UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION
William T. Russell, Director

In the Matter of	)	Docket No. 50-293
	)	
BOSTON EDISON COMPANY	)	License No. DRP-35
	)	
(Pilgrim Nuclear Power Station)	)	(10 CFR 2.206)

DIRECTOR'S DECISION UNDER 10 CFR 2.206

I. INTRODUCTION

Ms. Mary Elizabeth Lampert and 62 other individuals (Petitioners) submitted a Petition dated March 10, 1995, pursuant to 10 CFR 2.206 requesting action with regard to the Pilgrim Nuclear Power Station (Pilgrim), operated by the Boston Edison Company (licensee).

The Petition requested that: (1) during the refueling outage and In-Vessel Visual Inspection scheduled for March 25, 1995, by the licensee, certain technical concerns be addressed, and that before Pilgrim goes back on-line, appropriate repairs be made or corrective action be taken; (2) the U.S. Nuclear Regulatory Commission (NRC or Commission) discuss the status of such repairs or corrective actions with the public in Plymouth, Massachusetts; and (3) the NRC terminate its policy of issuing Notices of Enforcement Discretion (NOEDs) and begin enforcing the regulations again.

As the bases for these requests, the Petitioners identified three groups of technical concerns: (1) age-related deterioration of 25 safety related reactor internals; (2) parts and components "known to be a problem at Pilgrim," including the core shroud, water level indicators, quality assurance for fuel pool cooling system during loss-of-coolant accident/loss of offsite power, motor-operated valves, containment integrity, drywell liner corrosion vulnerability, station blackout vulnerability, and Rosemount transmitters; and (3) parts and components "potentially a problem at Pilgrim," including potential fuel rod corrosion and substandard and/or counterfeit parts. The Petitioners contend that allowing the reactor to operate under a NOED cannot pose less risk to the public health and safety than keeping the reactor shut down until NRC regulations are met.

II. BACKGROUND

By letter dated April 19, 1995, the NRC acknowledged receipt of the Petition and offered a public meeting, which was held in Plymouth, Massachusetts on May 11, 1995. At that meeting, the results of the licensee's inspections conducted during the outage were discussed.

I have completed my evaluation of the Petition. As explained below, Petitioners have failed to raise any safety concern which would warrant delaying restart of the Pilgrim Nuclear Power Station (which occurred on June 2, 1995), and the Petitioners' request that the NRC terminate the use of NOEDs is denied.

III. DISCUSSION

A. Age-Related Deterioration of Reactor Internals

Many components inside boiling-water reactor (BWR) vessels (i.e., internals) are made of materials such as stainless steel and various alloys that are susceptible to corrosion and cracking. As materials age, they degrade. This degradation can be accelerated by stresses from temperature and pressure changes, irradiation effects on material properties, chemical interactions, and other corrosive environments. As BWRs age, the amount of cracking is expected to increase. Several cases of internals cracking and degradation have been reported to the NRC over the years. In a number of cases, the NRC has concluded that full power operation of the reactor with time-dependent degradation, related to the operating environment, of reactor vessel internals is acceptable as long as the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) safety margins are satisfied and maintained. In the remaining cases, replacement or repairs were performed on the degraded components or internals. The NRC has met with industry every year since 1988 to review the generic safety implications of reactor internals potentially susceptible to age-related cracking. Additionally, a special industry review group, the Boiling Water Reactor Vessels and Internals Project (BWRVIP), was formed to focus on resolution of reactor vessel and internals degradation.

Several industry standards and regulatory requirements and guidelines are in place to address inservice inspections (ISIs) of reactor components. Moreover, the NRC and industry have responded as new issues emerge. For example, issued Generic Letter (GL) 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds (IGSCC) in Boiling Water Reactors," "in July 1994 requesting Licensees to inspect their shrouds and provide an analysis justifying continued operation until inspections could be completed. General Electric issued Services Information Letter (SIL) No. 588, "Top Guide and Core Plate Cracking," in February 1995 providing specific recommendations for inspections of BWR top guides and core plates. In addition to addressing emerging the BWRVIP is working on a comprehensive plan that will provide detailed guidance on managing cracking in all BWR internals. The plan will address cracking susceptibility, safety consequences, inspection scope and methodology, flaw evaluation, repair strategies, and mitigation of degradation. Several top level executives and technical staff of the Licensee are on the various BWRVIP committees that are developing generic standards for ISI and repairs.

Petitioners request that 25 components be inspected during the 1995 refueling outage (RFO No. 10), and that they be free of any signs of IGSCC or other kind of fatigue. During RFO No. 10, the licensee indicated completion of the ISI examinations for the third period of the second Pilgrim 10-year inspection interval in accordance with Section XI of the ASME Code, 1980 Edition with Winter 1980 Addenda. This included all 25 components requested by the Petitioners, except the steam separator, neutron source holder and surveillance sample holders which are not safety-related components. The in-core neutron flux monitor components, in-housings, guide tubes, dry tubes, the vessel head cooling spray nozzle, and the fuel supports are not required by NRC regulations to be inspected. The NRC inspected Pilgrim's ISI program and related activities during the 1994 RFO No. 9 and concluded that the second interval program plan was sufficiently comprehensive to ensure safety and met the requirements of the ASME Code, and thus 10 CFR 50.55a(a)(2). The ISI examinations conducted in RFO No. 10 included the core support structure, control rod drive housing, core spray internal piping and spargers, and feedwater spargers.

