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

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OF THE THIRD TEN-YEAR INTERVAL INSERVICE INSPECTION

REQUESTS FOR RELIEF

GPU NUCLEAR CORPORATION

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

The Technical Specifications for Oyster Creek Nuclear Generating Station state that the inservice inspection 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). 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 difficulty 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 preservice 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 inspection interval, subject to the limitations and modifications listed therein. The applicable edition of Section XI of the ASME Code for the Oyster Creek Nuclear Generating Station, third 10-year inservice inspection (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.

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By letter dated February 4, 1993, the licensee, GPU Nuclear Corporation (GPUN) submitted Request for Relief R12 from the requirements of ASME Boiler and Pressure Vessel Code Section XI, subparagraph IWA-5250(a)(2). This subparagraph contains corrective measures for leaking bolted connections. Similarly, in a letter dated February 8, 1993, the licensee requested relief from the hydrostatic testing requirements of subparagraph IWA-5214(a) for components repaired by welding. On February 11, 1993, a conference call was held with the licensee to discuss the requests for relief contained in these submittals. As a result of the conference call, the licensee provided additional information in two letters dated February 11, 1993.

2.0 EVALUATION AND CONCLUSIONS

The staff, with technical assistance from its contractor, the Idaho National Engineering Laboratory (INEL), has evaluated the information provided by the licensee in support of its Request for Relief No. R12, Parts A, B, and C, and the one time schedular request for relief for hydrostatic testing from IWA-5214. Based on the information submitted, the staff adopts the contractor's conclusions and recommendations presented in the Technical Evaluation Summary attached.

Request for Relief R12, Part A, has been denied. The alternatives contained in Request for Relief R12, Part B, and Request for Relief No. R12, Part C, are authorized pursuant to 10 CFR 50.55a(a)(3)(i) provided the proposed alternative actions described by the licensee are followed. In addition, the one time schedular relief request for hydrostatic testing from IWA-5214 is authorized pursuant to 10 CFR 50.55a(a)(3)(i) provided the licensee performs the required hydrostatic test during the next regularly scheduled system hydrostatic test in the current ten-year interval. Although Code Case N-416, "*Alternative Rules for Hydrostatic Testing of Repair or Replacement of Class 2 Piping, Section XI, Division 1*" is acceptable for generic use by reference in Regulatory Guide 1.147, Revision 10, the licensee's intent to use N-416 for schedular relief is not applicable since N-416 applies to Class 2 piping and the system in question is Class 3.

Based on the above, the staff has determined that the requested relief (R12, Parts B and C) provides an acceptable level of quality and safety and may be granted pursuant to 10 CFR 50.55a(a)(3)(i).

Principal Contributor: K. Battige

Date:

ENCLOSURE 2

TECHNICAL EVALUATION SUMMARY
OF THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION
REQUESTS FOR RELIEF
FOR
GPU NUCLEAR CORPORATION
OYSTER CREEK NUCLEAR GENERATING STATION
DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated February 4, 1993, the licensee, GPU Nuclear Corporation, (GPUN) submitted Request for Relief R12 from the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, subparagraph IWA-5250(a)(2). This subparagraph contains corrective measures for leaking bolted connections. Similarly, in a letter dated February 8, 1993, the licensee requested relief from the hydrostatic testing requirements of subparagraph IWA-5214(a) for components repaired by welding. On February 11, 1993, a conference call was held with the licensee to discuss the requests for relief contained in these submittals. As a result of the conference call, the licensee provided additional information in two letters dated February 11, 1993. The Idaho National Engineering Laboratory (INEL) staff has evaluated the subject requests for relief in the following sections.

2.0 EVALUATION

The information provided by the licensee in support of the requests for relief has been evaluated and is documented below. Based on the interval start date of March 15, 1993, the applicable edition of Section XI of the ASME Code for the Oyster Creek Nuclear Generating Station, third 10-year ISI interval is the 1986 Edition.

A. Request for Relief No. R12, (Part A), Paragraph IWA-5250(a)(2), Corrective Measures For Class 1 Bolted Connections on Pumps and Valves

Code Requirement: Paragraph IWA-5250(a)(2), Corrective Measures, states that if leakage occurs at a bolted connection, the bolting shall be removed, visually examined (VT-3) for corrosion, and evaluated in accordance with IWA-3100.

