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NUCLEAR REGULATORY COMMISSION ISSUANCES

August 1995

U.S. NUCLEAR REGULATORY COMMISSION

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NUCLEAR REGULATORY COMMISSION ISSUANCES

August 1995

This report includes the issuances received during the specified period from the Commission (CLI), the Atomic Safety and Licensing Boards (LBP), the Administrative Law Judges (ALJ), the Directors' Decisions (DD), and the Decisions on Petitions for Rulemaking (DPRM).

The summaries and headnotes preceding the opinions reported herein are not to be deemed a part of those opinions or have any independent legal significance.

U.S. NUCLEAR REGULATORY COMMISSION

Prepared by the
Division of Freedom of Information and Publications Services
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
(301/415-6844)

COMMISSIONERS

Shirley A. Jackson, Chairman
Kenneth C. Rogers

B. Paul Cotter, Jr., Chief Administrative Judge, Atomic Safety and Licensing Board Panel

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Commission Issuances

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONER:

Shirley A. Jackson, Chairman[1]

In the Matter of

Docket Nos. 70-00270
30-02278-MLA
(Byproduct License
No. 24-00513-32; Special
Nuclear Materials License
No. SNM-247)
(TRUMP-S Project)

CURATORS OF THE
UNIVERSITY OF MISSOURI

August 22, 1995

The Commission denies the University of Missouri's petition for reconsideration seeking a clarification that the "Site Area Emergency" classification for its MURR facility comes into play only when a fire or accident involving nuclear materials could lead to radiation exposures possibly approaching 1-rem whole-body dose *at the site boundary*. The Commission rules that a reference to the site boundary is already implicit in the existing "Site Area Emergency" condition. In addition, the Commission *sua sponte* requires the University either (i) to require evacuation of all persons (except emergency personnel) to a point at least 150 meters from the Alpha Lab whenever an Alert is declared as a result of a fire involving TRUMP-S materials or (ii) to provide the NRC Staff sufficient information to determine that the existing Emergency Plan and procedures (or any proposed modifications of the Plan and procedures) adequately protect the public within the site boundary in the case of a fire involving TRUMP-S materials.

[1] This decision was made by Chairman Jackson under delegated authority, as authorized by NRC Reorganization Plan No. 1 of 1980, after consultation with Commissioner Rogers. Commissioner Rogers has stated his agreement with this decision.

MEMORANDUM AND ORDER
(Petition for Partial Reconsideration)

For a second time, the University of Missouri has asked the Commission to reconsider and clarify its decision to require the University's emergency plan to include a "Site Area Emergency" classification for certain accidents involving nuclear materials. *See* CLI-95-1, 41 NRC 71, 154-56 (1995). The University is concerned that the Commission's decision on the initial reconsideration petition does not specify that a Site Area Emergency comes into play only when a fire or accident involving nuclear materials could lead to radiation "exposures possibly approaching the EPA PAG lower level (1 rem whole body dose)" *at the site boundary*. *See* CLI-95-8, 41 NRC 386, 390-92 (1995).

The Commission considers the reference to the site boundary already implicit in the "Site Area Emergency" condition, and therefore denies the University's petition for reconsideration. The potential for significant exposures at the *site* boundary is what triggers a *Site* Area Emergency. This point was reinforced in CLI-95-8 where the Commission described as "well taken" the University's argument, *inter alia*, that significant releases possibly approaching EPA PAG levels *at the site boundary* should be classified as a Site Area Emergency. CLI-95-8, 41 NRC at 390.

Moreover, definitions in current NRC rules (10 C.F.R. §§40.4, 70.4) and record evidence in this case, including the NRC's Response Technical Manual, ANSI standards, and the University's own Emergency Plan, confirm this understanding of the Site Area Emergency classification — which is designed to designate accidents with potential significant radiation consequences *off site*. *See* CLI-95-1, 41 NRC at 154-56. Conversely, these same materials make clear that an "Alert" (the emergency level immediately below a Site Area Emergency) is the appropriate classification for events *not* likely to spawn radiation consequences *outside the site boundary*. The Commission does not understand its prior decisions in this proceeding to suggest otherwise.

Although we are denying the University's reconsideration request as unnecessary, our further examination of the classification of emergencies arising out of the TRUMP-S Project has brought to light an additional concern: whether the University's "Action" responses are adequate to protect those members of the public *within* the site boundary from radioactive exposure exceeding 1 rem due to a fire involving TRUMP-S nuclear materials. Although the Commission has found such exposures highly unlikely beyond a radius of 150 meters, even in a worst-case fire scenario (*see* CLI-95-1, 41 NRC at 151-52 & nn. 125-126), exposure of 1 rem or more within that radius is a risk that bears further examination.

The University has indicated that its "Alert" classification permits a case-by-case determination "whether any on-site personnel should be evacuated" (by contrast, a Site Area Emergency classification requires "automatic evacuation"). *See* Licensee's Petition for Partial Reconsideration at 6 (filed Mar. 31, 1995). But because a fire involving the TRUMP-S materials may lead to relatively quick radiation releases and because the Commission wishes to minimize any exposure to members of the general public during such releases,[2] the Commission directs the University either (i) to require evacuation of all persons (except emergency personnel) to a point at least 150 meters from the Alpha Lab whenever an Alert is declared as a result of a fire involving TRUMP-S materials or (ii) to provide the NRC Staff sufficient information to determine that the existing Emergency Plan and procedures (or any proposed modification of the Plan and procedures) adequately protect the public within the site boundary in the case of a fire involving TRUMP-S materials. To the extent Staff concludes that further protective measures are necessary, it is instructed to require the University to take such measures.

It is so ORDERED.

For the Commission

JOHN C. HOYLE
Secretary of the Commission

Dated at Rockville, Maryland,
this 22d day of August 1995.

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[2] *See* CLI-95-1, 41 NRC at 155 ("the amount of time available to mitigate the effects of a materials fire would presumably be shorter than the time available to mitigate the effects of an equally serious fire affecting the reactor"). *See also* CLI-95-8, 41 NRC at 391 ("Actual radiation measurements normally come after-the-fact. Site area emergencies are declared on the basis of predictive judgments based on site conditions.").

Atomic Safety and Licensing Boards Issuances

ATOMIC SAFETY AND LICENSING BOARD PANEL

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James P. Gleason,* *Deputy Chief Administrative Judge (Executive)*
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LICENSING BOARDS

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Peter B. Bloch, Chairman
Dr. James H. Carpenter
Thomas D. Murphy

In the Matter of

Docket Nos. 50-424-OLA-3
50-425-OLA-3
(ASLBP No. 93-671-01-OLA-3)
(Re: License Amendment; Transfer to Southern Nuclear)

GEORGIA POWER COMPANY,
et al.
(Vogtle Electric Generating Plant, Units 1 and 2)

August 3, 1995

The Atomic Safety and Licensing Board held that a secretary's communications, recorded in a note by her employer's attorney, are unlikely to be discoverable because they are privileged communications of a client to an attorney. However, the Board ordered the *in camera* inspection of the notes before reaching a final determination concerning the specific factual circumstances present in this case and the applicability of the purposes of the attorney-client privilege.