Augmented examinations were also conducted in which various internals were examined, including the shroud support and access hole covers, jet pump riser braces, shroud head bolts, jet pump sensing lines, steam dryer support, steam dryer baffle plate, top guide, core plate, and control rod stub tubes.

Control blades (control rods for BWRs) are replaced at specified intervals. The licensee also implemented a preemptive repair of its core shroud due to the high susceptibility to IGSCC. See Section III.B.(1), below. As discussed during the May 11, 1995, meeting between the NRC and the public, the inspection results from RFO No. 10 did not reveal any indications of significant time-dependent deterioration of the reactor internals.

The NRC staff concludes that the inspections, examinations, and repairs performed by the licensee during RFO No. 10 and previous outages are sufficient to provide reasonable assurance that no age-related failure of components or internals would occur during the next operating cycle, which is scheduled to end March 21, 1997. Design features, plant procedures, and operator training are developed to ensure safety in the unlikely event that a failure were to occur. The NRC will continue to take regulatory action on a plant-specific or generic basis, as may be appropriate, when time-dependent degradation issues are identified. During the next refueling outage, the licensee will again conduct an in-vessel inspection of safety-related interval components.

Accordingly, Petitioners have not raised a safety concern regarding age-related degradation of reactor internals at Pilgrim which would have warranted prohibiting restart after RFO No. 10.

B. Parts and Components Known to Be a Problem at Pilgrim

(1) Core Shroud

Petitioners express concern about the type of repairs that would be done to the core shroud during RFO No. 10, based on "the different approach taken in Germany at the Wuergassen NPS and at the Oyster Creek NPS in NJ." Petitioners state that German nuclear regulators required replacement of shrouds with cracking, rather than repair of the shroud. Petitioners state that at Oyster Creek, ten tie rods are attached to holes in Type 304 stainless steel, which is subject to IGSCC and is welded to the bottom of the core shroud assembly. Petitioners are concerned that if the same approach were used at Pilgrim, there would be problems with the structural integrity of the materials the tie rods are welded to and with "loose parts."

Officials of PreussenElektra AG, the owner of Wuergassen, initially intended to replace the core shroud at Wuergassen, as reported in *Nucleonics Week* on November 24, 1994. Differences in the design of Wuergassen and NRC-licensed BWRs exist which would make replacement of the core shroud at Wuergassen less complicated than at NRC-licensed plants. For example, the shroud at Wuergassen is bolted on to the shroud support, whereas shrouds of NRC licensees are welded. However, in a press release issued June 1, 1995, PreussenElektra AG decided to decommission the Wuergassen NPS based on economic considerations. As a result, replacement of a BWR core shroud, foreign or domestic, has yet to be undertaken.

By letter dated November 25, 1994, the NRC staff issued the "Safety Evaluation Regarding the Oyster Creek Core Shroud Repair," which approved the scheduled repair as an acceptable alternative to the standards of the ASME Boiler and Pressure Vessel Code. *See* 10 C.F.R. § 50.55a(a)(2) and 50.55a(a)(3)(i). Oyster Creek and Pilgrim are utilizing similar tie-rod assemblies to structurally replace the core shroud during normal and accident conditions. The difference in the number of tie-rod assemblies used, i.e., ten tie-rod assemblies at Oyster Creek and four tie-rod assemblies at Pilgrim, is related to the contracted vendor's loading distribution design and the associated hardware on the tie-rod assembly. The NRC staff has thoroughly reviewed the Pilgrim repair design and conducted inspections during the core shroud repair process. The staff issued the "Safety Evaluation Regarding Pilgrim Nuclear Power Station Core Shroud Repair," dated May 12, 1995. A synopsis of our review follows.

The design of the Pilgrim shroud repair consists of four (4) stabilizer assemblies, which are installed 90° apart in the shroud/reactor vessel annulus, between attachment points at the top of the shroud and the gusset assemblies on the lower shroud support plate. Each stabilizer assembly consists of a tie rod, and upper spring, a lower spring, an upper bracket and other smaller parts. The tie rod provides the vertical load transfer from the upper bracket to the reactor pressure vessel (RPV) gusset attachment and supports the springs. The upper spring provides radial load transfer at the top guide elevation from the shroud to the RPV. The lower spring provides radial load transfer from the shroud at the core plate elevation to the RPV. The upper bracket provides an attachment to the top of the shroud and restrains the upper shroud weld. Upper-mid and lower-mid supports along the tie rod length provide radial load transfer for the mid sections of the shroud and increase the natural frequency of the tie rods to reduce flow-induced vibration. Two wedges between the core support plate and the shroud are also installed at each stabilizer location to prevent relative motion of the core plate to the shroud. Each cylindrical section of the shroud between welds H1 through H9 is prevented from unacceptable lateral motion by the stabilizers. The section between H9 and H10 is prevented from unacceptable motion by the existing gussets. The lower end of the stabilizers are attached to pins which are placed in holes cut into gusset plates at the bottom. The gusset assemblies and their welds are Inconel and are not considered subject to cracking by industry and the NRC staff. Inconel is a nickel based alloy which is less likely to corrode and degrade than stainless steel, which is an iron based alloy. However, these welds, including those attaching the gussets to the vessel and to the lower shroud support plate (which must resist the vertical stabilizer loads) have been inspected for cracks during this outage, and no crack indications were found. Together, the tie rods and lateral restraints resist both vertical and lateral loads resulting from normal operation and design accident loads, including seismic loads and postulated pipe ruptures.

