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NUCLEAR REGULATORY COMMISSION ISSUANCES

November 1996

U.S. NUCLEAR REGULATORY COMMISSION

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NUCLEAR REGULATORY COMMISSION ISSUANCES

November 1996

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
Office of Information Resources Management
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
(301-415-6844)

COMMISSIONERS

Shirley A. Jackson, Chairman
Kenneth C. Rogers
Greta J. Dicus
Nils J. Diaz
Edward McGaffigan, Jr.

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

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COMMISSION

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

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Greta J. Dicus
Nils J. Diaz
Edward McGaffigan, Jr.

In the Matter of **Docket No. 55-21849-OT**

EMERICK S. McDANIEL
(Denial of Application for Reactor Operator License) **November 13, 1996**

On September 11, 1996, Mr. Emerick S. McDaniel filed a Petition for Review of the Initial Decision in this case, LBP-96-17, 44 NRC 79 (1996), in which the Presiding Officer rejected Mr. McDaniel's challenge to the NRC Staff's rejection of his claim that he had passed his reactor operator examination. The Commission denies the Petition for Review because it fails to raise any substantial question justifying Commission review as provided under 10 C.F.R. § 2.786(b)(4), *incorporated into Subpart L in* 10 C.F.R. § 2.1253.

ORDER

On September 11, 1996, Mr. Emerick S. McDaniel filed a Petition for Review of the Presiding Officer's Initial Decision in this case, LBP-96-17, 44 NRC 79 (1996), in which the Presiding Officer rejected Mr. McDaniel's challenge to the NRC Staff's rejection of his claim that he had passed his written examination to become a reactor operator at the Vogtle Electric Generating Plant. The Presiding Officer ruled that Mr. McDaniel had correctly answered less than 80% of the questions and had therefore failed the exam.

We deny the Petition for Review because it fails to raise any substantial question justifying Commission review as provided under 10 C.F.R. § 2.786(b)(4), *incorporated into Subpart L in* 10 C.F.R. § 2.1253. We see no basis to question the Presiding Officer's factual finding that Mr. McDaniel had failed the written exam. *See generally Kenneth G. Pierce* (Shorewood, Illinois), CLI-95-6, 41 NRC 381 (1995).

Mr. McDaniel's Petition for Review is therefore DENIED.

For the Commission

WILLIAM M. HILL
Acting Secretary of the
Commission

Dated at Rockville, Maryland,
this 13th day of November 1996.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Shirley Ann Jackson, Chairman
Kenneth C. Rogers
Greta J. Dicus
Nils J. Diaz
Edward McGaffigan, Jr.

In the Matter of | **Docket Nos. 70-7001**
70-7002

U.S. ENRICHMENT CORPORATION
(Paducah, Kentucky, and Piketon, Ohio) | **November 22, 1996**

The Commission denies two motions for reconsideration of CLI-96-10, 44 NRC 114 (1996), which rejected two petitions for review of an Initial Director's Decision approving certificates of compliance for the United States Enrichment Corporation's gaseous diffusion plants in Piketon, Ohio, and Paducah, Kentucky. The Commission also denies two petitions for review of the initial Director's decision and rejects a third petition for review as late-filed.

RULES OF PRACTICE: PETITION FOR REVIEW UNDER PART 76

To be eligible to petition for review of a Director's Decision on the certification of a gaseous diffusion plant, an interested party must have either submitted written comments in response to a prior *Federal Register* notice or provided oral comments at an NRC meeting held on the application or compliance plan. 10 C.F.R. § 76.62(c).

RULES OF PRACTICE: PETITION FOR REVIEW UNDER PART 76

Individuals who wish to petition for review of an initial Director's decision must explain how their "interest may be affected." 10 C.F.R. § 76.62(c). For guidance, petitioners may look to the Commission's adjudicatory decisions on standing. *See, e.g., Georgia Institute of Technology* (Georgia Tech Research Reactor, Atlanta, Georgia), CLI-95-12, 42 NRC 111, 115-17 (1995).

NEPA: ENVIRONMENTAL ASSESSMENT/ENVIRONMENTAL IMPACT STATEMENT

No environmental assessment or environmental impact statement is required for the issuance, amendment, modification, or renewal of a certificate of compliance of gaseous diffusion enrichment facilities pursuant to 10 C.F.R. Part 76. 10 C.F.R. § 51.22(c)(19). Although NRC regulations do not require a general review of the environmental impacts associated with the issuance of certificates of compliance, an environmental assessment of the impacts of compliance plan approval is required.

CERTIFICATION OF GASEOUS DIFFUSION PLANTS UNDER PART 76: ASSESSMENT OF ACCIDENTS

An analysis of potential accidents and consequences is required by 10 C.F.R. § 76.85 and should include plant operating history that is relevant to the potential impacts of accidents.

MEMORANDUM AND ORDER

I. INTRODUCTION

On September 19, 1996, the NRC published in the *Federal Register* (61 Fed. Reg. 49,360-63) notice of the certification decision of the Director, Office of Nuclear Material Safety and Safeguards (Director), for the U.S. Enrichment Corporation (USEC) to operate the two gaseous diffusion plants (GDPs), one located at Paducah, Kentucky (referred to hereafter as the Paducah plant), and the other at Piketon, Ohio (referred to hereafter as the Portsmouth plant). NRC also issued a Finding of No Significant Impact (FONSI) concerning NRC's approval of the compliance plans[1] prepared by the U.S. Department of Energy (DOE) and submitted by USEC.

[1] The compliance plans set forth USEC's plan and schedule for achieving full compliance with NRC regulatory requirements.

USEC, or any person whose interest may be affected and who had submitted written comments in response to the prior *Federal Register* notice on the application or compliance plan under 10 C.F.R. § 76.37, or provided oral comments at an NRC meeting held on the application or compliance plan under 10 C.F.R. § 76.39, were eligible to file a petition to the Commission requesting review of the Director's decision within 15 days after publication of the Director's decision. 10 C.F.R. § 76.62(c).[2]

The NRC received five petitions for review of the Director's decision. A previous memorandum and order issued by the Commission in this proceeding, on October 18, 1996 (CLI-96-10, 44 NRC 114), rejected two of these five petitions for failure to meet the eligibility requirements of section 76.62(c). The two rejected Petitioners have petitioned for reconsideration. The Commission's previous memorandum and order also addressed certain threshold procedural matters raised in the remaining petitions, denying a request for an additional period for seeking review and submitting comment on the Director's decision, and denying a request for expansion of the right to seek the Commission's review of the Director's decision to any person.

This Memorandum and Order addresses the two petitions for reconsideration and the remaining issues raised in the petitions not previously rejected. For the reasons set forth below, these petitions are rejected in their entirety.

