UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PLAN

REQUEST FOR RELIEF NO.R 15

GPU NUCLEAR CORPORATION

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NUMBER: 50-219

1.0 INTRODUCTION

The Technical Specifications for Oyster Creek Nuclear Generating Station state that the inservice inspection (ISI) and testing of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Paragraph 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) on the date 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable edition of the ASME Code, Section XI, for Oyster Creek Nuclear Generating Station during the third 10-year ISI interval, is the 1986 edition. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval. Pursuant to 10 CFR 50.55a(g)(5), if the licensee determines that conformance with an examination requirement of Section XI of the ASME Code is not practical for its facility, information shall be submitted to the Commission in support of that determination and a request made for relief from

the ASME Code requirement. After evaluation of the determination, pursuant to 10 CFR 50.55a(g)(6)(i), the Commission may grant relief and may impose alternative requirements that are determined to be authorized by law, will not endanger life, property, or the common defense and security, and are otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed.

By letter dated September 12, 1996, GPU Nuclear Corporation, the licensee for Oyster Creek Nuclear Generating Station, requested a relief from the requirements of the 1986 edition of the ASME Boiler and Pressure Vessel Code, Section XI, in regard to corrective measures for system pressure tests, as stated in Subsection IWA-5250 (a)(2). The September 12, 1996, letter supersedes in its entirety the licensee's earlier request dated August 19, 1996.

The staff has reviewed and evaluated the licensee's request and the supporting information on the proposed relief request R15 for Oyster Creek Nuclear Generating Station pursuant to the provisions of 10 CFR 50.55a(a)(3)(i).

2.0 EVALUATION

1986 ASME Code Section XI Requirement: Subsection IWA-5250 (a)(2) states that the source(s) of leakage detected during the conduct of a system pressure test shall be located and evaluated by the Owner for corrective action. For leakage occurring at a bolted connection, the bolting shall be removed, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3100.

Licensee's Code Relief Request: (as stated)

"This relief request is intended to authorize alternatives to the removal of bolting at pressure retaining connections when evidence of leakage is detected during system pressure testing. Relief is requested from compliance with IWA-5250(a)(2), for removal of bolting from leaking bolted connections. GPU Nuclear would implement the alternative of performing corrective measures, as deemed necessary by GPU Nuclear Engineering Staff evaluations."

Licensee's Basis for Relief: (as stated)

"Compliance with the ASME Code requirement for the removal of bolting has resulted in undue hardship to the plant without a compensating increase in the level of quality or safety. Removing a system or subsystem from service or potentially shutting down the plant in order to remove bolting that has not been degraded by leakage is impractical. Experience has shown that equipment can be damaged, personnel exposure can be excessive, and components designed for interference fit make it impossible to remove studs when a bonnet is in place. The existence of a leak does not justify the use of such potentially extreme measures. There are many other factors that must be considered in order to make a responsible and timely decision. Considerations that are important in assessing leakage through pressure retaining bolted connections include: location of the leak in the plant or system; time in the plant cycle, leaking medium, materials exposed to the leak, Technical Specification limitations, ability to monitor or isolate the leak, and the ability to redirect or capture the leak.

"Numerous industry studies on the degradation and failure mechanisms of bolting in nuclear power plants have been documented. These studies have quantified the experience of bolting failures and identified the primary failure mechanisms associated with bolt degradation. The documents have shown that bolt failures have primarily occurred in pressurized water reactors, in both ambient and elevated temperature environments. The following three causes of bolting failures have been identified and have been evaluated for any possible impact at the Oyster Creek facility:

"1. Stress Corrosion Cracking (SCC): This mechanism requires a wet or humid environment, high pre-load stresses, use of lubricants containing molybdenum disulfide (MoS_2), and/or improper heat treatment of material.

"2. Fatigue: This failure is primarily induced by improper preload torquing.

"3. Borated Water: This is a chemical attack caused by borated water leakage.

"GPU Nuclear has examined the conditions which are directly associated with the failure of bolts and evaluated their applicability to Oyster Creek. Records of operating history, maintenance procedures, Inservice Inspection Program results, and material specifications for susceptibility to corrosion have been evaluated. GPU Nuclear has determined that the present scope of ASME XI NDE [nondestructive examination] examination requirements for post bolted flange leakage to be undesirable when the likelihood of these failure modes is considered with the increase in personnel radiation exposure which would result.

"1. SCC: The majority of bolting material installed at Oyster Creek meets ASTM [American Society for Testing and Materials] A 193, grade B7 specifications, except for the Control Rod Drive (CRD) bolts which are discussed below. This is a chromium-molybdenum material which is considered low strength and generally not susceptible to stress corrosion cracking. All bolting materials have been purchased in accordance with the GPU Nuclear Quality Assurance Plan.

"Approved lubricants are controlled by procedures. The primary lubricant at Oyster Creek is Chesterton, a nickel based lubricant that does not contain MoS_2.