The NRC staff found that the proposed repair does not affect the ability of operators to insert control rods, the performance of the ECCS, particularly the core spray system, or the ability to reflood and cool the core. The staff concluded that the proposed repair does not pose adverse consequences to plant safety; therefore, plant operation is acceptable with the proposed core shroud repair installed.

In compliance with 10 CFR 50.55a(a)(3)(i), the core shroud repair has been designed as an alternative to the requirements of the ASME Code. Based on a review of the shroud modification hardware from structural, systems, materials, and fabrication considerations, the NRC staff concludes that the proposed modifications of the Pilgrim core shroud would provide an acceptable level of quality and safety. The staff has determined that the licensee's repair of the core shroud will not result in any increased risk to the public health and safety and is, therefore, acceptable.

(2) Water Level Indicators

Petitioners assert that because of a pipe design deficiency, water level indicators at Pilgrim are not fully operable due to high-pressured gas in the water, and that operator training is not the appropriate solution.

Level anomalies were observed in reactor vessel water level indication at several BWRs during controlled depressurization, while commencing plant outages or following reactor trips. These anomalies consisted of "spiking" or "notching" of level indication, and in one instance, a sustained error in level indication. The root cause of these level indication anomalies is the effect of non-condensible gas dissolved in the reference leg of "cold reference leg" type water level instruments. Under rapid depressurization conditions, non-condensible gases can cause significant errors in the level indication.

Cold reference leg water level instruments measure reactor vessel water level by measuring the differential pressure of two columns of water, i.e. the variable leg and the constant height reference leg. The reference leg is maintained filled to a constant height of water by the condensate chamber. Steam is condensed in the condensate chamber and keeps the reference leg full. Excess condensate is returned to the vessel through the steam supply line. Non-condensible gases, such as hydrogen and oxygen, formed by radiolysis in the reactor vessel, are present in the steam supplied to the condensate chamber. The gases can collect in the condensate chamber and can accumulate to high partial pressures. The gases then become dissolved in the water at the top of the reference leg, and the dissolved gases can be transported down the reference leg by small leaks in valves and fittings at the bottom of the reference leg, diffusion, and/or thermal convection.

Dissolved gases in the reference leg do not present a problem unless the instrument is depressurized. When depressurized, the gases come out of solution and form bubbles that travel up the reference leg. During slow depressurization, level indication has been seen to temporarily "spike" or "notch" while a bubble moves through the vertical sections of the piping. Significant spiking may automatically actuate such systems as the primary containment isolation system (PCIS). This occurred at the Pilgrim plant. After spiking, which is of short duration, the indicated water level returns to actual level. Level spiking is of little significance. Bubbling of the gases may eject a significant amount of water from the reference leg. Loss of reference leg inventory will cause an erroneously high level indication. This occurred during a normal plant cooldown on January 21, 1993, at Washington Nuclear Power Unit 2 (WNP-2), resulting in a 32-inch error in level indication that gradually recovered over a period of 2 hours. If the reactor is rapidly depressurized, as would occur during a design basis loss-of-coolant accident (LOCA) or opening of the automatic depressurization system (ADS) valves, even larger errors in the level indication could result. However, analyses presented by the industry indicated that significant errors would not be expected until the reactor is depressurized below approximately 450 psi.

The NRC staff has taken several actions to address this problem. The BWR Owners Group (BWROG) Regulatory Response Group (RRG) was activated during July 1992. The staff also issued Information Notice 92-54 in July 1992, GL 92-04 in August 1992, and Information Notice 93-27 in March 1993 to alert licensees to the potential problem and to request information concerning actions taken or planned by licensees in response to potential errors in level indication. The BWROG conducted a test program to support their efforts to resolve this issue. The results of the BWROG reference leg de-gas test program [illegible] that no significant errors in level indication will occur until the reactor is depressurized below 450 psig, and that large errors in level indication are possible once the reactor is depressurized to lower pressures.

The NRC staff received additional information from the BWROG pertaining to reactor vessel water level instrumentation inaccuracies during normal depressurization due to the effects of non-condensible gas. At the staff's request, the BWROG submitted a report on May 20, 1993, discussing the impact of level errors on automatic safety system response and operator actions during transients and accidents initiated from reduced pressure conditions during plant cooldown (shutdown mode). Based on this information, in addition to the January 21, 1993, WNP-2 event, and data from the reference leg de-gas testing that was conducted by the BWROG, the staff concluded that additional short-term actions needed to be taken for protection against potential events occurring during normal cooldown. On May 28, 1993, NRC Bulletin (NRCB) 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation," was issued, in which the staff requested each BWR licensee to implement additional short-term compensatory actions, and to implement a hardware modification to resolve this issue at the next cold shutdown after July 30, 1993.

The staff has received responses to NRC Bulletin 93-03 from all licensees. All licensees completed short-term compensatory actions and committed to install hardware modifications. Licensees for all affected plants have either completed installation of hardware modifications or are currently shutdown and will install the hardware modifications prior to restart.

To solve the problem identified in NRC Bulletin 93-03, Pilgrim installed a backfill modification to all safety-related water level instrumentation in July 1993. Non-safety-related control instrumentation was not modified by Pilgrim, because such instrumentation was not covered by the actions requested in NRC Bulletin 93-03.

As Petitioners note, an event occurred at Pilgrim on November 8, 1993, involving the non-safety-related water level instrumentation. This event was caused by failure of the licensee to back flush the feedwater control instrumentation reference legs prior to restart due to procedural inadequacy and failure to cross-check multiple indications of reactor vessel water level during startup due to operator error. This event is not safety significant for the following reasons:

(a) event initiation was the result of two independent errors which are not expected to have a high frequency of recurrence;
(b) safety systems and non-safety systems are separated by design; thus, the availability and capability of the safety systems should not be impacted by errors in the non-safety instrumentation and the ability of safety systems to protect the plant should not be compromised; and
(c) the safety systems responded to the event as expected.