II. PETITIONS FOR RECONSIDERATION

The Commission has received two petitions for reconsideration of the Commission's memorandum and order served October 18, 1996:

1. By a pleading dated October 24, 1996, Diana Salisbury, of Sardinia, Ohio, requested that the Commission reconsider its Memorandum and Order of October 18, 1996, and review her petition dated October 3, 1996, and amendment dated October 4, 1996.
2. By a pleading entitled "Verified Complaint, Administrative Petition for Action," dated October 25, 1996, Neilly Buckalew, Director, Kwanitewk NATIVE Resource/Network, of Meriden, New Hampshire, requested that the Commission reconsider its Memorandum and Order of October 18, 1996, CLI-96-10, and review their October petition.

The Commission rejected both of these Petitioners' petitions for review of the Director's decision for failure to comply with the eligibility requirements in

[2] Notice of receipt of the application had appeared in the *Federal Register* (60 Fed. Reg. 49,026) on September 21, 1995, allowing for a 45-day public comment period on the application and noticing public meetings to solicit public input on the certification. A second notice appeared in the *Federal Register* (60 Fed. Reg. 57,253) on November 14, 1995, providing for a 45-day public comment period on the compliance plan. Public meetings were held on November 28, 1995, in Piketon, Ohio, and on December 5, 1995, in Paducah, Kentucky.

section 76.62(c). That provision requires prior participation in the certification proceeding by submission of either written comments or oral comments at a public meeting. The Commission provided a full opportunity for members of the public to submit timely written or oral comments during the proceeding. *See* note 2, *supra.* The Commission explicitly informed the public of the requirement to submit written or oral comments in order to be eligible to petition for review of the Director's decision in the *Federal Register* notices. *Id.*

Both Petitioners cite 5 U.S.C. § 553(e), a provision of the Administrative Procedure Act (APA), giving interested persons the right to petition for the issuance, amendment, or repeal of a rule. However, if Petitioners wish to exercise their right to petition for a change in the eligibility rule in section 76.62(c), they must do so in a petition for rulemaking under 10 C.F.R. § 2.802, stating their basis for requesting the rule change.

Additionally, the cited section of the APA is inapplicable to support Petitioners' right to petition for review of the Director's decision, which is in the nature of an adjudication, not a rule.

Petitioners do have the right to challenge the Commission decision dismissing their petitions for review of the Director's decision. However, Petitioners have presented no information that would indicate that the previous decision was in error and have presented no new information that would justify reconsideration.

Petitioners also state various arguments to support the assertions that they are persons "whose interest may be affected" (section 76.62(c)) and therefore are eligible to petition for review of the Director's decision. However, since Petitioners have not satisfied the prior participation requirement stated in the rule,[3] we need not address these arguments.

Therefore, these petitions are denied.

III. PETITIONS FOR REVIEW

The three remaining petitions and related NRC actions to date are as follows:

1. By letter dated September 30, 1996, Vina K. Colley of McDermott, Ohio, who serves as President of P.R.E.S.S., Portsmouth-Piketon Residents for Environmental Safety and Security, petitioned for Commission review of the Director's decision. Her petition (hereafter referred to as the "Colley petition") was docketed at the NRC on October 4, 1996. Ms.

[3] Petitioner Salisbury asserts that section 76.62(c) is grammatically constructed to create two separate categories of eligibility: "The corporation or any person whose interest may be affected" and "who had submitted comments in response to the *Federal Register* notice. . . ." However, it is evident by the placement of the comma after "Corporation," the lack of a comma after the clause "any person whose interest may be affected," and the use of the pronoun "who" rather than "any person who" in the clause about submission of comments, that Petitioner's interpretation is in error.

Colley had spoken at the NRC's public meeting in Piketon, Ohio, on November 28, 1995, regarding the application and compliance plan. On October 4, 1996, the Secretary of the Commission served a copy of the Colley petition on USEC and persons who had provided written comments on the application or compliance plan during the comment period or had provided oral comments at a meeting held on the application and compliance plan. The Secretary invited those served to file comments on Ms. Colley's petition by October 15, 1996. Comments were subsequently received from Ronald Lamb,[4] dated October 14, 1996; from Jotilley Dortch,[5] dated October 15, 1996; and from USEC, dated October 15, 1996.

2. By letter dated October 2, 1996, two individuals, Mark Donham and Kristi Hanson, of Brookport, Illinois, petitioned for review. Mr. Donham had spoken at the NRC's public meeting in Paducah, Kentucky, and Donham and Hanson had jointly submitted written comments during the comment period. The petition (hereafter referred to as the "Donham/Hanson petition") was docketed at the NRC on October 8, 1996. On October 9, 1996, the Secretary served the petition on the service list, and invited those served to comment on this petition by October 21, 1996. Comments were subsequently received from Jotilley Dortch (*see* note 5), dated October 15, 1996; from USEC, dated October 21, 1996; and from the U.S. Environmental Protection Agency (EPA), Region 5, dated October 22, 1996.[6]
3. By letter dated October 10, 1996, A.B. Puckett, member of the Coalition for Health Concern, of Kevil, Kentucky, petitioned for review. Mr. Puckett had spoken at the public meeting in Paducah, Kentucky.

IV. DISMISSAL OF LATE PETITION

The petition of A.B. Puckett was dated October 10, 1996, and postmarked October 14, 1996. Under section 76.62(c), the 15-day period for petitions for review of the Director's decision commenced with the publication of the *Federal Register* notice on September 19, 1996, and concluded on October 4, 1996. Therefore, Mr. Puckett's petition was untimely filed.

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[4] The response of Ronald Lamb stated its support of the objections of the Colley petition without further elaboration.

[5] Although the letter filing of Jotilley Dortch purports to be a response to both the Colley petition and the petition filed by Mark Donham and Kristi Hanson, it does not address the issues raised in either petition, but instead raises new issues. Therefore, this correspondence will not be considered as a response to the petitions but will be forwarded to the Staff for appropriate response.

[6] The response of EPA, Region 5, commented on the Donham/Hanson petition's request for more time for public comment. This portion of that petition was considered and denied in CLI-96-10.

This Petitioner does not even refer to the untimely filing, let alone attempt to establish that there is good cause to accept the late filing. *See* 10 C.F.R. § 76.74(b) ("good cause" required to extend time deadlines in Part 76). There is no other indication in the petition itself of late information that would plausibly excuse the late filing. Furthermore, the petition, which deals with the impacts of uranium mining and milling and of dumping nuclear waste on Indian lands, raises no issues that are directly relevant to this proceeding.

We find that Petitioner has not established and we cannot otherwise conclude that there was good cause for the late filing. Therefore, the substantive matters in the petition of A.B. Puckett will be referred to the Staff for an appropriate response and will not be considered by the Commission as a petition for review of the Director's decision.

V. STANDING OF PETITIONERS

Section 76.62(c) limits eligibility to petition for review of the Director's decision to those persons "whose interest may be affected" and who also have previously participated in the proceeding by submitting written comments or oral comments at any meeting on the application or compliance plan. The phrase "whose interest may be affected" is also used in section 189a of the Atomic Energy Act concerning those who have a right to a hearing in certain proceedings.