RULES OF PRACTICE: ATTORNEY-CLIENT PRIVILEGE; FACTS TOLD TO ATTORNEY BY SECRETARY

When the client is a corporation, the attorney-client privilege applies to communications by any corporate employee regardless of position when the communications concern matters within the scope of the employee's corporate duties and the employee is aware that the information is being furnished to enable the attorney to provide legal advice to the corporation. *Upjohn Co. v. United States*, 449 U.S. 383, 396-97, 101 S. Ct. 677, 685-86 (1981).

RULES OF PRACTICE: ATTORNEY-CLIENT PRIVILEGE; SIMPLE FACTS

When a claim of attorney-client privilege is made for a document containing a simple report of facts, the Atomic Safety and Licensing Board may examine the document further in order to ascertain whether granting privilege to the document is consistent with the purposes of the attorney-client privilege.

MEMORANDUM AND ORDER
(Request for Discovery Concerning Ester Dixon)

Allen Mosbaugh (Intervenor) has requested discovery of an attorney's notes of an interview of Ester Dixon conducted in 1992.[1] Georgia Power has persuaded us that Ms. Dixon's communications with its attorney are unlikely to be discoverable because they are privileged communications of a client to an attorney. However, we shall order the *in camera* inspection of those documents before reaching a final determination concerning the specific factual circumstances present in this case and the applicability of the purposes of the attorney-client privilege.

The prevailing standard in this case is found in 10 C.F.R. §§ 2.740(b)(1) and 2.740(b)(2). Section 2.740(b)(1) authorizes discovery of "any matter, not privileged." Section 2.740(b)(2) expands the scope of discovery for trial preparation materials but only if they are "otherwise discoverable under paragraph (b)(1) of this section." However, privileged material is not "otherwise discoverable."

A similar issue already was decided by us and is the law of this case. In *Georgia Power Co.* (Vogtle Electric Generating Plant, Units 1 and 2), LBP-93-18, 38 NRC 121, 124, 125 (1993), we said:

> We accept the following statement of GPC as accurately setting forth the law concerning the attorney-client privilege:[2]

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[1] Intervenor filed its "Motion to Compel Production of Licensee's Notes of Interview of Ester Dixon" (Motion) on June 30, 1995, and Georgia Power Company, *et al.* (Georgia Power) filed its "Response to Intervenor's Motion to Compel Production of Licensee's Notes of Interview of Ester Dixon" (Response) on July 17, 1995. On July 24, 1995, we received by facsimile transmission all but the first page of "Intervenor's Motion to Compel Production of Licensee's Notes of Interview of Ester Dixon." To the extent that the motion deals with the pending motion about Ester Dixon, it is a nonauthorized response and has been disregarded.

[2] GPC Response at 17.

The United States Supreme Court has held that, when the client is a corporation, the attorney-client privilege applies to communications by any corporate employee regardless of position when the communications concern matters within the scope of the employee's corporate duties and the employee is aware that the information is being furnished to enable the attorney to provide legal advice to the corporation. *Upjohn Co. v. United States*, 449 U.S. 383, 396-97, 101 S. Ct. 677, 685-86 (1981); *see also Admiral Ins. Co. v. United States Dist. Court*, 881 F.2d 1486, 1492 (9th Cir. 1989). The Court in *Upjohn* declined to establish an all-encompassing test for application of the attorney-client privilege to corporations. Instead, it held that each case must be evaluated to determine whether application of the privilege would further its underlying purposes of encouraging candid communications between client and counsel and providing effective representation of counsel. *Upjohn, supra*, 449 U.S. at 389, 390-91, 396-97, 101 S. Ct. at 682-86.[3]

* * *

. . . Management may decide it wants to investigate a problem and ascertain the truth. It may need to ask very probing questions. To encourage this kind of appropriate management action, in a complex regulatory setting in which an enforcement action was reasonably foreseeable, GPC used its lawyers. It is appropriate that these professionals should be given as much information as possible without having to risk public disclosure of their work. The attorney-client privilege protects this activity, and the company need not later reveal the affidavits it compiled.

The materials now being sought are a 1992 interview of the person who typed some key documents in this case. This interview, conducted by Georgia Power's attorney, was the first time Ms. Dixon was asked to recall relevant details. Subsequently, Ms. Dixon was deposed in this case in 1994 and gave live testimony in 1995. Ms. Dixon's testimony helps to establish the order in which key documents were typed. It has a bearing on Intervenor's allegation that the "Cash List" was prepared after the "Successful Starts Slide," even though the Cash List is alleged by Georgia Power to have been prepared in order to document the number of starts cited in the Successful Starts Slide.

Careful consideration of the specific facts of this case indicates that Georgia Power's lawyer's interaction with Ms. Dixon was about a simple and straightforward factual matter: when certain documents were typed. Compared to other matters that might be involved in an attorney-client interaction, this matter is relatively straightforward and calls for little attorney sophistication and relatively little trust from the employee. Nor does the employee appear to have a clear personal stake in how the issues concerning company documents may be resolved. If, pursuant to the *Upjohn* test, there is any specific situation in which the attorney-client privilege does not apply, we are close to that case here. As we already said, *Upjohn* requires that:

[3] *See also Duke Power Co.* (Catawba Nuclear Station, Units 1 and 2), CLI-83-31, 18 NRC 1303, 1305 (1983).

each case must be evaluated to determine whether application of the privilege would further its underly ing purposes of encouraging candid communications between client and counsel and providing effective representation of counsel.

We note that there is one other factor operating here. The interview in this case was conducted 2 years after the events. It does not involve fresh recollections, even though it contains the earliest notes of recollections. There have been two subsequent efforts by Intervenor to test those recollections, both in depositions and at the hearing. Since even the earliest recollections were not fresh, there is less reason to consider releasing attorney-client material contained in these notes; and we are aware that *any* release of such material, however justified, would have some dampening effect on subsequent attorney-client communications, particularly within the same company.