This issue is closed because the licensee took adequate corrective actions in response to the November 8, 1993, event. *See* NRC Inspection Report 50-293/93-20, dated January 11, 1994.

Based on the above, Petitioners have not raised a substantial safety concern regarding safety-related water level instrumentation at Pilgrim.

(3) Quality Assurance for Fuel Pool Cooling System During LOCA/LOOP

The Petitioners asserted that workers would be exposed to fatal levels of radiation while manually activating the backup cooling system during a LOCA.

In November 1992 two engineers working under contract at Susquehanna Steam Electric Station filed a 10 CFR 21.21 report. The report detailed design concerns at Susquehanna that could lead to the sustained loss of forced cooling for the stored spent fuel under certain accident or abnormal conditions. The engineers postulated that the environmental conditions developed following a loss of forced cooling would adversely affect equipment necessary for safe-shutdown and accident mitigation. The engineers concluded that these issues had generic implications.

Between November 1992 and October 1994, the NRC staff performed an extensive evaluation of the Susquehanna spent fuel pool cooling design concerns. The staff concluded that these concerns were of low safety significance in the "Final Safety Evaluation By the Office of Nuclear Reactor Regulation Regarding Loss of Spent Fuel Pool Cooling Events," dated June 19, 1995. This conclusion was based on the fact that the probability of recovering forced cooling of the stored spent fuel with access to the necessary equipment was high, and the probability of experiencing a severe core damage accident, which may prevent access to systems need to cool the spent fuel pool, was low.

The staff issued Information Notice 93-83, "Potential Loss Of Spent Fuel Pool Cooling Following A Loss Of Coolant Accident," (October 7, 1993), describing the Section 21.21 report related to Susquehanna. The information notice did not require specific action by licensees. Recognizing the plant-specific design features and operational controls of most spent fuel pool cooling system designs, the staff concluded that further evaluation of spent fuel pool storage safety issues at other plants was warranted to determine the need for further generic action.[1]

The staff has developed and begun implementing a generic action plan to evaluate generic issues. On-site safety assessments of spent fuel storage at selected reactor facilities have been completed. Monticello Nuclear Power Plant is similar to Pilgrim and was one of the nuclear facilities assessed during the week of March 27, 1995. The assessment team concluded that the potential for a sustained loss of spent fuel pool cooling or a significant loss of spent fuel pool coolant inventory at the site visited was remote based on observed design features and operational controls. Based on the above, the

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[1] In the near future, the staff will issue an additional information notice describing the results of its detailed evaluation of the Susquehanna facility. This information notice will be an interim communication and will not represent the end of the staff's generic review.

NRC staff has concluded that the Petitioners have not identified any safety concerns at Pilgrim regarding spent fuel pool cooling during a LOCA/LOOP.

(4) Motor-Operated Valves

Petitioners request information on the status of the motor-operated valve (MOV) program at Pilgrim, and inquire why Pilgrim has not been required to fix all MOVs during the March 1995 outage.

The NRC issued GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance" (June 28, 1989) to request that licensees verify the capability of all safety-related MOVs to perform their design basis functions. GL 89-10 requested that licensees complete differential pressure and flow testing for the verification of MOV design basis capability within 5 years after the issuance of GL 89-10 or three refueling outages after December 1989, whichever was later.

Pilgrim is scheduled to complete its MOV Design Basis Capability Verification by April 1997. Although this is somewhat later than some other plants, the licensee is being given the same number of outages (three outages with 24 month cycles) as other licensees to complete the verification, and the program commenced somewhat later at Pilgrim due to the 1990 restart from an extended outage.

During the implementation of GL 89-10, licensees have discovered more MOV concerns and experienced greater difficulty in conducting MOV tests at full design basis differential pressure and flow than envisioned when the GL 89-10 schedule was established. Where significant MOV problems are identified, the NRC ensures that licensees resolve these problems promptly. Further, when the evaluation of NRC-sponsored MOV test results indicated potential problems with specific MOVs in high pressure systems at boiling-water reactor (BWR) nuclear power plants, the NRC issued Supplement 3 to GL 89-10 in October 1990. Supplement 3 requested that BWR licensees promptly evaluate the capability of MOVs used for containment isolation in the steam lines of the high-pressure coolant injection and reactor core isolation cooling systems and in the supply line to the reactor water cleanup system. Further, the staff issued Supplement 5 to GL 89-10 in June 1993, requesting that licensees ensure that new information on the increased inaccuracy of MOV diagnostic equipment be addressed. These two actions were satisfactorily completed by Pilgrim.

The NRC staff has been monitoring the progress of the GL 89-10 program at Pilgrim closely. From December 13 to 17, 1993, and March 22 to 25, 1994, the NRC staff conducted an inspection of the GL 89-10 program at Pilgrim. As stated in NRC Inspection Report 50-293/92-80, the NRC staff had the following findings as a result of the March 1992 inspection:

(a) The method used to set the MOV torque switches using diagnostic testing equipment was inadequate;
(b) the torque switch settings on several safety-related MOVs were not set in accordance with the plant design documents;

(c) corrective actions taken in response to an internal audit of the GL 89-10 Program regarding the torque switch settings of safety-related valves were inadequate;
(d) the GL Supplement 3 response for the reactor water cleanup system isolation valve 1202-5 was inadequate;
(e) plans for conducting design-basis differential pressure testing have not been clearly established;
(f) the current work instructions for performing design basis reviews and switch setting calculations lack adequate detail; and
(g) a considerable effort remains to implement the GL 89-10 program in a timely manner.

