± 35 inches and use decay heat to produce sufficient steam pressure to conduct the test at nominal operating temperature. At the projected time of shutdown for the March 2004 maintenance outage, PSEG will have a run time of approximately 10 months since startup from the Cycle 11 refueling outage. The maintenance of SRV assemblies and CRD mechanisms is projected to be complete within 11 days after plant shutdown. While the decay heat load is too high for the water-solid method discussed above, there is not sufficient decay heat available to perform the test within a reasonable time period to support completion of the maintenance outage. It would require approximately 37 hours, after 5 days of decay, to reach the pressure of 1005 psig needed to perform the test required by the Code based upon decay heat projections and the current schedule is for approximately 11 days.

During a similar but much shorter 2003 maintenance shutdown to replace gaskets on the SRV assemblies, the decay heat method was used to pressurize the system for testing. The testing, although performed successfully, proved to be an extreme challenge to the operators to maintain level, pressure and temperature rate.

Each of the methods discussed above presents a hardship or unusual difficulty to PSEG.

Proposed Alternate and Basis for Use

Pursuant to 10 CFR 50.55a(a)(3)(ii), compliance with the required system leakage test under IWA-4540(c) would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety.

PSEG proposes to perform a VT-2 examination on the mechanical joints of the SRV assemblies and CRD flanges during the normal operational start-up sequence at a minimum of 900 psig following a 10 minute hold time (for uninsulated components) in lieu of the nominal operating pressure associated with 100% reactor power (approximately 1005 psig).

The use of the normal method of Reactor start-up represents the safest approach to controlling the reactor pressurization and heat-up evolution. Application of this alternative test maintains reasonable levels of personnel safety and reduces the opportunity for the introduction of undesirable operational challenges.

Requiring normal operating temperature and pressure sub-critical core conditions prior to conducting a normal plant start-up will result in additional thermal cycling of the reactor vessel. This would represent an unnecessary challenge to the vessel from both a fatigue usage and brittle fracture margin perspective.

Examinations of the affected portions of the RCPB are reasonably expected to be performed successfully, even under core critical conditions, since access and ambient temperatures are not significantly different prior to and following criticality. Radiation exposure for the small scope of examinations performed at low power levels is not a concern.

Maintaining applicable Mode conditions (i.e. no core criticality) to conduct this pressure test of the RCPB can result in an unnecessary cycling of the RCPB and unnecessary operation of associated components due to Mode limitations. This can contribute to degradation of the structural components, which is contradictory to the goal of safe operation.

While PSEG does not expect that leakage will occur, any leakage at the bolted connection would be related to the differential pressure across the connection. A reduction in test pressure is less than 10%, and is not, therefore, expected to affect the ability of the VT-2 examination to detect leakage at the bolted connection.

In the event that leakage would occur at the mechanical joints at higher pressures associated with 100% reactor power, leakage from these mechanical connections would be detected by the drywell monitoring systems, which include drywell pressure monitoring, the containment atmosphere monitoring (CAM) system, and the drywell floor drain sumps. Leakage monitoring is required by PSEG Hope Creek Technical Specifications.

In addition, if there is an unplanned shutdown with a drywell entry before the next refueling outage, another inspection of these bolted connections will be performed to look for any evidence of leakage.

This alternative method for a system leakage test is particularly applicable for the PSEG Hope Creek maintenance mini-outage, which is of limited scope, and where the only components on the primary system that are being replaced are the main steam "J" and "P" SRV assemblies and CRDM O-rings attached via mechanical connections.

PSEG believes this alternative will provide an acceptable verification of the integrity of the mechanical joints without unnecessary radiation exposure and operational challenges.

Duration of Proposed Alternative

PSEG requests NRC authorization to perform the proposed alternative test on a "one-time-only" basis for the system leakage test following repair/replacement activities on the mechanical joints of SRVs "J" and "P" and CRDM O-rings during the March 2004 planned maintenance.

Precedence

1. Nebraska Public Power District (NPPD), Cooper Nuclear Station, [Letter dated February 26, 1998, Docket No. 50-298, TAC No. MA 0677].

2. Nuclear Management Corporation (NMC) Monticello Nuclear Generating Plant (MNGP), [Letter dated June 13, 2003Docket No. 50-263, TAC No. MB9538, ADAMS Accession Number ML031640464].