A problem we face in judging whether or not to apply the attorney-client privilege, in this borderline situation, is our lack of complete knowledge about the notes of the interview. We find that we cannot properly make an informed judgment in this case without an *in camera* examination of the allegedly privileged notes. Only by examining the document can we be satisfied that the purposes of the privilege are well served by applying it in this instance.[4]

Order

For all the foregoing reasons and upon consideration of the entire record in this matter, it is, this 3d day of August 1995, ORDERED that:

Georgia Power Company, *et al.*, shall promptly present for *in camera* inspection by this Board its notes of the interview of Ester Dixon conducted by its attorney in 1992. Unless the Atomic Safety and Licensing Board shall publish a subsequent opinion on this subject, Intervenor's "Motion to

[4] Somewhat relevant to our determination is *Southern Railway Co. v. Lanham*, 403 F.2d 119 (5th Cir. 1968). However, the opinion did not involve the attorney-client privilege. It stated, at 134, that: "If privileged communications between appellant and its counsel were encompassed by the [trial] court's order to produce, they must be deleted."

Compel Production of Licensee's Notes of Interview of Ester Dixon," filed on June 30, 1995, is *denied.*

FOR THE ATOMIC SAFETY
AND LICENSING BOARD

Peter B. Bloch, Chairman
ADMINISTRATIVE JUDGE

Rockville, Maryland

Directors' Decisions Under 10 CFR 2.206

DIRECTORS' DECISIONS

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF ENFORCEMENT

James Lieberman, Director

In the Matter of **Docket No. 50-245**

NORTHEAST NUCLEAR ENERGY COMPANY
(Millstone Nuclear Power Station, Unit 1) **August 2, 1995**

The Director of the Office of Enforcement has denied a petition filed by Clarence O. Reynolds requesting that the NRC take immediate escalated enforcement action with regard to Millstone Nuclear Power Station Unit 1 on the basis of alleged discriminatory actions taken against him. Specifically, Mr. Reynolds requested that multiple Severity Level II and III violations be issued against the Millstone Unit 1 Maintenance Department, that suspension of Maintenance Department management be instituted pending a complete investigation, and that he be immediately reinstated as maintenance mechanic pending completion of the investigation. The reasons for the denial are fully set forth in the Decision.

DIRECTOR'S DECISION UNDER 10 C.F.R. § 2.206

I. INTRODUCTION

On August 22, 1993, Clarence O. Reynolds (Petitioner) filed a request for enforcement action pursuant to 10 C.F.R. § 2.206 (petition). The petition requested that the Executive Director for Operations take immediate escalated enforcement action with regard to Northeast Nuclear Energy Company's (Licensee's) Millstone Nuclear Power Station Unit 1. Specifically, Mr. Reynolds requested that multiple Severity Level II and III violations be issued against the Millstone

Unit 1 Maintenance Department, that suspension of Maintenance Department management be instituted pending a complete investigation, and that the Executive Director for Operations' (EDO's) office insist that he be immediately reinstated as maintenance mechanic pending this investigation.

On September 21, 1993, the NRC acknowledged receipt of the request and denied the portion of the request that asked that the EDO's office insist on immediate reinstatement of Mr. Reynolds (following his suspension) to his position as a maintenance mechanic pending an investigation and requested additional information to provide the basis to act on the Petitioner's other requests. On October 19, 1993, the Petitioner responded with this additional information.

On June 29, 1994, Petitioner supplemented his original petition. In his supplement, Petitioner informed the NRC that his employment with Northeast Nuclear Energy Company had been terminated on June 27, 1994, and he alleged that the Licensee terminated him due to his raising of safety concerns. Petitioner requested that NRC take action to reinstate him to his employment and provide for back pay for lost wages.

On August 17, 1994, Petitioner again supplemented his petition, requesting that: (1) three Severity Level I violations be issued against Northeast Utilities Chief Executive Officer, Vice President of Nuclear Operations, and the Vice President at Millstone; (2) two Severity Level II violations be issued against the Unit One Director and the Maintenance Manager of Millstone; (3) a Severity Level III violation be issued against the first-line supervisor; (4) the NRC remove all managers mentioned above; and (5) the NRC require reinstatement of Petitioner until the matter is resolved.

Letters dated October 25, 1993, August 16, 1994, and January 27 and March 16, 1995, were received from Northeast Utilities, responding to Petitioner's assertions. Petitioner responded to some of the issues raised by Northeast Nuclear Energy Company in letters dated November 8, 1993, and September 12, 1994.

II. DISCUSSION

As a basis for his August 22, 1993 request, Petitioner asserted that he was suspended from his position at Millstone following his filing of nuclear concerns with Millstone management and the NRC, that there have been other complaints of retaliation that have occurred recently in his department, and that a recent NRC Inspector General's report indicated that there have been a significant number of complaints by employees being discriminated against at Millstone after bringing forth nuclear concerns. As a basis for his June 29, 1994 supplement, Petitioner states that he was terminated by Northeast Nuclear Energy Company due to his raising safety concerns.

Petitioner bases his requests for sanctions on his assertion that he was a victim of discrimination. Therefore, the decision as to whether the requested actions should be taken must be based on a finding as to whether discrimination occurred. Petitioner describes two specific incidents as alleged discrimination, namely (1) his suspension without pay in August 1993, and (2) his termination by Licensee in June 1994.

With respect to Petitioner's request for immediate reinstatement, the NRC informed him in a letter dated September 21, 1993, that the NRC has no authority to order a direct personal remedy such as reinstatement of an employee and that, if the Petitioner sought reinstatement, he should file a complaint with the Department of Labor (DOL). This response referred to the Petitioner's request for reinstatement following his suspension, but it also applies to reinstatement of Petitioner following his termination. Therefore, the Petitioner's August 17, 1994 request that NRC reinstate the Petitioner to his position at Millstone is denied.

In his letters of August 22, 1993, and August 17, 1994, Petitioner describes his suspension without pay. The NRC's Office of Investigations (OI) investigated this allegation and concluded, in a report dated April 18, 1995, that the allegation that this suspension was for discriminatory reasons was not substantiated (1-93-047R). This conclusion was based on, among other things, a record of Petitioner's history of attendance problems, his excessive sick leave, and his continued problems controlling his temper and abrasive personality. The Petitioner had also been given several verbal and written warnings for similar conduct prior to being suspended.

With respect to Petitioner's termination which he described in his letter of June 29, 1994, OI concluded that the allegation that his termination was for discriminatory reasons was not substantiated. Specifically, OI cited Petitioner's poor performance, insubordination, and attendence problems, noting that the act that caused the Petitioner to be terminated (being outside of the protected area and absent from his work station without supervisory approval) was similar to the act which caused the Petitioner to be suspended without pay in August 1993. Petitioner also filed a complaint with the DOL concerning his termination and the DOL Area Director notified Petitioner on September 22, 1994, that his complaint was being dismissed on the basis that he had failed to make a *prima facie* showing of discrimination. The Area Director stated: "Our finding is that the firm would have reached the same decision with respect to your termination even in the absence of your protected conduct and activities." Petitioner has appealed this decision to the DOL Administrative Law Judge but, to date, there has been no decision on the matter.[1]

III. CONCLUSION

As explained above, neither the conclusions of the Department of Labor nor the report of the NRC Office of Investigations support Petitioner's claim that he was subject to discrimination. From our review of the DOL and OI findings, we have concluded that the matters complained of by Petitioner did not involve discrimination or a violation of 10 C.F.R. § 50.7. In the absence of sufficient evidence of a violation involving discrimination, there is no basis for the NRC to take the enforcement actions requested by Petitioner. Therefore, the petition filed on August 22, 1993, as supplemented by letters dated October 19, 1993, June 29, 1994, and August 17, 1994, is denied.