The NRC staff found considerable progress in the licensee's MOV program since the initial NRC team inspection in March 1992. Particularly, the staff concluded that the findings from the March 1992 inspection had been satisfactorily addressed. *See* Inspection Report No. 50-293/93-22 (April 14, 1994). In addition, the testing of differential pressure and/or static pressure of all of the Priority 1 (highest risk) MOVs that can be tested was completed by the end of RFO No. 10. Additionally, the licensee has evaluated all of the GL 89-10 MOVs for susceptibility to pressure locking and thermal binding and, by the end of RFO No. 10, completed modifications on the few valves that were considered susceptible. The staff concludes that the licensee is on schedule to meet its April 1997 completion date.

Based on the progress made to date by the licensee in implementing its GL 89-10 program at Pilgrim, the NRC staff did not consider it necessary that the licensee complete its GL 89-10 program during RFO No. 10. In addition to review of the licensee's submittals in response to GL 89-10 and its supplements, the NRC staff is conducting an extensive inspection program to evaluate the MOV program implemented in response to GL 89-10 at Pilgrim, as well as at other nuclear power plants. The NRC staff concludes that the licensee has substantially reduced the concerns with MOV operation under design basis conditions and is progressing significantly toward completing the GL 89-10 program. Nevertheless, if significant MOV problems are identified at Pilgrim, the licensee will be responsible for addressing those problems in accordance with their safety significance, irrespective of the GL 89-10 completion schedule. Further, the NRC will continue to take regulatory action on a plant-specific or generic basis, as appropriate, when MOV problems are identified.

Based upon the actions taken to date by the licensee to address safety-related MOV issues and the NRC's inspections regarding the licensee's actions on the GL 89-10 program, the NRC staff concludes that no corrective actions are required.

(5) Containment Integrity

Petitioners ask whether the hardened wetwell vent system (HWWVS), referred to as the "Torus Vent", which "allows venting of radioactive effluents directly into our atmosphere," will be corrected in RFO No. 10.

The licensee installed the HWWVS modification during the 1986-1988 outage, thus providing the capability to establish alternate containment decay heat removal if RHR torus cooling capability is lost. The direct torus venting minimizes the potential for core damage and containment failure. The HWWVS has the capability of mitigating a wide range of events including many that are beyond the Design Basis Accidents for the facility. Its installation, along with the procedures for its use, will reduce the likelihood of a core melt from accident sequences involving the loss of long-term decay heat removal. This accomplished by preventing any further damage to safety equipment in the reactor building by ensuring that the piping from the containment to the venting stack will not fail. Further, as a mitigation measure, the vent pathway is located in the wetwell air space. This location ensures that the vented non-condensible gases will pass through the suppression pool thereby significantly scrubbing the fission products. The HWWVS is an improvement that the NRC staff recommended in its Mark I Containment Performance Improvement Program, which identified plant modifications that could enhance the capability to both prevent and mitigate the consequences of severe accidents.

The HWWVS has valves that are kept closed during plant operation, assuring containment integrity. Additionally, the HWWVS design incorporates a device called a rupture disc, which provides an additional leak-tight barrier to further prevent the transport of the containment atmosphere in the wetwell to the atmosphere. The HWWVS is not in use during normal plant operation, nor is it expected to be used during anticipated transient conditions. Petitioners have not demonstrated any basis why this system should be "corrected."

(6) Drywell Liner Corrosion

Petitioners request information on the status of drywell liner corrosion vulnerability and asks whether it would be corrected during RFO No. 10.

The NRC issued GL 87-05, "Request For Additional Information-Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells," as a result of the November 1986 discovery of corrosion of the Oyster Creek steel drywell in the area of the sand cushion. GL 87-05 did not establish any regulatory requirements other than for Mark I licensees to provide the staff with information as to what actions, if any, were being taken as a result of the Oyster Creek finding. The licensee responded to GL 87-05 by letter dated May 11 1987. The licensee implemented a surveillance program to detect whether a corrosive environment exists on the external surface of the drywell. This is done by checking the drywell liner air gap drain lines for the presence of water during every refueling outage.

In January 1987, prior to issuance of GL 87-05, the licensee conducted ultrasonic inspections of the interior of the drywell liner in the area of the sand drains, which confirmed liner integrity. In January 1988, the drain lines were verified not to be blocked by using a boroscope. As of the last surveillance, conducted on March 31, 1995, no water leakage had been detected. Petitioners have not demonstrated any basis for correcting this system.

(7) Station Blackout

Petitioners request information on station blackout vulnerability and ask whether it would be corrected during RFO No. 10.

On December 23, 1993, the NRC issued "NRC Pilot Station Blackout Team Inspection," a report concerning the Pilgrim plant, Inspection Report 50-293/93-80. The purpose of that inspection was to review Pilgrim's programs, procedures, training, equipment and systems, and supporting documentation for implementing the Station Blackout (SBO) Rule, 10 CFR 50.63. The actions taken to implement the station blackout rule are important because many of the systems required for decay heat removal and containment cooling are dependent on the availability of alternating current (ac) power. In the event of a station blackout, relatively few systems that do not require ac power are depended upon to remove decay heat, until ac power is restored.