Licensee's Code Relief Request: The licensee requested relief from the corrective measures required by IWA-5250(a)(2) for leakage at bolted connections that is detected during system pressure tests. Relief is requested for Class 1, raised face flange joints and similar configurations where the bolt is exposed in the area of the leakage path. The components included in this request are (1) the body-to-casing seals for Recirculation Pumps A, B, and C, (2) Recirculation Valve V-37-11, and (3) Main Steam Isolation Valves V-1-7 and V-1-8.

Licensee's Basis for Requesting Relief: The licensee stated:

"Numerous industry studies on the degradation and failure mechanisms in 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. These documents have shown that bolt failures have primarily occurred in pressurized water reactors (PWR), at 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 preload 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.

GPUN 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 examination requirements for post bolted flange leakage is 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 A 193, grade B7 specifications. 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 under nuclear quality control program or have been tested in accordance with the GPUN QA 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 torqued to preload stresses of less than 50% of the yield strength. This has been the standard practice at Oyster Creek, and is closely monitored by the Plant Engineering Department, Mechanical Section.

3. Borated Water: Unlike pressurized water reactors, Oyster Creek does not use borated water in its primary coolant system. The reactor coolant system is pure demineralized water and is frequently monitored for chemical composition and contaminants. No corrosion inducing additives are used or allowed. It is the GPUN position that chemical corrosion is not a 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 connections of oxygen, mitigating the process of both chemical corrosion and stress corrosion cracking.

As the corrosion addressed by the Code is caused by leakage, there is no need to inspect bolts which are not in the leakage stream. This position is reflected in section IWA-5250(a)(2) of the 1990 addenda to ASME XI. A majority of Oyster Creek's bolted joint configurations in the primary coolant system permit easy access for inspection of bolts to VT-1 requirements while in place and under tension. The midsection of the studs and bolts can be closely examined due to existing space between the mating flanges. Any corrosion which could occur in the flange area would be maximized at the bolt-flange interface where the metal-water-air combination exists. This area is visible while the bolt is under tension.

GPUN requests that the specified relief requested from the Code requirements be granted, as the major factors which can result in bolt corrosion are: 1) not applicable to Oyster Creek (such as chemically accelerated corrosion); 2) under control by existing procurement and examination programs (such as the quality assurance requirements for procurement and the existing ISI requirements for examination); and 3) not existent at Oyster Creek as evidenced by extensive operating history, controlled maintenance procedures, and inservice inspection records compiled since 1969."

Licensee's Proposed Examination: The licensee has proposed a VT-1 visual examination on at least one bolt in the area of interest for evidence of corrosion while the bolt is in place and under tension.

Evaluation: The Code requires that if leakage occurs at bolted connections, the bolting shall be removed and visually examined for corrosion. In lieu of this requirement, the licensee has proposed to perform a VT-1 visual examination of at least one bolt in the area of interest with the bolt in place and under tension. This is not an acceptable alternative to the Code requirements. The concept of examining one bolt in the leakage stream has been adopted by the Code in the 1990 Addenda to the 1989 Edition. The 1990 Addenda requires the removal of at least one bolt at leaking bolted connections to ensure that corrosion is not occurring at that connection. Since the licensee's alternative does not include the removal of at least one bolt, and since the impracticality of the Code requirements have not been supported, relief should be denied.

B. Request for Relief No. R12, (Part B), Paragraph IWA-5250(a)(2), Corrective Measures for Control Rod Drive Housing Connections

Code Requirement: Paragraph IWA-5250(a)(2), Corrective Measures, states that if leakage occurs at a bolted connection, the bolting shall be removed, visually examined (VT-3) for corrosion, and evaluated in accordance with IWA-3100.

Licensee's Code Relief Request: The license requested relief from the Corrective Measures required by subarticle IWA-5250(a)(2) for leakage at control rod drive (CRD) housing connections 14-19, 14-47, 22-35, 22-51, 46-39, and 42-43.

Licensee's Basis for Requesting Relief: The licensee stated:

"Numerous industry studies on the degradation and failure mechanisms in 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. These documents have shown that bolt failures have primarily occurred in pressurized water reactors (PWR), at 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 preload 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.

GPUN 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. GPUN has determined that the present scope of ASME XI NDE examination requirements for post bolted flange leakage is 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 A 193, grade B7 specifications. 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 under nuclear quality control program or have been tested in accordance with the GPUN QA 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 torqued to preload stresses of less than 50% of the yield strength. This has been the standard practice at Oyster Creek, and is closely monitored by the Plant Engineering Department, Mechanical Section.

