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components; (e) the time-to-boil calculation is dictated by the amount of decay heat generated and the volume of water in the fuel pool rather than the number of reactors at a site that store irradiated fuel in a separate pool; (f) NRC documents state that the time-toboil calculation for FitzPatrick following a loss-of-coolant accident is 8 hours, and NYPA documents state that the time-to-boil calculations in two cases are 11.86 and 5.36 hours. Finally, nothing indicates that the time-to-boil calculation at OCNGS is longer than the time-to-boil calculation at the Susquehanna facility; and (g) the NRC and the licensee have failed to establish whether redundant components and power supplies to the OCNGS fuel pool cooling system have been qualified as Class 1E systems.

The Petitioners' requests that the Commission immediately suspend the OCNGS operating license were denied in my letter of October 27, 1994, to the Petitioners, because (1) OCNGS was in a refueling outage, had inspected core shroud welds, and was making structural modifications before restart of the unit to address some weld cracks found during the inspection, and (2) inspections and corrective actions recommended by General Electric Company and the American Society of Mechanical Engineers Boiler and Pressure Vessel Code for various reactor internals had been and continued to be performed by the Licensee.

The Petitioners' request for treatment of their letter of December 13, 1994, as a formal appeal of the NRC staff's denial of their request of September 19, 1994, for immediate suspension of the OCNGS operating license, was denied in my letter of April 10, 1995, to the Petitioners. The Petitioners provided no basis for revisiting the denial of their request of September 19, 1994, for immediate suspension of the license. As discussed below, the Licensee completed all ASME Code Section XI reactor vessel internal inspections and BWROG recommended inspections and took appropriate remedial action before re-start of OCNGS in December 1994. The NRC staff was also aware of the potential problem for United States BWRs raised by cracking in top guide and core plates of foreign BWRs before the restart of OCNGS. The NRC staff determined, as explained below, that cracks in these components would not adversely affect safety of the plant because of differences in the OCNGS design as compared to the affected foreign reactors.

Regarding the OCNGS spent fuel pool cooling system capability, the staff determined that the time to the onset of

spent fuel pool boiling following a loss of spent fuel pool cooling during periods where the reactor vessel contains irradiated fuel at single unit BWR sites, such as OCNGS, is long enough to allow compensatory measures. The probability of a sustained loss of spent fuel pool cooling creating adverse environmental conditions that may cause failure of essential equipment is extremely low. Therefore, the staff has concluded that immediate action to address the concerns the Petitioners have identified at OCNGS is not justified. As stated in my letter of October 27, 1994, spent fuel pool safety is being reviewed generically by the staff and this review has not yet been completed.

The Petitioners' request for a public meeting was denied in my letter of April 10, 1995.¹ The issue of internals cracking has been discussed at several public meetings, including a public meeting on November 4, 1994, that a representative of NIRS attended regarding the OCNGS core shroud. With respect to spent fuel pool cooling, the staff has held several public meetings and public briefings with the Advisory Committee on Reactor Safeguards. Summaries of these public meetings are available in the NRC Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC, and at the local public document rooms for the affected BWR plants. Transcripts of ACRS meetings are also available.

The NRC staff's review of the issues related to cracking of reactor internal components, raised by Requests (1) and (2) of the September 19, 1994, Petition, and Request (1) of the December 13, 1994, supplemental Petition, is now complete. For the reasons set forth below, the Petition is denied with respect to these requests. A Director's Decision concerning the issues related to irradiated fuel pool cooling and fuel pool boiling, raised by Requests (3) and (4) of the September 19, 1994, Petition and Requests (2), (3), and (4) of the December 13, 1994, supplemental Petition will be issued upon completion of the NRC staff's review regarding those matters.

II. Background

Intergranular stress corrosion cracking (IGSCC) of BWR internal components has been identified as a technical issue of concern by both the NRC staff and the

nuclear industry. The core shroud is among the internal reactor components susceptible to IGSCC. Identification of cracking at the circumferential beltline region welds in several plants during 1993 led to the publication of NRC Information Notice (IN) 93-79, "Core Shroud Cracking at Beltline Region Welds in Boiling-Water Reactors,⁵ issued on September 30, 1993. Several licensees inspected their core shrouds during planned outages in the spring of 1994 and found cracking at the circumferential welds. The NRC has closely monitored these inspection activities. Additionally, licensees have inspected other BWR reactor vessel internal components as discussed below. NRC issued IN 94-42, "Cracking in the Lower Region of the Core Shroud in Boiling-Water Reactors," on June 7, 1994, and Supplement 1 to IN 94-42, on July 19, 1994, concerning cracking in the core shroud found at Dresden Unit 3 and Quad Cities Unit 1. IN 95-17, "Reactor Vessel Top Guide and Core Plate Cracking," issued on March 10, 1995, concerned reactor vessel top guide and core plate cracking. The NRC has monitored Licensee inspection activities of these components at the OCNGS as discussed below.

III. Discussion

A. Petitioners request that the NRC suspend the OCNGS license until the Licensee inspects and repairs or replaces all safety-class reactor internal component parts subject to embrittlement and cracking. Nuclear power reactor licensees, including GPUN, are required by 10 C.F.R. § 50.55a to implement inservice inspection programs in accordance with the guidelines of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The scope of the inservice inspection programs for reactor pressure vessels and their internal components is prescribed by ASME Code, Section XI, Division 1, Subsections IWA and IWB. The Licensee is also required by ASME Code, Section XI, Article IWA-6000, to submit the results of these inspections to the NRC within 90 days of completion. The NRC staff performs periodic audits of licensee-implemented inservice inspection programs to determine compliance with applicable codes and regulations. These audits are documented in NRC inspection reports, which are publicly available at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the OCNGS located at the Ocean County Library, Reference

¹ In addition, the NRC staff determined, in accordance with the guidance in NRC Management Directive 8.11, "Review Process for 10 CFR 2.206 Petitions," that an informal public hearing was not warranted because the Petition did not present new information or a new approach for evaluating the concerns Petitioners raised.