The staff concluded in Inspection Report 50-293/93-80 that:

(a) Pilgrim had sufficient condensate inventory to cope with an 8-hour SBO duration;
(b) all areas which contained equipment needed for SBO coping had proper cooling;
(c) there was sufficient evidence that the torus temperature and the reactor vessel conditions would be maintained according to the plant TSs;
(d) the overall communications capability available during an SBO were adequate;
(e) adequate emergency lighting was available to support plant personnel operations during a station blackout; and
(f) plant modifications were properly installed, and post-modification and pre-operational tests were conducted in accordance with proper test procedures. Quality assurance and maintenance practices, operator training, and staffing levels were appropriate to cope with an SBO.

Accordingly, the Pilgrim plant is in compliance with Section 50.63 and the plant does not have a SBO vulnerability requiring "correction" during RFO No. 10.

(8) Rosemount Transmitters

Petitioners request information on the status of Rosemount transmitters at Pilgrim, and ask whether all would be inspected and corrected during RFO No. 10.

On December 22, 1992, the NRC staff issued Bulletin 90-01, Supplement 1, "Loss of Fill-Oil in Transmitters Manufactured by Rosemount," which requested that licensees take appropriate corrective actions for Model 1153, Series B and D, and Model 1154 Rosemount transmitters manufactured before July 11, 1989, and used in safety-related applications or Anticipated Transient Without Scram

(ATWS) systems. The performance of a transmitter that is leaking fill-oil gradually deteriorates and may eventually lead to failure. Although some failed transmitters have shown symptoms of loss of fill-oil prior to failure, it has been reported that in some cases the failure of a transmitter that is leaking fill-oil may be difficult to detect during operation. Transmitter failures that are not readily detectable increase the potential for common mode failure and may result in the affected safety system not performing its intended safety function. Supplement 1 identified specific actions for replacement or enhanced surveillance monitoring of the these transmitters, used in high pressure (greater than 1500 psi), medium pressure (greater than 500 psi and less than 1500 psi), and low pressure (less than 500 psi) applications.

The licensee responded to the requested actions of Bulletin 90-01, Supplement 1, on March 5, 1993 and August 30, 1993. There are a total of 40 Model 1153B transmitters currently in service, 14 medium pressure transmitters and 26 low pressure transmitters. The licensee committed to include each of these transmitters in its enhanced surveillance monitoring program. The licensee stated that there were no Model 1153D or 1154 transmitters currently in service.

The licensee also stated that there were 33 Model 1153B transmitters, manufactured after July 1989, in service. Such transmitters are not subject to the Bulletin 90-01, Supplement 1, requested actions because Rosemount corrected the oil leakage problem by an improved manufacturing and quality assurance process. Although Supplement 1 does not require these transmitters to be included in an enhanced surveillance monitoring program, the licensee has chosen to include them in its program. The licensee's enhanced surveillance program is based on both the trending of operating drift data and calibration drift data, and is in accordance with Rosemount Technical Bulletin No. 4.

The NRC, with assistance from its contractor, reviewed the licensee's response to Supplement 1, and in a letter dated November 29, 1994, concluded that the licensee satisfied the reporting requirements and conformed to the requested actions of Bulletin 90-01, Supplement 1. Accordingly, no further actions by the licensee were required with respect to this Rosemount Issue during RFO No. 10.

C. Parts and Components Potentially a Problem at Pilgrim

(1) Fuel Rod Corrosion

Petitioners request information regarding the status of zirconium alloy tubes installed at Pilgrim, and asks if their susceptibility to nodular corrosion would be corrected during RFO No. 10.

Nodular corrosion is a phenomena seen in plants that have copper in the reactor water at a concentration in the 20-30 parts per billion (ppb) range. Pilgrim systems design limits copper levels to less than 1 ppb in the reactor water. Additionally, all fuel rod cladding in use at Pilgrim has been subject to the GE Nuclear Energy in-process heat treatment (IPHT) process[2], which is a heat treatment process that evenly distributes the composition of the alloy thus lowering the susceptibility to nodular corrosion. Pilgrim has not experienced nodular corrosion, and failure of fuel rods is not expected from this phenomenon.

The NRC staff conducted two inspections of Teledyne Wah Chang Albany (TWCA), the manufacturer of zirconium alloy tubes. In April 1990, an employee of Teledyne Wah Chang Albany (TWCA) raised two concerns regarding the efficacy of TWCA's "beta quench" process, a step in the manufacture of zircaloy tube shells which improves the corrosion resistance of that product: (1) the accuracy of temperature indicating devices as a predictor of the temperature of the bulk profile of the zircaloy billet the beta quench process was measuring, and (2) even if the profiles of the induction furnaces are accurate, the induction furnaces cannot reproduce the profile conditions for each production zircaloy billet as the heating in the furnace is very sensitive to the position of the billet in the furnace.

Neither of the two NRC inspections substantiated the employee's concerns. See Inspection Reports 99901229/91-01 (November 27, 1991) and 99901229/94-01 (January 31, 1995). These inspection reports are available in the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. TWCA also investigated these concerns. In a letter to the NRC dated January 10, 1991, TWCA forwarded the results of its investigation, concluding that these concerns were unfounded, although the employee continued to have concerns.

Based on the above, Petitioners have not demonstrated any basis for fuel rod corrosion corrective actions.