"2. Fatigue: Fasteners at the Oyster Creek site are typically torqued to a preload stress of 50% (±5%) of the yield strength. Exceeding this limit requires an Engineering Evaluation. This has been the standard practice at Oyster Creek, and is closely monitored by the Engineering Division.

"3. Borated Water: Unlike pressurized water reactors, Oyster Creek does not use borated water in its primary coolant system. The reactor coolant system is frequently monitored for chemical composition and contaminants. No corrosion inducing additives are used or allowed.

It is the GPU Nuclear position that chemical corrosion is not the cause of bolt failure in the Oyster Creek Class 1 systems. Additionally, the atmosphere in the drywell during operation is required by Technical Specifications to be inerted with nitrogen. This starves the bolted connection of oxygen, mitigating the process of both chemical and stress corrosion cracking.

"4. CRD: CRD [control rod drive] housing leakage has been primarily noted at Oyster Creek when the primary system was pressurized prior to heat-up and/or CRD Scram Time Testing. This leakage drastically decreased when the vessel metal temperature reached the normal operating band and the gaskets and o-rings were properly seated by the required scram time tests. This change in leakage has been documented and evaluated by the vendor and found to be acceptable. Subsequent VT-1 examinations of the CRD bolts during normal maintenance evolutions have revealed no degradation caused by corrosion.

"During the exchange of CRDs, the bolts are cleaned and ASME XI examinations are performed. GPU Nuclear utilizes these examinations as opportunities to evaluate the bolts for degradation. The sample of bolts that is inspected is a sufficient representation to allow identification of degradation trends. In previous refueling outages since the plant went on line in 1969, there have been scheduled CRD exchanges. In the 27 years of operation, hundreds of CRD inspections have revealed no reports of CRD bolt failures due to corrosion. GPU Nuclear will continue to inspect the bolts during these periods of opportunity and also will employ alternative methods of examination if the need is justified. Although there is a small possibility that one of the eight CRD bolts might fail due to a design flaw, it is highly unlikely that a CRD would separate from its housing flange. As few as three uniformly distributed bolts can support full CRD loading while remaining within the stress limits identified by ASME Codes.

"The primary coolant system pressure test is done at reduced temperatures. It has been observed that the total amount of unidentified leakage in the drywell decreases significantly for the first few weeks of operation following a refueling outage. Inspections of drywell components made immediately after shutdown for refueling have repeatedly identified minimal or no leakage from the primary coolant system. The leakage found during the low temperature system pressure test is much greater than the leakage identified when the system is at normal operating temperatures."

Licensee's Proposed Alternative: (as stated)

"The source of all leakage detected by VT-2 examination during a system pressure test shall be evaluated to determine the susceptibility of the bolting to corrosion and potential failure. This evaluation will consider the following variables at a minimum:

1. Location of leakage
2. History of leakage
3. Fastener materials
4. Evidence of corrosion with the connection assembled
5. Corrosiveness of the process fluid
6. History and studies of similar fastener material in a similar environment
7. Other components in the vicinity that may be degraded due to the leakage

"When the evaluation of the above variables is concluded and the evluation determines that the leaking condition has not degraded the fasteners, then no further action is necessary. However, reasonable attempts to stop the leakage shall be taken.

"If the evaluation of the variables above indicates the need for further evaluation, or no evaluation is performed, then a bolt in the leakage path will be removed. The bolt will receive a visual VT-1 examination, and be evaluated in accordance with IWB-3140, 'Inservice Inspection Visual Examinations'. This visual VT-1 examination may be deferred to the next outage of sufficient duration if the evaluation supports continued service. When the removed bolting shows evidence of rejectable degradation, all remaining bolts shall be removed and receive a visual VT-1 examination and evaluation in accordance with IWB-3140."

3.0 <u>CONCLUSION</u>

The licensee has provided adequate justification and has sound engineering basis in requesting the above relief from the Code. In accordance with the 1986 edition of the ASME Code, Section XI, when leakage occurs at bolted connections, all bolting is required to be removed for VT-3 visual examination. In lieu of the Code-required removal of bolting to perform a VT-3 visual examination, the licensee has proposed to perform an evaluation of the bolted connection to determine the susceptibility of the bolting to corrosion and the potential for failure. If the initial evaluation indicates the need for a more in-depth evaluation, a bolt in the leakage path will be removed, VT-1 examined, and evaluated in accordance with IWB-3140. The VT-1 visual examination of the bolt proposed by the licensee detects discontinuities and imperfections on the surface of the bolt, including such conditions as cracks, wear, corrosion, or erosion, and is believed to be more conservative than that of the Code-required VT-3 visual examination to evaluate the general mechanical and structural condition of the component. This alternative allows the licensee to utilize a systematic approach and sound engineering judgement, provided that as a minimum, all of the seven evaluation factors listed in the licensee's proposed alternative are considered. As a result, the licensee's alternative to the Code-required removal of bolting at a joint when leakage occurs will provide an acceptable

level of quality and safety, as the integrity of the joint will be maintained. Therefore, the staff concludes that the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

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