September 19, 1994

Mr. James Taylor
Executive Director of Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555

PETITION FOR EMERGENCY ENFORCEMENT ACTION UNDER PROVISIONS OF 10 CFR 2.206 WITH REGARD TO OYSTER CREEK NUCLEAR POWER STATION

INTRODUCTION

Nuclear Information and Resource Service and Oyster Creek Nuclear Watch (hereinafter refered to as the petitioners) hereby petition the staff of the Nuclear Regulatory Commission (NRC or staff) to immediately suspend Oyster Creek's license until corrective actions are taken on the part of the licensee and the NRC with regard to the following open issues.

While the petitioners have combined the following issues and requested actions under one petition, it is their contention that either one of the open items is sufficient to warrant the immediate suspension of the license.

BACKGROUND AND REQUESTED ACTIONS

1) Age-Related Deterioration of Safety-Class Reactor Internal Components

NRC has identified that a number of safety-class reactor internal component parts in General Electric Boiling Water Reac ors (BWR) are becoming increasingly vulnerable to age-related deterioration. NRC has issued the report "Boiling Water Reactor Internals Aging Degradation Study" Phase 1 (September, 1993) analyzing the issues and NRC Generic Letter 94-03 (GL 94-03) "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors" (July 25, 1994) which stipulates that all licensees of BWRs "1) inspect the core shrouds no later than the next scheduled refueling outage, and perform an appropriate evaluation and/or repair based on the results of the inspection; and 2) perform a safety analysis supporting continued operation of the facility until inspections are conducted."[1]

NRC has defined "safety-class items"[2] as internal components that are parts of the core support system, used for reactor coolant flow control, and core heat transfer enhancement in

[1] NRC Generic Letter 94-03: "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reacto s." July 25, 1994, U.S. Nuclear Regulatory Commission.

accordance with the rules and regulations of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code.

As a result of inspections for Intergranular Stress Corrosion Cracking (IGSSC) of the core shroud component part, licensees for 12 domestic and overseas BWRs have discovered extensive cracking on welds of the core shroud, including one unit where it was discovered to have a crack in a weld fully circumferential and 1.55 inches deep on a 2 inch wall. The core shroud is a safety-class component part that provides 1) lateral restraint to the reactor core (nuclear fuel, control rods, etc.), 2) support structure for maintaining the proper spacing of the upper ends of the fuel assemblies, and 3) support and guidance for control rod guide tubes. NRC issued GL 94-03 identifying the technical issues and safety significance of IGSCC on the core shroud and states that "NRC has an overall concern with cracking of BWR internals and encourages licensees to work closely with the BWR Owners Group on coordination of inspections, evaluations, and repair options for internals cracking."[3] Additionally, NRC has informed BWR Owners Group of concerns with regard to the lack of analysis of the "synergistic effects" of multiple internal component cracking.[4] These concerns are based in part on the Oak Ridge National Laboratory's "Boiling Water Reactor Internals Aging Degradation Study" published in 1993 which states that:

"In summary, 19 of the 25 selected BWR internals are susceptible to SCC (stress corrosion cracking), including IASCC (irradiation assisted stress corrosion cracking). Specifically, 6 out of the 19 are susceptible to IASCC."[5]

Oyster Creek station is the oldest operating General Electric Mark I BWR with its initial criticality commencing in May, 1969 and the third oldest operating reactor in the United States.

It is the contention of the petitioners that among the BWRs, Oyster Creek has been subjected to the longest period of operational conditions that cause safety-class reactor internal components to embrittle and crack. These conditions include the thermal and mechanical loading associated with start-up, reactor operation, and shutdown, hydrodynamic induced

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2 "Boiling Water Reactor Internals Aging Degradation Study," NUREG/CR-5754, Oak Ridge National Laboratory, September, 1993, p. 3.

3 GL 94-03

4 USNRC and Boiling Water Reactor Owner Group Management Meeting, September 8, 1994, Rockville, MD

5 NUREG/CR-5754, p.26.

vibrations, fatigue, contact with thermally hot and corrosive coolant water, long-term exposure to intense radioactive bombardment, and additionally, conditions promoting cracking arising out of specific methods used for fabricating components (welding and cold-working). Internals made of cast austenitic stainless steel (Type 304) have an extensive history of susceptibility to embrittlement after prolonged operations and are prone to cracking.

The core shroud is but one safety-class component part listed as susceptible to IGSCC and irradiation-assisted stress corrosion cracking (IASCC). The BWR Owners Group has stated that cracking of the core shroud is a warning signal that additional safety-class reactor internals are increasingly more susceptible to the same age-related deterioration.[6] It is the contention of the petitioners that cracking of any single part or multiple components endangers the respective safety-class functions and jeopardizes the safe operation of the nuclear station.