(2) Substandard and/or Counterfeit Parts

Petitioners state that Pilgrim was one of several plants identified in a 1990 study by the United States Government Accounting Office as using parts which did not meet government standards, but that the NRC has not asked plants such as Pilgrim to replace those parts. Petitioners request information on the status of substandard or counterfeit parts at Pilgrim, such as nuts, bolts, pipe fittings, circuit breakers and fuses, and whether corrective action would be required during RFO No. 10.

The NRC has been pursuing the issue of counterfeit and substandard parts as a two prong process for a number of years. The first process is reactive, directly addressing the possibility that substandard or counterfeit parts may have been supplied to nuclear power plants, assessing the safety significance

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[2] TWCA does not produce fuel clad tubing, but supplies an intermediate product form to customers that do, including GE Nuclear Energy, who performs the IPHT on the forms.

and, if needed, replacing the parts. The second process is a proactive approach of improving the assurance that parts are of a high quality before they are put into use.

Since 1988, the NRC has performed over 200 inspections of vendors. During these inspections, the staff occasionally identified suspect practices and referred those cases to the Office of Investigations to determine if wrongdoing had been committed. The NRC also quickly published and disseminated the information to the entire nuclear industry. Over the past several years, the NRC has issued numerous Bulletins and Information Notices having to do with potential counterfeit and/or substandard parts and material. However, the staff has not yet identified an issue that, from a safety standpoint, resulted in any plant shutdowns. Nonetheless, the NRC determined that several issues could potentially reduce the margin of safety in some plants and requested some actions by licensees, usually through a Bulletin.

If the NRC obtains information that some licensees are identified as potential customers of a vendor suspected of supplying counterfeit or substandard parts, an Information Notice is issued. The issuance of an Information Notice does not mean that the identified licensee(s) did, in fact, receive the questionable parts, but rather that they were potential customers. The licensees are responsible for reviewing their own procurement records to identify if they received the suspect parts. Their actions are subject to NRC review and inspection.

The 1990 GAO report, "Nuclear Safety and Health: Counterfeit and Substandard Products Are a Governmentwide Concern," lists a wide range of products as having been received or suspected of having been received by nuclear plants. The information provided by the GAO report regarding products used in nuclear operations was obtained from the NRC and all of the information was made public through various NRC Information Notices and Bulletins. The Pilgrim station was listed in the GAO report as having received counterfeit or substandard fasteners and circuit breakers. Pilgrim was also listed as being suspected of receiving counterfeit or substandard pipe fittings/flanges and fuses.

On November 6, 1987, the NRC issued Bulletin 87-02, "Fastener Testing to Determine Compliance With Applicable Material Specifications." The Bulletin requested all licensees to review their receipt inspection requirements and internal controls for fasteners and to determine, through testing, whether fasteners in stores at their facilities met required mechanical and chemical material specification requirements. Licensee responses were summarized in NUREG-1349, "Compilation of Fastener Testing Data Received in Response to NRC Compliance Bulletin 87-02." NUREG-1349 identified that, of over 3500 fasteners tested, 8 percent of safety-related and 12 percent of nonsafety-related fasteners were found to be nonconforming. However, only 2 percent of the safety-related fasteners were found to be sufficiently out of specification to cause a concern regarding their ability to perform their intended safety function. As a result of the licensees' responses to Bulletin 87-02, the NRC issued a temporary inspection instruction to ensure that licensees verified that fasteners used in nuclear plants met the requisite specifications and that operability of safety-related components was not affected.

In response to Bulletin 87-02, Pilgrim tested 35 safety-related and 29 non-safety-related fasteners. Three safety-related and 6 non-safety-related fasteners were identified as having hardness values slightly out of specification. These slight deviations were not considered safety significant since the hardness deviations consisted of only 1 to 2 Rockwell points which is very close to the test accuracy of ± 1.0 Rockwell point. Furthermore, it is commonly recognized in the industry that this property is most easily influenced by variations in chemistry, heat treatment, and surface treatments.

On May 6, 1988, the NRC issued Bulletin 88-05, "Nonconforming Materials Supplied by Piping Supplies, Inc. at Folsom, New Jersey and West Jersey Manufacturing Company at Williamstown, New Jersey." That Bulletin required NRC licensees to submit information regarding materials supplied by the named companies and requested the licensees to assure that the materials complied with ASME Code Section III, Subarticle NCA-3800 and design specifications requirements, or were suitable for their intended use, or to replace the materials. Following the issuance of that Bulletin and actions taken by licensees, the NRC met with representatives of the Nuclear Management and Resources Council (NUMARC) to discuss the status of licensee actions. NUMARC presented information on licensee and NUMARC/Electric Power Research Institute (EPRI) testing and evaluation methodology of numerous flanges. The information presented at that meeting showed that the material in question had acceptable strength and that continued use of the fittings and flanges did not present a safety problem. Therefore, the NRC issued Supplement 2 to Bulletin 88-05 on August 3, 1988, announcing that it was appropriate to suspend the actions requested by the Bulletin. NUMARC follow-up reports were analyzed by the staff and judged acceptable. Therefore, no further actions were required.

In response to Bulletin 88-05, Pilgrim identified and tested a number of suspect flanges. All were found to be satisfactory, with the exception of one which tested low in hardness. An engineering evaluation performed by Pilgrim determined the flange was acceptable and did not need to be replaced.

On July 8, 1988, the NRC issued Information Notice 88-46, "Licensee Report of Defective Refurbished Circuit Breakers," which alerted licensees to the possibility of defective circuit breakers being supplied to the nuclear industry. Following the issuance of the notice, the NRC issued Bulletin 88-10, "Nonconforming Molded-Case Circuit Breakers," which requested licensees to take action to provide reasonable assurance that those molded-case circuit breakers that did not have verifiable traceability to the circuit breaker manufacturer were able to perform their safety function. In response to the Bulletin, Pilgrim identified only one of 978 circuit breakers in its warehouse as not being traceable to the original equipment manufacturer. That breaker was the only one purchased on its purchase order and was subsequently discarded.