A copy of this Decision will be filed with the Secretary of the Commission for the Commission to review in accordance with 10 C.F.R. § 2.206(c). As provided by that regulation, the decision will constitute final action of the Commission 25 days after issuance, unless the Commission, on it own motion, institutes a review of the Decision within that time.

James Lieberman, Director
Office of Enforcement

Dated at Rockville, Maryland,
this 2d day of August 1995.

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[1] In accordance with its normal practice, the Staff will monitor the DOL process and will consider the need for enforcement action if DOL finds that discrimination occurred.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF ENFORCEMENT

James Lieberman, Director

In the Matter of **Docket Nos. 50-245**
50-336

NORTHEAST NUCLEAR ENERGY COMPANY
(Millstone Nuclear Power Station, Units 1 and 2) **August 2, 1995**

The Director of the Office of Enforcement has denied petitions filed by Anthony J. Ross requesting that the NRC take escalated enforcement action with regard to violations at Millstone Nuclear Power Station arising from alleged discriminatory acts committed by his supervisors. Mr. Ross asks that the NRC issue Severity Level II and III violations and other sanctions against the supervisors who committed the alleged acts of discrimination, and that Severity Level I violations be issued against senior managers for failing to rectify the problem. The reasons for the denial are fully set forth in the Decision.

DIRECTOR'S DECISION UNDER 10 C.F.R. § 2.206

I. INTRODUCTION

On August 7, 1993, Anthony J. Ross (Petitioner) filed a request for enforcement action pursuant to 10 C.F.R. § 2.206 (petition). The petition requested that the Executive Director for Operations take escalated enforcement action with regard to alleged violations at Millstone Nuclear Power Station. Specifically, Petitioner requested that a Severity Level II violation be issued against his department manager and a Severity Level III violation be issued against his first-line supervisor for alleged violations of the provisions of 10 C.F.R. § 50.7, that sanctions be instituted against these individuals for engaging in deliberate misconduct in violation of 10 C.F.R. § 50.5, and that the first-line supervisor be removed from his position until a satisfactory solution to the problem can be achieved. The NRC acknowledged receipt of this petition on September 1, 1993 (58 Fed. Reg. 47,769 (Sept. 10, 1993)).

On May 23, 1994, Petitioner filed another petition, requesting that the NRC issue a Severity Level II violation and other sanctions against the Maintenance Manager at the Millstone plant (Unit 1) and remove the Maintenance Manager from his position until resolution of the issues raised in his complaint. The NRC acknowledged receipt of this additional petition on June 16, 1994 (59 Fed. Reg. 32,246 (June 22, 1994)). This additional petition was supplemented in an August 17, 1994 letter requesting that Severity Level I violations and other sanctions be issued against the Senior Vice President and the Chief Executive Officer at Millstone for their knowing failure to rectify the alleged harassment and discrimination issue and that these individuals be removed from their positions until a satisfactory solution to the problem can be achieved.

Letters dated October 12, 1993, and August 4, 1994, were received from Northeast Nuclear Energy Company, providing information concerning this petition.

II. BACKGROUND

As a basis for his August 7, 1993 request, Petitioner stated that he had been subjected to acts of harassment, retaliation, and discrimination since reporting to the Nuclear Safety Concerns Program that he observed his first-line supervisor performing work on a 10 C.F.R. Part 50, Appendix R, emergency light without a work order. He alleged that the discriminatory acts were committed by his department manager and first-line supervisor. Petitioner did not provide details concerning the specific acts of harassment, retaliation, and discrimination that he had experienced.

As a basis for his May 23, 1994 request, Petitioner stated that the Maintenance Manager told him that Petitioner was "obligated" to share his safety concerns with the Maintenance Manager. He alleged that this statement was a violation of 10 C.F.R. §§ 50.5 and 50.7 and that the maintenance manager was inhibiting the free flow of information.

In his August 17, 1994 supplement to the petitions, Mr. Ross alleged that the Chief Executive Officer and Senior Vice President at Millstone nuclear plant were aware of the harassment and intimidation that he had experienced and they had done nothing to rectify the problem. He also stated that an unjust written reprimand was written by the Maintenance Manager about him on December 13, 1993, and that his annual review was lowered by the Maintenance Manager, all of which was alleged to be further evidence of harassment and discrimination by management against the Petitioner.

III. DISCUSSION

Petitioner bases his requests for sanctions against individuals on his assertion that he was a victim of discrimination. Therefore, the decision as to whether the requested actions should be taken must be based on a finding of whether his claims of discrimination are substantiated. In his letters of August 7, 1993, and May 23 and August 17, 1994, which contain the substance of his petition, Mr. Ross discusses three specific instances of discrimination. While the August 7 letter does not allege any specific instances of discrimination, his May 23 letter alleges that he was discriminated against when the Maintenance Manager told him he was obligated to bring his safety concerns to the Maintenance Manager. In his August 17 letter, Petitioner discusses an "unjust written reprimand" and the "lowering of [his] annual review grades." While certain other allegations by the Petitioner were received by the NRC and not specifically included by him in his petition, some of these other allegations were considered by the Office of Investigations (OI) in its investigation (OI No. 1-93-044R) into Petitioner's claim of discrimination.[1]

Petitioner also filed five complaints with the U.S. Department of Labor (DOL) alleging discrimination. These complaints do not bear directly on the specific instances of discrimination claimed by Mr. Ross in his petitions, but were considered in formulating this Director's Decision since such allegations bear on Petitioner's claim of continuing acts of harassment, retaliation, and discrimination.

The Department of Labor received a complaint from Petitioner on August 2, 1994. Following a finding by the DOL Area Director on August 15, 1994, that Petitioner had not made a *prima facie* showing of discrimination, this complaint was dismissed by a DOL Administrative Law Judge (ALJ) in a Recommended Decision and Order Granting Motion for Summary Judgment dated March 1, 1995 (94-ERA-039). This case awaits final disposition by the Secretary of Labor. Complaints received by DOL on August 18 and December 14, 1994, were dismissed by the Area Director on September 29, 1994, and January 17, 1995, respectively, for failure to make a *prima facie* showing of discrimination. NRC records reflect an appeal only of the second of these complaints. The DOL ALJ recently recommended that that appeal and complaint be dismissed with

[1] Separate from the 2.206 process, and subsequent to his filing of the petitions addressed herein, Mr. Ross has raised some additional allegations concerning discrimination. These issues are pending before the Staff.

prejudice.[2] Complaints received by DOL on January 18 and 26, 1995, initially were dismissed by the Area Director for failure to make a *prima facie* showing. Both these decisions were appealed and, following a notification from Petitioner that he intended to withdraw his complaints, the ALJ recommended dismissal of the complaints on May 2, 1995. The ALJ's recommended dismissals in these cases, 95-ERA-025 and 95-ERA-027, were approved by the Secretary of Labor, who dismissed the cases on June 9, 1995.