Neither of the petitions before us directly addresses the "interested person" issue in sufficient detail. We note, however, that Petitioners did participate in the Piketon and Paducah public meetings and appear to live in the vicinity of the plants. In addition, this is the first time the Commission has entertained petitions under Part 76 and Petitioners, who are appearing *pro se*, may not have understood their obligation to explain their "interested person" status. Thus, we are unwilling to hold Petitioners to a formalistic pleading-type requirement and instead will assume that Petitioners are "interested persons." We therefore will consider the merits of the Colley petition with regard to the Portsmouth plant and the Donham/Hanson petition with regard to the Paducah plant.

The Commission cautions, however, that in future Part 76 certification decisions, it will expect Petitioners more specifically to explain their "interested person" status. For guidance, Petitioners may look to the Commission's adjudicatory decisions on standing. *See, e.g., Georgia Institute of Technology* (Georgia Tech Research Reactor, Atlanta, Georgia), CLI-95-12, 42 NRC 111, 115-17 (1995).

VI. ANALYSIS AND RESPONSE TO ISSUES RAISED IN THE COLLEY PETITION

The Colley petition enumerated six "comments, objections and petitions for action" which we will refer to and treat as Issues 1 through 6 using Ms. Colley's nomenclature (*see* Colley Petition at 1). Issues 1, 2, and 3 dealt with threshold procedural matters — extending the 15-day time limit for filing a timely petition for Commission review of an initial Director's decision, and expansion of the categories of persons eligible to file a petition for review of the Director's decision — and those requests were denied in the previous Commission memorandum and order dated October 18, 1996 (CLI-96-10, *supra*). The remaining Issues 4, 5, and 6 are addressed here.

A. Colley Issue 4: Petition for NRC to Hold National Public Hearings

Petitioner asks that NRC hold public hearings nationally regarding the continued operation of the GDPs in Ohio and Kentucky. This request is made as an adjunct to Petitioner's requests, previously denied, for extension of the time period for the filing of petitions and for expansion of the right to file petitions to any person. Petitioner supports her request with arguments that the continued operation of the GDPs will affect all U.S. taxpayers and that "it is *U.S. taxpayer dollars that have provided the capital for these plants to operate* for the last 40 years and will *continue to provide the necessary funds* to maintain operation of these plants. . . ."

Prior to issuing the certification decision, the Staff provided a broad opportunity for public comment by publishing *Federal Register* notices concerning the receipt of USEC's applications and compliance plans, and holding public meetings in the vicinity of each site. *See* note 2, *supra.* From a health and safety perspective, it is the people who live in the vicinity of the facilities who may have an interest that might be affected. Accordingly, the NRC made special efforts to ensure that those people were informed.[7] We find that adequate opportunity for public participation in this proceeding has been provided, and that no reason is apparent either from the record or from Petitioner's arguments that additional hearings would produce any significant additional information. Therefore, the request for additional public hearings is denied.

[7] We note that the Staff used several additional means to publicize the certification process, obtain public comments, and coordinate with other interested agencies. These included: establishment of local public document rooms near each site, press releases, notices of technical meetings with USEC open to the public, paid advertisements in local newspapers, media interviews, individual letters seeking comments from interested parties, and meetings with labor union officials, local government officials, DOE, the EPA, and the Occupational Safety and Health Administration (OSHA).

B. Colley Issue 5: Objection to the Finding of No Significant Impact Regarding USEC's Compliance Plan

Petitioner's Issue 5 is supported by nine individual bases that Petitioner labels (a) through (i). We adopt the same labeling for convenience, and address each individual basis below.

We first address a fundamental premise raised by the Petitioner regarding the FONSI. Petitioner's argument apparently rests on the belief that the environmental assessment (EA) of the impacts of the proposed compliance plan approval should encompass all the impacts of ongoing operations, not just impacts associated with compliance plan approval. We note here that several of Petitioner's nine bases for this issue assert that there is an inadequate evaluation of the environmental impacts of ongoing or past operations, and none of the nine bases focus on any impact associated with compliance plan implementation.

As part of the same rulemaking that promulgated 10 C.F.R. Part 76, 10 C.F.R. Part 51 was modified to provide a categorical exclusion from the requirement for an environmental impact statement or EA for the "issuance, amendment, modification, or renewal of a certificate of compliance of gaseous diffusion enrichment facilities pursuant to 10 C.F.R. Part 76." 10 C.F.R. § 51.22(c)(19). This action was taken because the two GDPs had already been subject to environmental review pursuant to the National Environmental Policy Act of 1969 (NEPA) inasmuch as DOE had prepared an environmental impact statement for the Portsmouth plant, and an EA for the Paducah plant. After review of the DOE environmental analyses, and the current operations of the plants, the NRC concluded that there were no significant differences in current operations that would result in significantly different environmental impacts from those already evaluated by DOE. *See* Supplementary Information, 59 Fed. Reg. 48,944, 48,958, (Sept. 23, 1994). The NRC further concluded that since the Commission's certification requirements were intended to be at least as stringent as existing DOE requirements, certification issuance, modification, or amendment would not allow the GDPs to operate in such a way as to result in any adverse environmental effects greater than those that currently existed or would be expected absent NRC oversight, and would not have a significant effect on the human environment.

Therefore, no general review of environmental impacts associated with issuance of the certificates of compliance, as proposed by the Director's decision, is contemplated or required by NRC regulations. However, the categorical exclusion does not extend to approval of the compliance plans, and, therefore, an EA was performed by the Staff for that purpose.

The *Federal Register* notice publishing the Director's decision included an EA of the environmental impacts associated with the contemplated approval of the USEC compliance plans. Examples of specific topics related to the compliance plan, and included in the EA, are filter testing and air sampling.

On the basis of the EA, the Staff determined that there would be no significant impact associated with approval of the compliance plans and issued the FONSI.

Therefore, the Petitioner's basic premise is flawed in that it wrongly presupposes that the Staff was required to perform a broad environmental review of ongoing GDP operations, when in fact only an assessment of the impacts of compliance plan approval is required.

We now turn to Petitioner's individual bases:

1. Colley Issue 5(a): The Notice (FONSI) Is Deficient in Not Reviewing or Accounting for the Impacts Resulting from Privatization of USEC

Petitioner asserts that NRC must review and account for the "impacts, changes, and full ramifications on the operation of the two plants and environmental compliance . . . from the actual process of privatization." Petitioner also asserts that "[t]he effects of privatization on environmental compliance must be fully analyzed including the economic ability of USEC to fully comply with environmental standards over the next projected 50 years of operation."

Petitioner's broad allegations do not contain enough detail to state a meaningful objection.[8] More importantly, as noted above, the EA or FONSI are required to consider only environmental impacts associated with approval of the compliance plans. Since the possibility of future privatization falls outside the scope of the compliance plan and this certification, the Petitioner's challenge is rejected on that basis.