3. Borated Water: Unlike pressurized water reactors, Oyster Creek does not use borated water in its primary coolant system. The reactor coolant system is pure demineralized water and is frequently monitored for chemical composition and contaminants. No corrosion

inducing additives are used or allowed. It is the GPUN position that chemical corrosion is not a 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 connections of oxygen, mitigating the process of both chemical corrosion and stress corrosion cracking.

CRD 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 decreases when the vessel metal temperature reaches normal operating band and the gaskets and o-rings were properly seated by the required scram 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 has revealed no degradation caused by corrosion.

During the exchange of CRDs, the bolts are cleaned and ASME section XI examinations are performed. GPUN 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. Every refueling outage since the plant went on line in 1969, there have been scheduled CRD exchanges. By the end of 1979 a total of 191 drives had been exchanged. During the 10 years ending in 1989, 128 more drives had been exchanged. Over the course of these twenty years of inspections there have been no reports of bolt failure due to corrosion. GPUN shall continue to inspect the bolts during these periods of opportunity and will also employ alternative methods of examination if the need is justified. Although there is a small possibility 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.

GPUN requests that the specified relief requested from the Code requirements be granted, as the major factors which can result in bolt corrosion are: 1) not applicable to Oyster Creek (such as chemically accelerated corrosion); 2) under control by existing procurement and examination programs (such as the quality assurance requirements for procurement and the existing ISI requirements for examination); and 3) not existent at Oyster Creek as evidenced by extensive operating history, controlled maintenance procedures, and inservice inspection records compiled since 1969."

Licensee's Proposed Examination: The licensee has proposed a VT-1 visual examination in accordance with Table IWB-2500-1, Examination Category B-G-2, Item B7.80 with the CRD housing disassembled. Additionally, if determined to be necessary by the VT-1 examination, a PT examination shall be performed on the bolts in accordance with General Electric's Service Information Letter (SIL) No. 483.

Evaluation: IWA-5250(a) requires that if leakage is detected during system pressure tests, the source of the leakage shall be located and evaluated by the owner. For bolted connections, corrective measures require the removal of the bolting for performance of a VT-3 visual examination for corrosion. The licensee has requested relief from removal of the bolts to perform the visual examination of the bolting at leaking CRD housings. The licensee's proposed examination is a VT-1 visual examination in accordance with Table IWB-2500-1, Examination Category B-G-2, Item B7.80.

The licensee has stated that leakage of the CRD housings occurs primarily when the primary system is pressurized prior to heat-up and/or during CRD scram time testing, and that this leakage drastically decreases when the vessel metal reaches normal operating temperature. This change in leakage has been evaluated and found acceptable by the vendor (General Electric). Furthermore, engineering evaluations indicate that three (of eight) uniformly distributed bolts can support the full CRD loading while remaining within the stress limits identified by the ASME Code. The possibility of a failure in five bolts is minimal.

The licensee stated that VT-1 will be performed on bolts from CRD housings that are disassembled. In addition, a liquid penetrant surface examination will be performed in accordance with GE SIL No. 483 if determined necessary by the VT-1 visual examination. This examination will provide an acceptable level of quality and safety in that the sample of bolts examined will be sufficient to identify degradation trends that do occur. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative should be authorized provided that the subject bolts receive a VT-3 (or VT-1) visual examination for corrosion whenever a CRD housing disassembled.

C. Request for Relief No. R12, (Part C), Paragraph IWA-5250(a)(2), Corrective Measures, For Class 1 Bolted Connections

Code Requirement: Paragraph IWA-5250(a)(2), Corrective Measures, states that if leakage occurs at a bolted connection, the bolting shall be removed, visually examined (VT-3) for corrosion, and evaluated in accordance with IWA-3100.

Licensee's Code Relief Request: The licensee requested relief from the Code requirements in IWA-5250(a)(2) for leakage at Class 1 bolted connections with flush face bolting configurations that is detected during system pressure tests.

Licensee's Basis for Requesting Relief: The licensee stated:

"Numerous industry studies on the degradation and failure mechanisms in 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. These documents have shown that bolt failures have primarily occurred in pressurized water reactors (PWR), at 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 preload 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.

GPUN 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. GPUN has determined that the present scope of ASME XI NDE examination requirements for post bolted flange leakage is 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 A 193, grade B7 specifications. 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 under nuclear quality control program or have been tested in accordance with the GPUN QA 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 torqued to preload stresses of less than 50% of the yield strength. This has been the standard practice at Oyster Creek, and is closely monitored by the Plant Engineering Department, Mechanical Section.