On May 26, 1995, the NRC Office of Investigations (OI) issued a report on its investigation into Petitioner's allegations of harassment and discrimination that are the subject of the instant petitions. As to Petitioner's claim that he was told that he was obligated to report safety concerns to the Maintenance Manager, OI did not substantiate that this constituted discrimination. The Petitioner, in fact, provided a tape of his conversation with the Maintenance Manager on this matter and the Staff has reviewed both the tape and a transcript of the meeting reflected on the tape. It is true that, in one instance, the Maintenance Manager stated that Mr. Ross had an obligation to bring his safety concerns to Licensee management, but Petitioner responded that he had a choice to provide his concerns to NRC or to the Licensee and that, based on his past experiences, he chose to take his concerns to the NRC. From our review of the taped record of this meeting, it is clear that the Maintenance Manager was stressing the importance of reporting safety concerns to the Licensee so that the concerns could be addressed in a timely manner (and prior to returning equipment with potential problems to an operable status), but he did not direct that concerns be brought to Licensee management,[3] nor did he indicate that the Petitioner could not report his concerns to the NRC instead of the Licensee. Based on our review of the tape, we cannot conclude that the Maintenance Manager was attempting to dissuade Petitioner from going to the NRC.

With respect to Petitioner's allegation that he had been given an unjust written reprimand, OI reviewed Licensee records and letters documenting the reasons for the disciplinary action taken against the Petitioner. The OI report referred to a Licensee investigation that noted that an event that resulted in the loss of availability of a critical safety system was attributed to Petitioner's inattention to detail. The OI report also noted that Petitioner had previously received a

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[2] *See* 95-ERA-17, Order Recommending Dismissal of Complaint, April 28, 1995.

[3] This appears to be different from the situation in *Saporito v. Florida Power & Light Co.* (89-ERA-007, 89-ERA-017), a case involving threats and actual disciplinary action against the employee by the licensee when the employee refused to comply with management's direction that he disclose safety concerns to management. In the instant case, no action was taken against Petitioner when he indicated his preference to go to the NRC. Rather, it is clear from the tape of the meeting that the Maintenance Manager was concerned about Petitioner's safety concerns with equipment that the Licensee was about to declare operable and the manager was urging Petitioner to disclose his concerns so that they could be addressed. Although the Petitioner indicated that he would take his concerns to the NRC, he did not indicate when that might occur or whether the Licensee would ever be apprised of the concerns. The Maintenance Manager emphasized the Licensee's responsibility for safe operation of the facility and urged the Petitioner to ensure that the Licensee was made aware of potential safety problems.

verbal reprimand and that it was customary for this Licensee to issue a written reprimand to individuals who had already received verbal reprimands for past events. The OI report concluded that the Petitioner's allegation could not be substantiated.

OI also investigated Petitioner's allegation that his annual review grades were lowered due to his raising safety concerns. OI noted that Petitioner had been promoted from Electrician B to Electrician A between 1992, when Petitioner received a very good appraisal, and 1993, when the grades in Petitioner's appraisal were lower. The OI report referenced Licensee managers who said that the position description for an Electrician A had more stringent performance criteria and that Petitioner's appraisal was not a derogatory appraisal. One Licensee manager referred to a category of the annual appraisal called "Dependability," and noted that the lower rating in that category reflected the excessive number of sick days taken by Petitioner. The OI report concluded that the allegation that the Petitioner's annual appraisal was lowered due to his reporting safety concerns could not be substantiated.

With respect to Petitioner's allegation that he received a half-day suspension and an oral reprimand for reporting safety concerns, OI reviewed Licensee records and letters documenting the reasons for the disciplinary action taken against the Petitioner. The Licensee pointed out that the disciplinary action was taken because Petitioner's relationship with his managers had deteriorated in 1993, including an incident in which Petitioner called his supervisor a liar. Based on this information, OI concluded that the allegation could not be substantiated.

OI also investigated Petitioner's allegation that his automatic pay raise was delayed due to his reporting a safety concern. The Licensee attributed the delay in Petitioner's pay raise to errors in the computerized personnel information system and said that when Petitioner pointed out the discrepancy, the Personnel Department looked into the matter, discovered additional errors, and corrected them. Additional problems were discovered in the administration of the proper procedure, e.g., whether a performance appraisal was necessary before a pay raise could be approved. Petitioner's pay raise was corrected and he was reimbursed retroactively. Based on the evidence developed during the investigation, OI concluded that the allegation that Petitioner's automatic pay raise was delayed due to his reporting a safety concern could not be substantiated.

IV. CONCLUSION

As explained above, neither the findings of the Department of Labor nor the investigation by the NRC's Office of Investigations support Petitioner's claim that he was subjected to discrimination. From our review of the DOL and OI findings, we have concluded that the matters alleged by the Petitioner did not involve discrimination or a violation of 10 C.F.R. § 50.7. In the absence of adequate evidence of a violation involving discrimination, there is no basis for the NRC to take the enforcement actions requested by Petitioner. Therefore, the petitions filed on August 7, 1993, and May 23, 1994, as supplemented by Petitioner's letter dated August 17, 1994, are denied.

A copy of this Decision will be filed with the Secretary of the Commission for the Commission to review in accordance with 10 C.F.R. § 2.206(c). As provided by that regulation, the Decision will constitute 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.

James Lieberman, Director
Office of Enforcement

Dated at Rockville, Maryland,
this 2d day of August 1995.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

William T. Russell, Director

In the Matter of **Docket No. 50-219**

GENERAL PUBLIC UTILITIES NUCLEAR CORPORATION
(Oyster Creek Nuclear Generating Station) **August 4, 1995**

The Director of the Office of Nuclear Reactor Regulation denies in part a petition dated September 19, 1994, filed with the Nuclear Regulatory Commission (NRC) by Reactor Watchdog Project, Nuclear Information and Resource Service (NIRS), and Oyster Creek Nuclear Watch (Petitioners), requesting that the NRC take action with respect to the General Public Utilities Nuclear Corporation (GPUN or Licensee) Oyster Creek Nuclear Generating Station (OCNGS). The petition requests that the NRC: (1) immediately suspend the OCNGS operating license until the Licensee inspects and repairs or replaces all safety-class reactor internal component parts subject to embrittlement and cracking, (2) immediately suspend the OCNGS operating license until the Licensee submits an analysis regarding the synergistic effects of through-wall cracking of multiple safety-class components, (3) immediately suspend the OCNGS operating license until the Licensee has analyzed and mitigated any areas of noncompliance with regard to irradiated fuel pool cooling as a single-unit boiling-water reactor (BWR), and (4) issue a generic letter requiring other licensees of single-unit BWRs to submit information regarding fuel pool boiling in order to verify compliance with regulatory requirements, and to promptly take appropriate mitigative action if the unit is not in compliance.