2. Colley Issue 5(b): Fugitive Uranium Deposits Pose Risks of Criticality and Should Be Cleaned Up Before Certification

Petitioner is apparently referring to existing uranium deposits in plant equipment, and asserting that they could worsen with continued plant operation and pose a risk of a nuclear criticality. Petitioner refers to a National Academy of Sciences report, *Affordable Cleanup* (National Research Council, 1996), noting that cleanup began in 1991 but is not complete. Petitioner asks that certification be withheld until cleanup of the uranium deposits is completed, in order to protect worker safety and the public health.

It is recognized that uranium deposits can form in process equipment and piping in the GDPs. USEC is required to follow Technical Safety Requirements which provide for surveillance, detection, and safe management of uranium deposits. For example, Portsmouth Technical Safety Requirement 2.7.3.14

[8] Insofar as the Petitioner's complaint may be read as a broad objection to privatization, Congress has spoken on this issue. In the USEC Privatization Act (Pub. L. No. 104-134), Congress directed USEC to implement a privatization plan to transfer the corporation to private ownership.

requires: (1) quarterly surveys for uranium deposits in the X-326 cascade facility, (2) measures to ensure criticality safety if identified deposits are above a certain size, and (3) actions to safely stabilize or remove deposits.

The cleanup that began in 1991 and is referred to in the National Research Council's report is the DOE high-enriched uranium suspension program. When it was determined that additional high-enriched uranium was no longer needed for defense purposes, a decision was made that the Portsmouth high-enrichment equipment could be retired from service. DOE has informed the NRC that significant deposits have been removed and the equipment has been retired in place.

Petitioner has offered no substantial basis for finding that the issue of uranium deposits has not been appropriately addressed by USEC and reviewed by the Staff. Therefore, we reject this issue as a basis for challenging the Director's decision.

3. *Colley Issue 5(c): Certification Should Be Withheld Until the Synergistic Impacts of Releases of Asbestos, Lead, Other Heavy Metals, and Uranium Are Analyzed*

Petitioner asserts that NRC has not reviewed the synergistic impacts of asbestos, lead, and other heavy metals, in addition to uranium, on workers or the public, and asks that certification be withheld until such impacts are fully documented and analyzed. (Petitioner also raises the issue of synergistic effects under Issue 5(f) below.)

The Energy Policy Act of 1992 required NRC to establish standards for the GDPs to protect the public health and safety from radiological hazards. The NRC Staff's review and the Director's decision are based on a determination that USEC's applications and compliance plans meet the standards NRC established for protection of public health and safety from radiological hazards associated with GDP operation. The basis for this determination is documented in the NRC Staff's Compliance Evaluation Reports (CERs).

The hazards from asbestos, lead, and heavy metals that Petitioner cites are regulated by OSHA and the EPA, and USEC must comply with OSHA and EPA regulations. Petitioner has not provided any information to indicate that these nonradiological substances are present in quantities that pose a health hazard, either by themselves or in combination with uranium, or that any such hazard falls under NRC jurisdiction over radiological hazards. Therefore, we reject Petitioner's request to withhold certification on account of synergistic impacts, and also reject this basis for finding the Staff's EA and FONSI defective.

4. *Colley Issue 5(d): Aging of Buildings Poses Significant Risks to Public Health, Worker Safety, and the Environment, Including Major Water Bodies*

Petitioner contends that the GDPs pose a significant contamination risk due to plant age, and that decontamination and decommissioning should commence immediately. However, Petitioner offers no information in support of her claim of significant risk. The report cited by Petitioner as supporting her position (*Affordable Cleanup,* National Research Council, 1996), addresses decommissioning issues but does not indicate that the operating plants pose a significant health risk.

Petitioner also alleges that there is a possibility of significant underground water contamination, and asserts that to allow the plants to operate in noncompliance will put major water bodies, including the Ohio River, at great risk. Petitioner provides no information in support of her argument and fails to demonstrate a relationship to the compliance plans or the Staff's EA or FONSI.

In its CER, the Staff determined that the Portsmouth effluent control program is in compliance with NRC requirements. Therefore, the Portsmouth compliance plan includes no requirement for new actions to control effluents. Petitioner does not challenge the Staff's findings in this regard.

For these reasons, we reject this basis for Petitioner's objection to the Staff's FONSI and the proposed Director's decision.

5. *Colley Issue 5(e): Decommissioning and Decontamination Budget Cuts Pose Risks to Public Health, Worker Safety, and the Environment*

Petitioner asserts that continued plant operation will increase onsite contamination, and that "recent D&D budget cuts" pose major risks. Petitioner concludes that the GDPs should not be allowed to continue to operate without secure financial resources for eventual cleanup.

Section 1403(d) of the Atomic Energy Act of 1954, as amended, provides that the responsibility for the decontamination and decommissioning costs that result from conditions existing before the transition date for the operations of USEC are the responsibility of DOE. Congress also created a specific fund and funding mechanism to pay these costs in section 1801 of the Act. Thus the bulk of the decommissioning costs are not the responsibility of USEC and have a mechanism for funding.

With regard to decommissioning costs that stem from USEC's operations, USEC has provided satisfactory financial assurance in compliance with 10 C.F.R. § 76.35(n) and this is discussed in the CERs, Chapter 14.

Therefore, we find that the Petitioner has not substantiated any basis for concern with this issue. This basis for Petitioner's objection to the Staff's EA and FONSI is rejected.

6. *Colley Issue 5(f): Serious Adverse Health Effects Have Occurred Offsite from Historical and Current Releases*

Petitioner alleges that serious offsite health effects may have occurred as a result of Portsmouth plant operations. Petitioner criticizes a study by the Agency for Toxic Substances and Disease Registry (ASTDR) as too narrow in scope, without providing any basis for that criticism. Petitioner refers to an unnamed report by "10 health planning agencies in the state of Ohio" and says the report found "*significant elevated cancer rates* in nine contiguous counties in southwest Ohio." Petitioner does not provide any specific information to link these alleged increased cancer rates with plant operations.

In its response to the Colley petition, USEC addressed this allegation by noting that, among other things, the Portsmouth plant is located in Pike County, and Pike County is not among those nine Ohio counties said by Petitioner to have higher cancer rates. USEC also notes that the ASTDR study on offsite health effects (which is criticized by Petitioner) concludes that "the Portsmouth Gaseous Diffusion Plant and its operations represent no apparent hazard to human health."

We find that the Petitioner has not provided a reasonable basis for her assertions. We also note that Petitioner fails to link these assertions regarding past occurrences with any aspect of the environmental impacts associated with approval of the compliance plans or the Staff's EA or FONSI, and we reject this issue as a basis for Petitioner's objection to the Staff's FONSI.