On April 26, 1988, the NRC issued Information Notice 88-19, "Questionable Certification of Class 1E Components," to alert licensees to a possible problem with the certification of Class 1E components by Planned Maintenance Systems (PMS) of Mt. Vernon, Illinois. Information provided to the NRC by a licensee raised questions regarding the validity of certifications issued by PMS for Class 1E fuses PMS supplied. In response to Information Notice 88-19, the licensee reviewed its procurement/QAD documents. There was no indication that the licensee had procured any material from PMS directly or through Bechtel or General Electric. Furthermore, the NRC review of PMS records indicated that PMS did not supply material or services through intermediate suppliers to the Pilgrim station.

In addition to the Information Notices and Bulletins which identified specifics about potential counterfeit or substandard materials, the NRC staff has issued two generic letters providing information to the industry regarding procurement program improvements to help prevent the acceptance and use of counterfeit and/or substandard material. The industry, through the efforts of the Nuclear Energy Institute (NEI, successor to NUMARC), has also taken a strong approach to improve procurement programs by means of a Comprehensive Procurement Initiative, which addressed five areas which included general procurement, vendor audits, tests and/or inspections, obsolescent, and information exchanges. The Comprehensive Procurement Initiative has greatly reduced the incidence of substandard and/or counterfeit parts in the industry.

In view of the above, no action regarding substandard or counterfeit parts needed to be taken by the licensee before start-up of the Pilgrim plant following RFO No. 10.

D. NRC Oversight and Enforcement Discretion

Petitioners state that since September 1989, the NRC has either waived or chosen not to enforce regulations at nuclear reactors more than 340 times, and that of the last 100 industry requests for enforcement discretion, the Commission has granted every one. Petitioners also state that the NRC has granted at least seven NOEDs to Pilgrim since 1989. Petitioners assert that permitting a reactor to operate cannot pose less risk to public health and safety than keeping the reactor shut down until it meets regulations.

The NRC Enforcement Policy, Section VII.C., permits the staff to exercise discretion not to enforce applicable TSs or license conditions by issuance of a NOED. Such enforcement discretion may be exercised only if the NRC staff is clearly satisfied that the action is consistent with protecting the public health and safety, in cases when a licensee's compliance with a TS Limiting Condition for Operation or other license condition would involve:

(a) an unnecessary plant transient; or
(b) performance of testing, inspection or system realignment that is inappropriate with the specific plant conditions; or
(c) unnecessary delays in plant startup without a corresponding health and safety benefit.

For an operating plant, the NOED is intended to (1) avoid undesirable transients as a result of forcing compliance with the license condition and, thus, minimize potential safety consequences and operational risks or (2) eliminate testing, inspection, or system realignment that is inappropriate for the particular plant conditions. For plants in a shutdown condition, the NOED is intended to reduce shutdown risk by avoiding testing, inspection, or system realignment that is inappropriate for the particular plant conditions, in that it does not provide an overall safety benefit, or may, in fact, be detrimental to safety in the particular plant condition.

For plants attempting to start up, the need for exercising enforcement discretion is expected to occur less often than for operating plants, because delaying startup does not ususally leave a plant in a condition in which it could experience undesirable transients. Thus, the issuance of NOEDs for plants attempting to start up must meet a higher threshold.

The use of enforcement discretion does not change the fact that a violation of a license requirement will occur, nor does it imply that enforcement discretion is being exercised for any violation that may have led to the violation for which the licensee requests issuance of a NOED. Where the NRC staff has chosen to issue a NOED, enforcement action is normally considered for the root causes, to the extent violations led to the noncompliance for which enforcement discretion was used.

Petitioners have provided no basis warranting a change in the Commission's policy regarding the exercise of enforcement discretion pursuant to Section VII.C. of the Enforcement Policy.

IV. CONCLUSION

The institution of proceedings in accordance with Section 2.206, as requested by the Petitioner, is appropriate only where substantial safety issues have been raised. See *Consolidated Edison Co. of New York* (Indian Point Units 1, 2, and 3), CLI-75-8, NRC 173, 175 (1975), and *Washington Public Power Supply System* (WPPSS Nuclear Project No. 2), DD-84-7, 19 NRC 899, 923 (1984). This is the standard I have applied to the Petition. Petitioners have not raised any substantial safety concerns regarding age-related deterioration of reactor internals, or with other parts and components at Pilgrim. To the contrary, all potential problems identified by Petitioners regarding reactor internals and components have been satisfactorily addressed by the licensee at Pilgrim. Therefore, Petitioner's request to delay startup of the Pilgrim plant is denied. Additionally, for the reasons discussed above, Petitioners request to terminate the NRC policy of issuing notices of enforcement discretion to reactor licensees is denied. Petitioner's request for a public meeting was granted.

A copy of the Director's Decision will be filed with the Office of the Secretary for the Commission to review in accordance with 10 CFR 2.206(c). As provided by Section 2.206(c), this Decision will constitute the final action of the Commission 25 days after issuance, unless the Commission, on its own motion, institutes a review of the decision within that time.

FOR THE NUCLEAR REGULATORY COMMISSION

W T Russell

William T. Russell, Director
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland,
this 31st day of August 1995