The December 13, 1994 Supplemental Petition requests that the NRC: (1) suspend the license of the OCNGS until the Petitioners' concerns regarding cracking are addressed, including inspection of all reactor vessel internal components and other safety-related systems susceptibie to intergranular stress corrosion cracking (IGSCC) and completion of any and all necessary repairs and modifications; (2) explain discrepancies between the response of the NRC Staff, dated October 27, 1994, to the Petition of September 19, 1994, and the time-to-boil calculations for the FitzPatrick plant; (3) require GPUN to produce documents for evaluation of the time-to-boil calculation for the OCNGS irradiated fuel pool; (4) identify redundant components that may be powered from onsite power supplies to be used for spent fuel pool cooling as qualified Class 1E systems; (5) hold a public meeting in Toms River, New Jersey, to permit presentation of additional information related to the petition; and (6) treat the Petitioner's letter of December 13, 1994, as a formal appeal of the denial of Petitioners' request of September 19, 1994, to immediately suspend the OCNGS operating license.

After 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, the petition is denied with respect to these requests because the issues raised by the Petitioners are being adequately addressed already. 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 NRC Staff's review regarding those matters. Petitioners' request for a public meeting and 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 by letter dated April 10, 1995.

PARTIAL DIRECTOR'S DECISION UNDER 10 C.F.R. § 2.206

I. INTRODUCTION

By letter dated September 19, 1994, Reactor Watchdog Project, Nuclear Information and Resource Service (NIRS), and Oyster Creek Nuclear Watch (Petitioners), submitted a petition pursuant to section 2.206 of Title 10 of the *Code of Federal Regulations* (10 C.F.R. § 2.206), requesting that the U.S. Nuclear Regulatory Commission (NRC) take action with regard to the Oyster Creek Nuclear Generating Station (OCNGS), operated by the GPU Nuclear Corporation (GPUN or the Licensee). By letter dated December 13, 1994, Petitioners supplemented the petition.

The September 19, 1994 Petition requests that the NRC (1) immediately suspend the OCNGS operating license until the Licensee inspects and repairs or replaces all safety-class reactor internal component parts subject to embrittlement and cracking, (2) immediately suspend the OCNGS operating license until the Licensee submits an analysis regarding the synergistic effects of through-wall cracking of multiple safety-class components, (3) immediately suspend the OCNGS operating license until the Licensee has analyzed and mitigated any areas of noncompliance with regard to irradiated fuel pool cooling as a single-unit boiling-water reactor (BWR), and (4) issue a generic letter requiring other licensees of single-unit BWRs to submit information regarding fuel pool boiling in order to verify compliance with regulatory requirements, and to promptly take appropriate mitigative action if the unit is not in compliance.

The December 13, 1994 Supplemental Petition requests that the NRC: (1) suspend the license of the OCNGS until the Petitioners' concerns regarding cracking are addressed, including inspection of all reactor vessel internal components and other safety-related systems susceptible to intergranular stress corrosion cracking (IGSCC) and completion of any and all necessary repairs and modifications; (2) explain discrepancies between the response of the NRC Staff dated October 27, 1994, to the Petition of September 19, 1994, and the time-to-boil calculations for the FitzPatrick plant; (3) require GPUN to produce documents for evaluation of the time-to-boil calculation for the OCNGS irradiated fuel pool; (4) identify redundant components that may be powered from on-site power supplies to be used for spent fuel pool cooling as qualified Class 1E systems; (5) hold a public meeting in Toms River, New Jersey, to permit presentation of additional information related to the petition; and (6) treat the Petitioners' letter of December 13, 1994, as a formal appeal of the denial of the Petitioners' request of September 19, 1994, to immediately suspend the OCNGS operating license.

The September 19, 1994 Petition sought relief concerning safety-class reactor internal components based on the following premises: (a) the core shroud in General Electric BWRs is vulnerable to age-related deterioration; (b) twelve domestic and foreign BWR owners have found extensive cracking on welds of the core shroud; (c) only ten of thirty-six U.S. BWR owners have inspected their core shrouds and nine of the ten core shrouds had cracks; (d) nineteen of twenty-five selected BWR internal components are susceptible to stress corrosion cracking and six of nineteen are susceptible to irradiation-assisted stress corrosion cracking; (e) as the oldest operating General Electric Mark I BWR and the third oldest operating reactor in the United States, OCNGS has been subjected to the longest period of operational conditions that cause embrittlement and cracking; (f) the BWR Owners Group (BWROG) stated that cracking of the core shroud is a warning signal that additional safety-class reactor internals are increasingly susceptible to age-related deterioration; (g) cracking of any single part or multiple components jeopardizes safe operation of that nuclear station; (h) Oyster Creek did not inspect for core shroud cracking prior to the current refueling outage and other safety-class reactor internals have not been adequately inspected for cracking; and (i) a safety analysis has not been performed on the potential synergistic effects of multiple-component cracking.

The September 19, 1994 Petition also sought relief concerning fuel pool cooling design deficiencies, based on the following premises: (a) various design defects in BWR fuel pool cooling systems pose a significant increase in risk to the public safety and violate 10 C.F.R. § 50.59, 10 C.F.R. Part 50, Appendix A, Criterion 63, 10 C.F.R. Part 50, Appendix B, Criterion III, and Regulatory Guides 1.13, 1.89, and 1.97; (b) OCNGS is a single-unit facility with no adjacent units to rely upon in the event that a design-basis event were to disable the fuel pool cooling system; and (c) OCNGS has not docketed any material with regard to BWR design deficiencies identified in the 10 C.F.R. Part 21 Report of Substantial Safety Hazard (November 27, 1992) of Messrs. Lochbaum and Prevatte, and thus OCNGS may be in violation of NRC regulatory requirements.