7. *Colley Issue 5(g): Inaccurate Assessment of Worker Deaths and Offsite Releases*

Petitioner asserts that a statement in the Staff's CER for the Paducah plant regarding incidents is untrue. The referenced statement is:

> no incidents at any of the GDPs have caused death or serious injuries to any plant personnel from exposure to radioactive materials or radiation nor have there been any incidents that have resulted in off-site release of radiation or radioactive materials that could cause committed doses in excess of established limits.[9]

Petitioner asserts that an unnamed document released in 1961 by Mr. Leo Goodman states that twelve cancer deaths among Portsmouth plant workers were

[9] Paducah CER at 8. The identical statement also appears in the Portsmouth CER.

linked with occupational exposure at the plant. Petitioner further alleges that a significant release of hexafluoride gas in the mid-1970s and numerous other incidents were hidden and denied by DOE. Petitioner then asserts that a thorough investigation of environmental releases and cumulative offsite impacts must be conducted before certification takes place.

USEC commented in its response that it was unable to locate a copy of the actual report released by Mr. Goodman, but contends that any confirmed causal relationship between occupational radiation exposure and cancer death resulting in twelve fatalities would be well known in the scientific literature and referenced in important treatises on the subject. USEC asserts that since this is not the case, even if there were twelve cancer fatalities, it has not been established that there is any cause-and-effect relationship between any worker radiation exposure and subsequent death by cancer.

USEC also points out that the mid-1970's incident that Petitioner refers to is documented in its application, in section 4.2 of the Portsmouth Safety Analysis Report. We note that the same incident, and others, are documented in section 1.5 of the Staff's CER for the Portsmouth plant.

We are satisfied that the issues of onsite and offsite releases have been adequately considered and analyzed in the CERs with respect to compliance with NRC standards. Petitioner has not demonstrated any basis for concluding that the potential impacts of releases have not been adequately assessed. Therefore, we reject this issue as a basis for any objection to the Director's decision or the Staff's EA and FONSI with respect to compliance plan approval.

8. *Colley Issue 5(h): Horizontal and Vertical Bedrock Fractures Are Not Well Understood and Pose Risk as a Migration Pathway*

Petitioner refers to a 1990 EPA document, "Environmental, Safety and Health Compliance Assessment of the Portsmouth Gaseous Diffusion Plant." We believe that the correct document is actually a 1990 DOE document by the same title. Petitioner quotes the report as saying that horizontal and vertical bedrock fractures beneath the plant may constitute a contamination migration pathway different from that determined by the monitoring well network, and that this potential pathway has not been completely assessed.

The finding in the 1990 document referred to by Petitioner actually relates to a groundwater quality assessment performed by DOE. DOE activities are not part of USEC's operations and are not subject to NRC jurisdiction. Petitioner does not allege that USEC is engaging in activities that could cause excessive groundwater contamination, and does not present any information to indicate that USEC is violating any NRC requirements related to groundwater contamination. Instead, Petitioner challenges the adequacy of DOE's ongoing program to evaluate existing groundwater contamination from other DOE activities at the Portsmouth site.

We find that Petitioner has not provided a reasonable basis to object to the Director's decision or the Staff's EA or FONSI related to compliance plan approval.

9. Colley Issue 5(i): Connection to Lack of Disposal for High-Level Waste

Petitioner objects to the continued operation of the GDPs because of problems associated with eventual disposal of the plants' output, after use as nuclear fuel, in the form of high-level waste.

The activities at the GDPs do not directly produce high-level radioactive waste and therefore this issue is not appropriate for consideration here. The use of fuel in nuclear reactors produces high-level waste, but NRC's licensing process for nuclear power plants has taken this issue into consideration. NRC has evaluated the issue of the adequacy of storage and disposal options for high-level radioactive waste and concluded that it has reasonable assurance that disposal is technically feasible and that the waste can be managed and stored in a safe manner until such disposal is available. *Rulemaking on the Storage and Disposal of Nuclear Waste* (Waste Confidence Rulemaking), CLI-84-15, 20 NRC 288 (1984); 55 Fed. Reg. 38,474 (Sept. 18, 1990).

We find Petitioner's issue to be outside the scope of this proceeding and reject it.

C. Colley Issue 6: Objection to Acceptance of DOE Overseeing Nuclear Safety

Petitioner objects to "acceptance of DOE overseeing nuclear safety currently and during the transition period to slated full privatization of the USEC. . . ."

The Petitioner errs in her understanding that DOE will retain regulatory jurisdiction over the GDPs until they are privatized. In fact, NRC plans to assume regulatory jurisdiction on March 3, 1997, following completion of the initial certification process. This schedule allows for a safe and orderly transition of regulatory authority from DOE to NRC and is unrelated to any privatization that may occur. We note that DOE's current role is as determined by law, not by NRC, and that Petitioner's objection is beyond the scope of NRC authority and unrelated to the Director's decision on compliance with NRC standards. Therefore, this issue is rejected.

VII. ANALYSIS AND RESPONSE TO ISSUES RAISED IN THE DONHAM/HANSON PETITION

The Donham/Hanson petition presents four separate issues, the first three of which are addressed below. The fourth issue is Petitioners' request for additional time to file comments on the Director's decision, beyond the 15-day period allowed by section 76.62(c); this request was addressed and rejected in the Commission's previous memorandum and order dated October 18, 1996.

A. Donham/Hanson Issue 1: Analysis of Offsite Radiological Consequences Pursuant to 10 C.F.R. § 76.85 Is Inadequate

In Issue 1, Petitioners challenge the NRC Staff's response to a comment previously made in the Petitioners' letter dated December 22, 1995. In that letter, Petitioners stated that they believed that "the cumulative effects of all the past releases in combination with any current or recent releases represents the primary hazard from the operation of the facility," and that consideration of such existing contamination should be required in assessment of the consequences of accidents. In response to this comment, in the Paducah CER, Appendix A, at A-5, the Staff replied:

> Cumulative effects from past operations are not part of an accident analysis. The primary hazard of this facility is the inadvertent release of UF6; the pathway of concern is inhalation. Exposure due to accumulation in the environment would be very small.

Petitioners object to this response and assert that section 76.85, "Assessment of accidents," requires relevant past operating history to be included in accident assessments.

Petitioners request that the Commission remand the application and require the Staff to fully address the offsite effects of releases of radioactive materials, including past releases. In support of their request, Petitioners state that: (1) there have been significant, regular releases of radioactive material offsite for the entire history of the facility; (2) there is evidence that radioactive substances, particularly plutonium and uranium in deer, are beginning to accumulate in the food chain off site; and (3) radioactive materials are being released into the environment through groundwater contamination off site.

The Commission notes that Petitioners have not challenged the Staff's conclusion that current releases are within regulatory limits but seem to believe that impacts from past operations should be assessed by the NRC and that this assessment is required by section 76.85. The Petitioners have misinterpreted the intent of section 76.85.