A representative list of reactor internals fabricated of Type 304 stainless steel and their safety-class function includes:

1) core spray sparger injects emergency cooling water into core in the event of a LOCA;

2) core spray internal piping supplies cooling water to the reactor fuel assemblies in the event of a LOCA;

3) core plate perforations provides lateral support and guidance to control rods;

4) top guide provides lateral support and maintains proper spacing of the upper ends of the fuel assemblies;

5) control rod drive housing provides access into the reactor pressure vessel for the control rods and supports the control rods;

6) fuel support assembly provides supporting structure for the reactor core.

7) control rod guide tubes provides lateral guidance to control rods and fuel assemblies.

The Boiling Water Reactor Owners Group has identified additional safety-class vessel internals to include the in-core housing, SRM/IRM dry tube, access hole cover, the LPCI coupling, and vessel ID brackets.

Prior to the current refueling outage (R15), Oyster Creek had not inspected for core shroud cracking.[7] This finding is again confirmed in the licensee's response (August 24, 1994) to

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6 "Core Shroud and Vessel Internals Concerns," Nuclear Regulatory Commission/ Boiling Water Reactor Owners Group (BWROG) Meeting, June 28, 1994, Handout of Viewgraphs.

7 NRC/BWROG Meeting, June 28, 1994, Owner Group Shroud Data Sheet.

Generic Letter 94-03. The petitioners contend that 1) additional safety-class reactor internals have not been adequately inspected to determine if cracking has occurred to those safety-class reactor parts and, 2) a safety analysis has not been performed on the potential synergistic effects of multiple component cracking that supports a continuation of operation without a full inspection and repair of all safety-class reactor internals at Oyster Creek .

A review of NRC documents has demonstrated that a large number safety-class internal components are vulnerable to IGSCC and IASCC as identified in the NRC report "BWR Internals Aging Degradation Study." Only 10 of 36 U.S. BWRs have inspected their core shrouds to date and 7 of those reactors were found to have cracks greater than 30 inches long and 2 reactors with cracks fully circumferential on the shroud. These inspections have been performed on reactors with significantly less exposure time to IGSCC and IASCC than Oyster Creek. Therefore, as it specifically relates to safety-class reactor internals at Oyster Creek, it is the contention of the petitioners that the NRC's role as regulator of the nuclear industry is to pursue a more aggressive action plan than merely "encourage licensees to work closely with BWR Owners Group on coordination of inspections, evaluations, and repair options for internals cracking."[8]

The petitioners acknowledge that the licensee is planning to fulfill the requirement of GL 94-03 with regard to the inspection of the core shroud and that the licensee is planning additional inspections to the vessel internals. However, because of the number safety class components vulnerable to age-related deterioration, the significant percentage and extensive degree of cracking already documented at BWRs with less exposure time than Oyster Creek, and the immediacy of the current outage at Oyster Creek, it is the expressed concern of the petitioners that the licensee's current inspection program include a focus on all safety-class reactor internals.

It is the requested action of the petitioners that;

A) NRC immediately suspend the Oyster Creek operating license until the licensee inspects and repairs or replaces <u>all safety-class reactor internal component parts</u> subject to embrittlement and cracking;

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8 GL 94-03

B) NRC immediately suspend the Oyster Creek operating license until the licensee provides an analysis on the synergistic effects of through wall cracking of multiple safety-class components;

2) Fuel Pool Cooling Design Deficiencies on Single Unit GE BWRs As It Pertains to Oyster Creek

For the past two years NRC has been aware of serious design deficiencies in BWRs regarding "Loss of Spent Fuel Pool Cooling" as the result of an Engineering Discrepancy Report (April, 1992) and the 10 CFR 21 Report of Substantial Safety Hazard (November 27, 1992) filed by Messrs. David Lochbaum and Donald Prevatte on contract for Pennsylvania Power and Light's (PP&L) Susquehanna Steam Electric Station. Messrs. Lochbaum and Prevatte have identified design deficiencies for the handling of loss of fuel pool cooling events associated with Design Basis Loss of Coolant Accidents (DBA LOCAs) at the two PP&L BWR units. Central to the Lochbaum and Prevatte findings "is the inability to remove decay heat from the spent fuel pools for the various design events which mechanistically incapacitate the normal fuel pool cooling system and the resultant effects from loss of normal cooling on the safety-related systems and components in the reactor building."[9] As a result of these defects, there is the potential for a meltdown of irradiated fuel outside primary containment, the failure of all safety-related systems in the reactor building, failure of containment systems, and the catastrophic off-site release of radiation. The engineers additionally identified that other BWRs with the same basic design deficiencies were vulnerable to the fuel pool boiling accident.