3. Borated Water: Unlike pressurized water reactors, Oyster Creek does not use borated water in its primary coolant system. The reactor coolant system is pure demineralized water and is frequently monitored for chemical composition and contaminants. No corrosion inducing additives are used or allowed. It is the GPUN position that chemical corrosion is not a 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 connections of oxygen, mitigating the process of both chemical corrosion and stress corrosion cracking.

Flush face flanges do not permit visible inspection without removing the bolt in question. To remove all of the bolts for the sole purpose of investigating suspect wastage is unnecessary when proper consideration is given to the controls employed at Oyster Creek to minimize the corrosion mechanism. ASME XI, 1990 Addenda, Section IWA-5250(a)(2) allows for the examination of at least one bolt from the area of interest in a leaking bolted joint. This provides an acceptable level of confidence that corrosion has not occurred and still minimizes the amount of radiation received by Maintenance and Quality Control NDE personnel.

GPUN requests that the specified relief requested from the Code requirements be granted, as the major factors which can result in bolt corrosion are: 1) not applicable to Oyster Creek (such as chemically accelerated corrosion); 2) under control by existing procurement and examination programs (such as the quality assurance requirements for procurement and the existing ISI requirements for examination); and 3) not existent at Oyster Creek as evidenced by extensive operating history, controlled maintenance procedures, and inservice inspection records compiled since 1969."

<u>Licensee's Proposed Examination</u>: The licensee has proposed a VT-1 visual examination on at least one bolt removed from leaking bolted connections. The bolt removed for examination shall be the one nearest to the source of the leakage.

<u>Evaluation</u>: The 1986 Edition requires that all bolting be removed and visually examined (VT-3) at leaking bolted connections. The licensee's proposed alternative is to remove one bolt from leaking bolted connections to perform a VT-1 visual examination. The concept of removing one bolt for visual examination has been incorporated into subparagraph IWA-5250(a)(2) of the 1990 Addenda. The 1990 Addenda allows for removal and examination of the bolt closest to the leakage source. When the removed bolt shows evidence of degradation, all remaining bolting in the connection shall be removed, VT-3 examined, and evaluated.

Removing and examining the bolt that is closest to the source of leakage is an acceptable approach because the bolt examined will represent worst case conditions for the bolting set. This method will provide an acceptable level of quality and safety for bolted connections. In addition, the licensee will perform a VT-1 examination in lieu of the required VT-3 examination. Since the VT-1 visual examination is a more stringent examination, this should be considered acceptable. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative examination should be authorized provided that all the associated requirements of subparagraph IWA-5250(a)(2) of the 1990 Addenda are met.

D. Request for Relief No. IWA-5214, Hydrostatic Testing of Repair Welds on the Condensate Transfer System

Code Requirement: Paragraph IWA-5214, Repair and Replacements, states that: (a) a component repair or replacement shall be pressure tested prior to resumption of service if required by IWA-4400 and IWA-4600, and (b) the test pressure and temperature for the hydrostatic test subsequent to the component repair or replacement shall comply with the system test pressure and temperature specified in IWB-5222, IWC-5222, and IWD-5223, as applicable to the system containing the repaired or replaced component.

Licensee's Code Relief Request: The licensee requested one-time relief from the scheduling requirements of IWA-5214(a) for a Class 3 portion of the condensate transfer system.

Licensee's Basis for Requesting Relief: The licensee stated:

"During the first few days of the current refueling outage (14R), the need to perform a Code Class 3 repair to a section of Condensate Transfer piping was identified. The Condensate Transfer system had been scheduled for a brief outage to allow for the correction of a non-related concern. During this outage, a repair weld was completed in accordance with Code requirements. However, the previously scheduled time available to remove the Condensate Transfer system from service did not allow for planning or implementation necessary to perform the system hydrostatic test.

In March 1991 during the previous refueling outage (13R), the Oyster Creek Station experienced a loss of electrical power supply redundancy. The NRC dispatched an Augmented Inspection Team which subsequently issued Inspection Report 50-219/91-80. The cover letter to that report states in part:

> '...concerns were raised regarding your (GPUN's) approach to outage planning and scheduling that did not include an evaluation of plant vulnerability with respect to equipment configurations...'

GPUN addressed this document in a serious and expeditious manner, resulting in defined Risk Management Plant Configurations for 14R. These conditions ensured that multiple sources of cooling, water inventory, reactivity control, and electric power were defined and available at all times during the present outage. To meet this requirement, extensive credit was taken for the Condensate Transfer system during nearly the entire outage.