An analysis of potential accidents and consequences is required by section 76.85, and the analysis should include plant operating history relevant to the assessment. The accident analysis is performed "to establish the basis for limiting conditions for operation of the plant with respect to the potential for releases of radioactive material." Past operating history must be considered to make sure a potential accident scenario is not overlooked in the analysis. Past accidents are described in the Paducah Safety Analysis Report in section 4.1 and in the Staff's CER in section 1.5. Petitioners do not challenge either the adequacy of information concerning past accidents, or the spectrum of accidents considered, either in USEC's application or the Staff's CER.

We find that Petitioners have provided no basis to contradict the Staff's view that any residual contamination from past releases that is present in the environment is at such low levels that it would not be relevant to the analysis of potential impacts of accidents. For the foregoing reasons, this issue is rejected.

B. Donham/Hanson Issue 2: The FONSI Is Inadequate

Petitioners challenge the FONSI that the Staff prepared and issued in support of approval of the compliance plans. The Petitioners assert that since the EA and FONSI were prepared and issued with no notice to the community and no opportunity for public comment, they do not meet the intent of NEPA. The Petitioners further assert that NEPA requires a hard look at the cumulative effects from past, present, and future actions, including all of the waste management activities in combination with the operation of the plant and the implementation of the compliance plan.

The Commission's regulations governing implementation of NEPA are provided in 10 C.F.R. Part 51. The NRC's regulations do not require prior notice or opportunity for public comment in connection with the issuance of an EA or a FONSI, and Petitioners do not claim otherwise. (We note that opportunity for public comment was provided on the compliance plans that are the subject of the EA and that the opportunity to petition for review constitutes another limited opportunity for input from the public.) Therefore, to the extent that Petitioners challenge issuance of the EA and FONSI, they challenge the adequacy of NRC's regulations for implementing NEPA. Such challenges cannot be entertained here.

Petitioners also challenge the EA and FONSI on the basis of inadequate scope, claiming that they should evaluate the cumulative effects of all past, present, and future actions, and all waste management activities, in combination with operation of the plant and implementation of the compliance plan. We disagree. As we discussed above in connection with Colley Issue 5, the Staff need only address the environmental impacts associated with compliance plan approval. A broad assessment, such as that claimed by Petitioners to be required, would be directly at odds with the categorical exclusion from environmental review in 10 C.F.R. §51.22(c)(19), which exempts from environmental review the issuance of certificates of compliance under Part 76. Because Petitioners' request is at odds with NRC regulations, and because Petitioners fail to take issue with any particular aspect of the Staff's EA and FONSI related to the impacts of compliance plan approval or implementation, we find that Petitioners have failed to substantiate a basis for review of the Director's decision.

C. Donham/Hanson Issue 3: Request for Public Input and/or Notification Regarding Implementation of Compliance Plan Items on Seismic Upgrading

The Petitioners request that the Commission establish a mechanism that would allow public input into the implementation of the seismic upgrading described in the compliance plan. This request does not challenge the Director's decision in any respect and is rejected as a basis for requesting review. Mechanisms for public involvement in the certification process and in NRC's regulatory oversight of the GDPs are provided for by the Commission's regulations, as appropriate. In accord with the Commission's Open Meeting Policy, any meetings with USEC to discuss compliance plan items will be noticed and open for the public to attend, except for those at which proprietary or classified information is discussed. Also, as stated in 10 C.F.R. §§76.37 and 76.45, opportunities for public comment will be provided for any certification renewal or significant amendment of the certificates.

For the foregoing reasons:

1. The petition for reconsideration dated October 24, 1996, from Diana Salisbury, of Sardinia, Ohio, is denied.[10]

2. The petition for reconsideration dated October 25, 1996, from Neilly Buckalew, Director, Kwanitewk NATIVE Resource/Network, of Meriden, New Hampshire, is denied.[11]

3. The petition for review dated October 10, 1996, from A.B. Puckett, member, Coalition for Health Concern, of Kevil, Kentucky, is rejected as untimely. However, the substantive matters in the petition are referred to the NRC Staff for review and appropriate response. The comments from Jotilley Dortch, dated October 15, 1996, are also referred to the Staff for review and appropriate response.

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[10] The substantive matters in Petitioner's petition for review of the Director's decision were previously referred to the Staff for appropriate response.

[11] *See supra* note 10.

4. The petition for review dated September 30, 1996, from Vina K. Colley, President of P.R.E.S.S., Portsmouth-Piketon Residents for Environmental Safety and Security, of McDermott, Ohio, is denied in its entirety.

5. The petition dated October 2, 1996, from Mark Donham and Kristi Hanson, of Brookport, Illinois, is denied in its entirety.

Commissioner Dicus did not participate in this matter.

It is so ORDERED.

For the Commission

JOHN C. HOYLE
Secretary of the Commission

Dated at Rockville, Maryland,
this 22d day of November 1996.

Atomic Safety and Licensing Boards Issuances

ATOMIC SAFETY AND LICENSING BOARD PANEL

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LICENSING BOARDS

Cite as 44 NRC 249 (1996) LBP-96-24

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

James P. Gleason, Chairman
Dr. Jerry R. Kline
G. Paul Bollwerk, III

Thomas D. Murphy, Alternate Board Member

In the Matter of | **Docket No. 40-8027-EA (ASLBP No. 94-684-01-EA) (Source Material License No. SUB-1010)**

SEQUOYAH FUELS CORPORATION and GENERAL ATOMICS (Gore, Oklahoma Site Decontamination and Decommissioning Funding)

November 5, 1996

This decision approves a settlement agreement between the Nuclear Regulatory Commission Staff and General Atomics, thereby terminating this proceeding.

LICENSING BOARDS: RESPONSIBILITIES (SETTLEMENT OF CONTESTED PROCEEDING)

RULES OF PRACTICE: SETTLEMENT OF CONTESTED PROCEEDING

The licensing board's function in reviewing settlement agreements, as delineated in 10 C.F.R. § 2.203, calls for settlements to be approved by the board and an adjudication of any issues that may be required in the public interest to dispose of the proceeding.

LICENSING BOARDS: RESPONSIBILITIES (SETTLEMENT OF CONTESTED PROCEEDING)

RULES OF PRACTICE: SETTLEMENT OF CONTESTED PROCEEDING

The rationale for providing due weight to the position of the Staff may be grounded on the merited understanding that, in the end, the Staff is responsible for maintaining protection for the health and safety of the public and, in the absence of evidence substantiating challenges to the exercise of that responsibility, the Staff's position should be upheld.

LICENSING BOARDS: RESPONSIBILITIES (SETTLEMENT OF CONTESTED PROCEEDING)

RULES OF PRACTICE: SETTLEMENT OF CONTESTED PROCEEDING

The issue is not whether the matter before the Board presents the best settlement that could have been obtained. The Board's obligation instead is merely to determine whether the agreement is "within the reaches of the public interest." *United States v. Gillette Co.,* 406 F. Supp. 713, 716 (1975).