Lochbaum and Prevatte have thoroughly documented numerous technical concerns and violations including;

A) Seismically and environmentally unqualified instrumentation on the BWR fuel storage pool;

B) Fuel pool instrumentation is on non-1E power system and not considered safety-related;

C) Emergency procedures in the event of a LOCA do not address the fuel pool cooling issue and will cause fuel pool cooling failure by de-energizing non-1E power systems;

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[9] **10CFR21 Report of Substantial Safety Hazard, November 27, 1992, Docket No. 50-387, David Lochbaum and Donald Prevatte.**

D) Reactor building Heating Ventilation and Air Conditioning (HVAC) has not been analyzed and qualified for the effects of heavy condensation and temperature increases resulting from a fuel pool boil event:

E) Standby Gas Treatment System (SGTS) is environmentally unqualified for boiling fuel pool conditions;

F) The currently calculated radiation levels resulting from a DBA at the fuel pool Emergency Service Water valves, which must be manipulated by hand to make up feedwater to a boiling fuel pool, would be in the range of thousands of Rems per hour.

These deficiencies represent violations of NRC regulatory requirements including 10CFR50.49, 10CFR50 App. A GDC 63, 10CFR50 Appendix B, Criterion III, Design Control and Reg. Guides 1.13, 1.89, and 1.97.

As a result of two years of meticulous documentation and persistent interaction with NRC and the licensee on the part of Messrs. Lochbaum and Prevatte, the NRC now acknowledges that design defects in the BWR fuel pool cooling system do in fact pose a significant increase in risk to the public safety at the Susquehanna units and additional reactors. Similarly, PP&L announced in writing to the NRC on June 1, 1994 that it was "committing to actions which will prevent spent fuel pool boiling."[10] Essentially, PP&L has decided to cross-tie Unit 1 and Unit 2 irradiated fuel storage pools by removing the fuel pool to cask storage pit gates during normal operations. As well, PP&L committed to make additional modifications to the plant including irradiated fuel pool instrumentation and numerous procedural changes. These modifications have brought the Susquehanna units closer to being in compliance with the regulatory requirements.

However, the resolution as put forward by PP&L is not applicable to any of the single unit BWRs. In fact, Susquehanna, a much newer power station built to more stringent regulations, was unable to show how a single unit alone could cool its irradiated fuel pool following a seismic event which disabled the fuel pool cooling system.

Oyster Creek is a single unit facility with no adjacent units to rely on. To the best of the petitioners knowledge, Oyster Creek has not docketed any material with regard to the BWR design deficiencies as identified by Messrs. Lochbaum and Prevatte. Consequently, it is the

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[10] **"Susquehanna Steam Electric Station Response to Request for Additional Information Concerning Loss of Spent Fuel Cooling Initiated by the Design Basis Seismic Event," PP&L, June 1, 1994.**

contention of the petitioners that Oyster Creek nuclear station, as well as the licensees of other single unit BWRs with non-safety related fuel pool cooling systems, currently may be in violation of NRC regulatory requirements. It is the contention of the petitioners that it is impossible to know if the plant is in regulatory compliance until Oyster Creek has docketed an analysis regarding their vulnerability to a DBA LOCA resulting in a boiling fuel pool as documented by the Lochbaum and Prevatte submittals.

It is the petitioners' requested action that;

A) the NRC immediately suspend the license of Oyster Creek nuclear power station until the licensee has analyzed and mitigated any areas of noncompliance with regard to the irradiated fuel pool cooling issue as a single unit BWR;

B) the NRC issue a Generic Letter requiring other licensees of single unit BWRs to provide information on the fuel pool boiling issue as it pertains to their specific units to 1) verify compliance with the regulatory requirements and, 2) take prompt and appropriate action to mitigate the issue if the units are found to be out of compliance.

DESCRIPTION OF THE PETITIONERS

Nuclear Information and Resource Service (NIRS) is a nonprofit organization whose work is related to nuclear power, radioactive waste, and renewable energy. Members include New Jersey residents whose health and safety are put at direct risk by the unsafe operation of the Oyster Creek nuclear power station. With an office in Washington, DC, NIRS has been a participant in nuclear regulatory affairs, including rulemakings, enforcement actions, and adjudications involving individual nuclear power stations since 1978.

Oyster Creek Nuclear Watch (OCNW) is a grassroots organization formed in June of 1994 by citizens concerned about the safety of the Oyster Creek Nuclear Generating Station. OCNW is currently in the process of incorporating as a New Jersey not-for-profit. Most of OCNW supporters are residents of Ocean County, New Jersey. Their health and safety therefore depends upon the safe operation of the Oyster Creek Nuclear Generating Station. The supporters of OCNW are disturbed by newspaper accounts and indications by experts that the Oyster Creek Nuclear Generating Station may have reactor internal component cracking and /or flaws in the irradiated fuel storage system. As a local citizens group lacking expertise in the technical issues

related to nuclear power, OCNW relies upon the technical knowledge of its co-petitioner, the Nuclear Information and Resource Service, in the technical aspects of the document.

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