In the 14R outage schedule, the short system outage on the Condensate Transfer to address a non-related concern was utilized to work the leak repair in parallel with the previously scheduled maintenance. However, due to the late discovery of the newly required maintenance, there was insufficient time to allow for the scheduling of a longer window to allow for the ASME hydrostatic test.

This delayed test was carried as an open item for the remainder of the outage. It was anticipated that emergent work could result in a second Condensate Transfer outage window. However, the end of 14R is presently scheduled within a few days of this letter, GPUN is requesting schedular relief to postpone the system hydrostatic test on the weld repair to the Condensate Transfer system to our next refueling outage, 15R.

The Condensate Transfer system was designed for 200 psig and 100°F. Therefore the system hydrostatic test required by the Code would be performed at 220 psig. As interim examinations, a system inservice leak test was performed at 165 psig, and a VT-2 inspection was completed. No leakage was noted. Additionally, a 100% dye penetrant (PT) test was performed to locate any surface indications. None were found.

Based on the low temperature and pressure of this system, and the Code inspections which have been performed, this repair has been determined to be technically acceptable for operation until a full hydrostatic test can be performed next outage. Additionally, this relief, if granted, will allow time to plan and schedule the requisite system outage for Condensate Transfer during 15R while still ensuring that adequate sources of water are available for the plant in all outage Risk Management configurations."

As the result of a February 11, 1993, conference call, the licensee provided additional information regarding the repair as follows:

1. "The crack which was repaired on the Condensate Transfer system was to a 2 inch piping socket weld on a "T" connection to a six inch header. In addition to the repair to the identified weld, a new support was added in the location of the connection to reduce stresses in the area.

2. By this letter, GPUN requests permission to invoke ASME Code Case N-416 to allow alternative examinations to be performed on a Class 3 component. Specifically, a system inservice leak test was performed and a VT-2 inspection performed. No leaks were noted. Additionally, a dye penetrant inspection was performed to meet the surface examination requirements of the code case. Relief is requested as the portion of the Condensate Transfer system in question is Class 3 piping, and N-416 allows alternative examinations to Class 2 piping."

<u>Licensee's Proposed Examination</u>: The licensee has proposed a VT-2 visual examination during the system inservice leak test at 165 psig. The licensee's original proposal was to perform the Code-required system hydrostatic test of the repaired condensate transfer system during the next outage (15R). In the February 11, 1993 submittal, the licensee requested to invoke Code Case N-416 for the repair, which would allow deferment of the hydrostatic test to the next regularly scheduled system hydrostatic test.

<u>Evaluation</u>: IWA-5214(b) requires a hydrostatic test subsequent to the component repair or replacement and prior to return to service. In lieu of this requirement, the licensee requested to use Code Case N-416 and

defer the hydrostatic test until the next (15R), or a later outage. Code Case N-416, *Alternative Rules for Hydrostatic Testing of Repair and Replacement of Class 2 Piping, Section XI, Division 1*, allows deferment of the Code-required (IWA-5214) hydrostatic test that cannot be isolated by existing valves or that cannot be isolated without securing safety and relief valves, to the next regularly scheduled hydrostatic test with the following provisions:

(a) Prior to or immediately upon return to service, a visual examination (VT-2) for leakage shall be conducted during the system functional test or during a system inservice test in the repaired or replaced portion of the piping system.

(b) The repair or replacement welds shall be examined in accordance with IWA-4000 and IWA-7000 using volumetric examinations methods for full penetration welds or surface examination methods for partial penetration welds.

The licensee has met the above provisions, including the performance of a 100% dye penetrant examination for surface indications. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed examination will provide an acceptable level of quality and safety and should be authorized. Although the licensee has not discussed the impracticality of isolating the subject portions of the condensate transfer system, it was previously determined that operation of this system is necessary for safe operation during refueling outages. Therefore, removal of the condensate transfer system from service to perform the Code-required system hydrostatic test is not warranted and deferral to the end of the interval with the regularly scheduled system hydrostatic test should be considered acceptable.

3.0 CONCLUSION

For Request for Relief R12, Part B and Part C, and Request for Relief IWA-5214, the proposed alternatives will provide an acceptable level of quality and safety, therefore, should be authorized pursuant to 10 CFR 50.55a(a)(3)(i). For Request for Relief R12, Part A, it is recommended that relief be denied.