MEMORANDUM AND ORDER

(Approval of Settlement Agreement and Dismissal of Case)

Pending Board approval in this proceeding is a Settlement Agreement (agreement) between the Nuclear Regulatory Staff (Staff) and General Atomics (GA).[1] Objections to the agreement have been filed by Native Americans for a Clean Environment and the Cherokee Nation (Intervenors) and the State of Oklahoma (State) with responses thereto by the Staff and GA.[2] The Board approves the agreement herein and terminates the proceeding.

[1] Staff and General Atomics' Joint Motion for Approval of Settlement Agreement (July 11, 1996).

[2] Intervenors' Opposition to Joint Motion (August 9, 1996); State's Response to Joint Motion (September 5, 1996); Staff Reply to Intervenors' Opposition and State's Response (October 11, 1996; General Atomics' Response to Intervenors' and State's Opposition (October 11, 1996).

BACKGROUND

This case involves an October 15, 1993 Order by the NRC to Sequoyah Fuels Corporation (SFC) and its parent corporation, GA, holding both organizations responsible for decommissioning funding of SFC's licensed facilities in Gore, Oklahoma. The agreement, appended hereto, proposes *inter alia,* to release GA from liability in exchange for a payment of either $9 million or $5.4 million, the amount to be determined by Internal Revenue rulings on tax status of the payments. The Staff, through a GA-created trust fund arrangement, is to approve the distribution of the funds, with GA having no control over the fund or the payments deposited therein. The Joint Motion for Approval of the Agreement requests suspension of all discovery activities in the proceeding pending any further reviews of this decision.[3]

The Board has previously approved a settlement agreement submitted by the Staff and SFC. That order and agreement, wherein SFC pledges its net assets and revenues to decommissioning of its facility and which culminated in a dismissal of SFC from the proceeding, is presently under review by the Commission.[4]

The Intervenors and State assert the agreement before us neither meets the financial assurance regulatory requirements for decommissioning nor demonstrates that the public interest objectives of the 1993 Order are met. In sum, the parties request additional information concerning the agreement and an adjudication of its terms. *See* State Response at 13 and Intervenors' Opposition at 31.

DISCUSSION

The pending agreement reads it is in full settlement of the NRC's 1993 Order to GA with both signatories affirming it represents a good faith, voluntary, and major effort to resolve their differences. In its basic arrangement, the following provisions are stipulated:

GA to establish trust fund with $9 million contribution but obligated for only $5.4 million pending IRS tax rulings

Payments from trust fund to be approved by NRC alone

GA to refrain from interference with SFC settlement

Two GA Officers to resign from SFC Board of Directors

[3] In light of the decision herein, it is not necessary to act on this request.
[4] Memorandum and Order (October 26, 1995), LBP-95-18, 42 NRC 150 (1995).

Staff to rescind October 1993 Order and refrain from other action against GA based on SFC affiliation

Staff to forego any claim against GA based on de facto licensee theory

If agreement not upheld, funds to be returned and status quo of controversy restored

The State's Response[5]

It is argued that those responsible for causing pollution or allowing contamination to occur at SFC's site must bear the costs of remediation, not the State or its citizens. The State views the settlement as falling short of the mandate of 10 C.F.R. § 40.36 which requires those responsible for the contamination to provide financial assurance of decommissioning costs. It contends the Board must protect the public interest by declining to accept the settlement agreement. State Response at 4-5.

The State opines further it will be precluded from litigating additional liability claims against GA if the settlement agreement is accepted due to an unexplained claim of federal preemption under law. Moreover, the State contends, provisions of the Atomic Energy Act (42 U.S.C. § 2021) require the Board to take into consideration the State's interest in its public interest determination under 10 C.F.R. § 2.203. The State suggests the settlement agreement does not meet the public interest threshold because the "NRC Staff have made a 180 degree turn in position, from vigorous pursuit of enforcement to reluctant compromise in the face of a well financed corporate defense." *Id.* at 5-6. It is the State's view that a public interest determination by the Board should not be based on the practical and individual concerns raised by the Staff and GA, but rather on an analysis of the adequacy of information pertaining to whether the agreements provide adequate financial assurance to meet the requirements of 10 C.F.R. § 40.36 and the risks if the agreement does not ensure completion of decommissioning.

In support of its position of prematurity for the Board to find the agreement in the public interest, the State alleges the following:

(1) The settlement agreement between the NRC Staff and SFC, coupled with the present agreement allows the transfer of funds from SFC to GA which will not be available to pay for decommissioning costs;

(2) There is no accurate prediction of the final cost of decommissioning at the SFC site;

(3) SFC's ability to pay decommissioning costs is dependent upon its agreement with ConverDyn and there is little public knowledge of the

[5] Due to the lengthy course of this proceeding — encompassing a 3-year period — the arguments of the State and Intervenors are set forth herein in detail.

terms of that agreement. Moreover, GA may be able to influence payments by ConverDyn to SFC;

(4) The State, due to federal regulatory preemption, will have no recourse against GA if the settlement agreement is accepted;

(5) GA has retained the ability to receive profits and taxpayer funds from other government contracts beyond the amount it is obligated to pay under the terms of the settlement agreement;

(6) The NRC is in the best position to force GA to pay for decommissioning costs;

(7) The public interest would best be served if the question of NRC jurisdiction over GA is litigated to its fullest extent;

(8) In absolving GA of responsibility as a parent corporation, the Board is establishing a "chilling effect" upon any tribunal considering the future of the settlement agreement. *Id.* at 8-13.

The State requests the Board to order the Staff and GA to provide further information that will demonstrate that the settlement agreement meets the requirements of section 40.36; to determine whether additional discovery concerning financial information is needed; to allow "appropriate participation" by the State and Intervenors in the Board's public interest determination; to stay the effectiveness of the settlement agreement signed between the NRC Staff and SFC; to delay a final decision on both settlement agreements until a final decommissioning cost estimate is obtained; and if the GA agreement is accepted, to rescind and litigate the settlement agreement executed between the NRC Staff and SFC. *Id.* at 13-14.

The Intervenors' Response

Intervenors assert the Staff has traded its claim that GA must share the full cost of cleanup for the minor sum of $9 million or less. This settlement deprives the public of reasonable assurance that the site cleanup will be completed in a safe and effective manner and, this they contend, will pose a threat to the Intervenors' health. Accordingly, the settlement agreement must be rejected because it lacks essential information on funding of decommissioning and fails to provide sufficient information to allow a positive public interest finding. Intervenors' Response at 1-2, 13-14.