The Petitioners assert the following bases to support their requests in the December 13, 1994 Supplemental Petition: (a) the October 27, 1994 letter of the NRC Staff, acknowledging receipt of the petition and denying the requests for immediate suspension of the operating license, failed to address concerns central to the petition, such as the Licensee's failure to recognize that IGSCC indicates that cracking could be occurring in additional safety-class reactor internal components and the Licensee's failure to perform inspections of all safety-class components to determine whether cracking is occurring; (b) recently discovered cracking in the top guide and core plates in foreign BWRs and cracking discovered on December 8, 1994, at the New York Power Authority's (NYPA's) FitzPatrick reactor underscore the Petitioners' concern that additional safety-class components at OCNGS are degrading; (c) the Licensee did not conduct an enhanced inspection of the core plate and top guide of the OCNGS facility during the current outage, despite notification by the General Electric Rapid Information Communication Service Information Letter (GE RICSIL) 071 dated November 22, 1994; (d) the Licensee, the NRC, and the BWR Owners Group (BWORG) have failed to provide an analysis of the synergistic effects of multiple-component cracking of additional safety-class reactor internal 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-to-boil 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 § XI reactor vessel internal inspections and BWROG-recommended inspections and took appropriate remedial action before restart 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

[1] In addition, the NRC Staff determined, in accordance with the guidance in NRC Management Directive 8.11, "Review Process for 10 C.F.R. 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.

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," 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 § XI, Division 1, Subsections IWA and IWB. The Licensee is also required by ASME Code § 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 Department, 101 Washington Street, Toms River, NJ 08753.

The Licensee performed inspections of the OCNGS reactor vessel and its internal safety-related components in accordance with the requirements of ASME Code § XI, and the NRC Staff has reviewed the Licensee's inservice inspection programs, as discussed below.

Cracking of the core spray piping was first detected during Licensee inspections at OCNGS in 1978, and its extent has been evaluated by the Licensee during each subsequent outage. The core spray piping was repaired in 1978 and 1980. Since that time, additional visual inspections by the Licensee have not identified any significant degradation of the piping or of the repairs made to the piping. The NRC's review of the Licensee's inspection results and disposition during the 14R outage, documented in NRC Inspection Report 50-219/92-22, dated March 19, 1993, and a letter to GPUN dated November 18, 1994, regarding the 15R inspection concluded that the Licensee inspections and dispositions of core spray system findings were appropriate.

The Licensee first detected cracking of the top guide in 1991 and has closely monitored it in successive outages. The NRC Staff conducted an inspection in June 1991, and concluded that the Licensee's disposition of the top guide crack as "acceptable as is" was adequate. The results of the inspection were reported in NRC Inspection Report 50-219/91-21, dated August 9, 1991. During an NRC inspection conducted in December 1992 and January 1993, the NRC Staff evaluated the results of a remote visual inspection of the top guide conducted by General Electric Corporation for GPUN. The Staff evaluated the quality of the Licensee's visual inspection of the top guide and agreed with the Licensee's determination that the top guide was acceptable to "use as is." The results of the inspection were reported in NRC Inspection Report 50-219/92-22, dated March 19, 1993.

The Licensee notified the NRC Staff during an October 11, 1994 telephone call that additional cracking in the top guide had been found. The Licensee also reported that cracks found in earlier inspections of the top guide had not shown any measurable growth. In addition, during the refueling outage for Cycle 15 of operation (15R refueling outage), which began in September 1994, the Licensee assessed all the cracks that had been identified to ensure they would not jeopardize the structural integrity or function of the top guide.

It should be noted that the location of the cracks that have been detected in the OCNGS top guide is different from that in the foreign reactor cited in the NIRS letter of December 13, 1994, and the subject of GE RICSIL-071. Moreover, both the top guide and the core plate at OCNGS are components of a GE BWR while the foreign plant is a non-GE BWR. Furthermore, the OCNGS core plate is bolted in place, and the top guide is restrained vertically by hold-down devices and horizontally by lateral supports. These configurations result in a highly redundant structure, and even if cracking similar to that observed in the foreign plant were to occur, it would not adversely affect the safety of the plant, and these components could still perform their safety-related functions.

The BWROG has addressed the issue of cracking in the internal components of reactor pressure vessels by recommending that BWR licensees perform inspections of various components pursuant to vendor recommendations of the General Electric Company. Among inspections recommended by the BWROG are examination of core spray spargers, core shrouds, top guides, return-line nozzles, and in-core instrumentation, which in the case of OCNGS are the intermediate-power-range monitors. The BWROG has also formed the Boiling Water Reactor Vessels & Internals Project (BWRVIP), chaired by five nuclear industry vice presidents, to develop a proactive program to address and mitigate cracking in reactor pressure vessel internal components. NRC Staff correspondence with the BWRVIP, Staff evaluation of the BWRVIP generic submittals, summaries of meetings with the BWRVIP, and Staff assessments of plant-specific submittals in regard to these subjects are also available to the public for review at the local public document room of each BWR plant.

The Licensee inspected the following safety-related components during the 15R refueling outage, which began in September 1994: core spray sparger and annular piping, steam dryer and separator assembly, core shroud head bolts, core support plate holddown bolts, guide rod and steam dryer support brackets, feedwater spargers, top guide assembly, four intermediate-power-range monitors, one low-power-range monitor, core shroud brackets, conical support to shell weld, and the core shroud. Cracking was observed on the core shroud and a steam dryer bracket, and required repairs to these components were made. Minor cracking was observed on the core spray piping, a tack weld on the keeper bolt of the feedwater spargers, and the top-guide cross beams. None of these cracks would have prevented the components from performing their normal operating and postulated accident functions. These indications were dispositioned as is. The Licensee submitted results of its core shroud inspection and its core spray sparger inspection to the NRC in separate letters, both dated November 3, 1994. As a result of a conference call on January 19, 1995, the Licensee submitted a summary of the results of its inspections of reactor vessel internal components performed during the 15R refueling outage. By a letter dated March 16, 1995, in accordance with 10 C.F.R. § 50.55a(g) and ASME § XI, IWA 6220 (1986 Ed. with no addenda), GPUN forwarded the reports of its inservice inspection activities conducted during the 15R refueling outage. In the report, GPUN lists the inspections performed and discusses unacceptable indications of certain components and their disposition. Inservice inspection of reactor vessel internal components is required by the ASME Code and the Licensee's inservice inspection program for future outages provides assurance that degradation of components will be detected and appropriate action will be taken. The documents discussed above are available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW, Washington, DC, and at the local public document room located at the Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

The Licensee's inspection of the OCNGS core shroud found that one of the ten circumferential welds (the H4 weld) had indications of substantial cracking. To ensure shroud integrity under all postulated accidents, the Licensee elected to install a modification, consisting of ten stabilizing tie-rods, designed to ensure that the core shroud would perform its design functions under normal operation and postulated accidents even if it were to develop 360° through-wall cracks. The NRC Staff reviewed this modification and issued a safety evaluation on November 25, 1994, which concluded that the core shroud modification proposed by the Licensee is acceptable and, therefore, is approved. The safety evaluation is also available at the public document rooms previously listed.

On the basis of the NRC Staff's review of various plant-specific and industry programs implemented by the Licensee, the NRC Staff concluded that the Licensee took appropriate actions to address embrittlement and cracking in, and thus to ensure the reliability of, the OCNGS reactor vessel internal components.

Based on the above, the Staff has concluded that suspension of the Oyster Creek Nuclear Generating Station operating license due to embrittlement and cracking of the reactor vessel internal components is not warranted. As stated previously, continued monitoring of reactor vessel internals as required by the

ASME Code and the Licensee's inservice inspection program will provide assurance that degradation of components will be detected and appropriate action will be taken.

B. Petitioners Request That the NRC Suspend the OCNGS Operating License Until the Licensee Provides an Analysis Regarding the Synergistic Effects of Through-Wall Cracking of Multiple Safety-Class Components

The majority of reactor internals are fabricated from high-toughness materials such as stainless steel and were designed with significant margins on allowable stresses. As such, cracking must be severe to adversely impact plant safety. It is unlikely that Licensee inspections would not find such severe degradation. In fact, identification and sizing of the cracks in the H4 location on the OCNGS core shroud are good examples of the effectiveness of the inspections. In addition, NRC Staff evaluation of the results from internals inspections performed to date at OCNGS resulted in the conclusion that ASME Code safety margins have been maintained.

The Licensee has not provided an analysis to NRC that addresses the synergistic effects of cracking in multiple safety-class components. The NRC Staff does not consider the lack of such an analysis to be a safety concern because of the inspection requirements that pertain to reactor internals and the results of inspections performed to date. *See* Section III.A, *supra.*

Continued monitoring of reactor vessel internals as required by the ASME Code and the Licensee's inservice inspection program will provide information about the structural integrity of reactor vessel internals in the long term. The NRC has asked the BWR Vessel Internals Project (BWRVIP), an industry group, to develop an assessment to address cracking in BWR reactor vessel internals. A report from the BWRVIP is expected on the long-term effects of reactor vessel internals cracking in late 1995. In addition, the NRC has undertaken a longer-term evaluation of the effects of cracking in multiple reactor vessel internal components that will be approached with appropriate treatment of the key variables (safety function, material susceptibility, loading, environment, etc.).

Based on the above, the Staff has concluded that suspension of the Oyster Creek Nuclear Generating Station license, due to the lack of an analysis of the synergistic effects of through-wall cracking of safety-class reactor internal components, is not warranted.

IV. CONCLUSION

The Petitioners requested that the NRC suspend the operating license of Oyster Creek Nuclear Generating Station until: (1) the Licensee inspects, repairs, or replaces all safety-class reactor internal components subject to embrittlement and cracking; and (2) the Licensee provides an analysis regarding the synergistic effects of through-wall cracking of multiple safety-class components. For the reasons discussed above, I conclude that the issues raised by the Petitioners are being adequately addressed and that there is no basis for suspending the OCNGS operating license or taking the other requested action. Accordingly, the Petitioners' above-referenced requests are denied.

A copy of this Partial Director's Decision will be filed with the Secretary of the Commission for review as stated in 10 C.F.R. § 2.206(c). This Decision will become the final action of the Commission 25 days after issuance unless the Commission, on its own motion, institutes review of the Decision within that time.

FOR THE NUCLEAR
REGULATORY COMMISSION

William T. Russell, Director
Office of Nuclear Reactor
Regulation

Dated at Rockville, Maryland,
this 4th day of August 1995.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

William T. Russell, Director

In the Matter of **Docket No. 50-293**
(License No. DRP-35)

BOSTON EDISON COMPANY
(Pilgrim Nuclear Power Station) **August 31, 1995**

The Director of the Office of Nuclear Reactor Regulation grants in part and denies in part a petition dated March 10, 1995, submitted by Mary Elizabeth Lampert and sixty-two other individuals pursuant to 10 C.F.R. § 2.206, and which requests action with regard to the Pilgrim Nuclear Power Station (Pilgrim), operated by the Boston Edison Company (Licensee).

Petitioners' request that the NRC not permit restart of Pilgrim until repairs are performed and corrective action is taken with respect to a number of certain reactor internals, parts, and components was denied because all potential problems identified by Petitioners had been satisfactorily addressed by the Licensee. Petitioners' request to terminate the NRC policy of issuing notices of enforcement discretion to reactor licensees was denied. Petitioners' request for a public meeting in Plymouth, Massachusetts, was granted.

DIRECTOR'S DECISION UNDER 10 C.F.R. § 2.206

I. INTRODUCTION

Ms. Mary Elizabeth Lampert and sixty-two other individuals (Petitioners) submitted a petition dated March 10, 1995, pursuant to 10 C.F.R. § 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 twenty-five 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 that 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, NRC 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 issues, 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 twenty-five 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 Ed. with Winter 1980 Addenda). This included all twenty-five 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 C.F.R. § 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 that 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 that 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, an 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 midsections 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 is attached to pins that 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 section 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 noncondensible gas dissolved in the reference leg of "cold-reference-leg" type water-level instruments. Under rapid depressurization conditions, noncondensible 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. Noncondensible 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 confirmed 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 noncondensible 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 shut down 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 backflush 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 that are not expected to have a high frequency of recurrence;
(b) safety systems and nonsafety systems are separated by design; thus, the availability and capability of the safety systems should not be impacted by errors in the nonsafety 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 C.F.R. § 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. Onsite 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 NRC Staff has concluded that the Petitioners have not identified any safety concerns at Pilgrim regarding spent fuel pool cooling during a LOCA/LOOP.

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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.

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 is 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 noncondensible 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, ensuring containment integrity. Additionally, the HWWVS design incorporates a device called a rupture disc, which provides an additional leaktight 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 ask 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 C.F.R. § 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 an 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 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 forty Model 1153B transmitters currently in service, fourteen medium-pressure transmitters and twenty-six 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 thirty-three 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 ask if their susceptibility to nodular corrosion would be corrected during RFO No. 10.

Nodular corrosion is a phenomenon seen in plants that have copper in the reactor water at a concentration in the 20-30 part 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 that 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 that 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.

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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.

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 that 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 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% of safety-related and 12% of non-safety-related fasteners were found to be nonconforming. However, only 2% 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 thirty-five safety-related and twenty-nine non-safety-related fasteners. Three safety-related and six 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 § III, Subarticle NCA-3800 and design specification 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 followup 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 that 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 1 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 that 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 that 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 that included general procurement, vendor audits, tests and/or inspections, obsolescence, 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 startup 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 usually 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 Petitioners, 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, Petitioners' 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. Petitioners' 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 C.F.R. § 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

William T. Russell, Director
Office of Nuclear Reactor
Regulation

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