It is claimed the Board is obligated to ensure that Intervenors have a meaningful opportunity to participate in the proceeding for resolution of the conflict and is required by presidential directive to consult with tribal governments prior to taking action that affects them. Insufficient funding of the cleanup, it alleges, would have an adverse impact on the Cherokee Nation's sovereign interests in protecting its citizens, property, and trust lands. *Id.* at 14-17. And, in order to ensure meaningful participation, there must be sufficient disclosure to allow

Intervenors to evaluate the proposed settlement. Intervenors claim they were deprived of essential information on the terms of the trust agreement, the degree of GA's continuing control of SFC, the costs of decommissioning, the adequacy of resources to pay for cleanup, and GA's decommissioning costs for its facilities in San Diego. *Id.* at 17-18.

Intervenors argue the agreement fails to provide assurance of adequate funds to complete decommissioning as contemplated in the Staff 1993 order. Instead of responsibility for any funding shortfall by SFC demanded by the October 1993 Order, GA can commit a single payment possibly yielding, after taxes, to as little as $3.9 million for the cleanup effort. This settlement should be rejected, say Intervenors, because the potential cost of cleanup might be $150 million more than cited in the order and the Staff has no independent knowledge of what the actual costs might be. The agreement is also assertedly defective because it says nothing about the expected contribution of SFC to the cleanup effort; it does not provide that GA supply funding in a timely manner in relation to the need for funds; and it has an undisclosed impact on an EPA-mandated cleanup effort since it provides for the retirement of two large loans being used to finance the EPA cleanup. It is contended that GA and the Staff must explain the impact of this measure on the EPA cleanup before a public interest finding can be made. *Id.* at 18-22.

Intervenors argue the settlement agreement fails to disclose the terms of the trust agreement; to resolve two tax liability issues related to obtaining an IRS opinion on whether the $9 million trust fund is taxable; and to provide support for GA's claim that it would suffer financial ruin if a large adverse judgment were to be entered against it. *Id.* at 22-25.

Intervenors urge the Board not to approve the settlement agreement without first requiring full disclosure of the costs of cleanup of GA's San Diego facilities. They reject the Staff and GA assertion that these costs are outside the scope of this proceeding. According to Intervenors, if GA's liability for the San Diego facilities has had any impact on the amount of settlement for the SFC site, the accuracy and reliability of its assertions is relevant in this case and must be subject to evaluation by the Board and parties. *Id.* at 25-26.

The Intervenors urge the Board to reject the Staff's and GA's arguments concerning litigation risk because the prospects of winning any case are never certain. In this case, they argue, the Staff position was assertedly a strong one and it should not have been given up in exchange for an amount of money small in comparison to the cost of cleanup. It alleges the Staff did not secure a fair bargain for Intervenors or the public and any litigation expenses are minor in comparison with the cost of cleanup for the SFC site. *Id.* at 27-28.

Intervenors renew their questions over the SFC agreement concerning whether SFC will be required to pay a $10.6 million debt; concerns over the degree of control that GA exercises over ConverDyn; uncertainty whether GA officials could later be appointed to the SFC Board; and concerns whether GA could exercise control over SFC through its subsidiaries Sequoyah Holding Corporation and Sequoyah Fuels International. These issues, it contends, must be resolved before the settlement agreement can be approved. *Id.* at 29.

Intervenors state the settlement providing for the resignation of two GA officers from SFC's Board of Directors runs counter to SFC's license which is based on expectation of close GA involvement with management of its safety operations. It also precludes the Staff's claim that GA is a *de facto* licensee which may preclude future enforcement action against GA for matters such as quality assurance. This goes beyond the scope of the 1993 order and effectively amends SFC's license without notice in violation of the Atomic Energy Act and NRC regulations. *Id.* at 30.

The Staff and GA propound different responses to the issues raised by the objecting parties. They concur that the agreement represents a fair and reasonable compromise of their positions. The possibilities of not prevailing in protracted litigation, in their view, with time, expense, and other financial considerations involved, attest that the agreement is in the public interest.

The Staff Reply

The Staff counters Intervenors' allegations by arguing that reasonable people can differ on the terms of an agreement, but the Board is required, under the standards set forth in 10 C.F.R. § 2.203, to accord due weight to the Staff's position; that it has available information concerning GA's financial position which, under the Commission's regulations (10 C.F.R. § 2.790(a)(4)) it is unable to disclose publicly; that even the lower amount of funds from GA — $5.4 million — justifies the agreement and the $72 million projected from ConverDyn to SFC should not be ignored in reviewing the funds pledged by GA.[6] Staff Reply at 4-15.

The Staff contends that disclosure of GA's financial information could threaten the company's competitive position and the agreement's funding. It is claimed that irrespective of the final cost of decommissioning, the agreement was in the public's interest and its provisions preclude GA from manipulating SFC's present or future assets and revenues. On the State's contention that continued litigation of GA's liability was "of significant interest," the Staff asserts that GA contributions were more in the public's interest than a lengthy and expensive adjudication. To the State's claim that the SFC agreement should be rescinded if the agreement under consideration is approved, the Staff avers the SFC agreement is beyond the jurisdiction of the Board. Staff Reply at 15-24.

[6] The Staff correctly characterizes as moot the Intervenors' argument against a provision authorizing GA to review NRC press releases on the agreement.

General Atomics' Response

In countering the Intervenors' and State's contentions together, GA argues that settlement of contested proceedings is encouraged by the Commission and the only consideration is whether the agreement is fair and reasonable. Citing NRC case law, decided prior to the regulation on settlement of enforcement orders, GA argues that a settlement agreement must be approved unless "patently arbitrary or contrary to law."[7] Citing a number of considerations involved in the settlement discussions as constituting the agreement as "fair and reasonable," GA argues that the opposing parties seek discovery on matters beyond the Board's jurisdiction, such as decommissioning costs at GA's NRC-licensed facilities in San Diego. GA concludes no evidence had been submitted to rebut a "heavy presumption" that the agreement was fair and reasonable. GA Response at 7-31.

DECISION

The substance of the several positions iterated by Intervenors and the State is that the agreement negotiated by the Staff and GA is not in the public interest and a variety of matters related to it require adjudication prior to its approval by the Board. These include, *inter alia,* current cost estimates of SFC's decommissioning, GA's financial condition, information on GA's licensed San Diego facilities and the ConverDyn arrangement. The Board's function in reviewing settlement agreements, as delineated in 10 C.F.R. § 2.203, calls for settlements to be approved by the Board and an adjudication of any issues that may be required in the public interest to dispose of the proceeding. The Board is enjoined therein to provide "due weight to the position of the Staff." The settlement of contested proceedings has long been encouraged by the Commission. *See* 10 C.F.R. §§ 2.759, 2.1241. And guidance on the subject encourages licensing boards to hold settlement conferences. *Statement of Policy on Conduct of Licensing Proceedings,* CLI-81-8, 13 NRC 452, 456 (1981).

A review of NRC cases concerned with settlements discloses limited information on standards utilized in support of Board approval of such agreements. These decisions are accompanied generally by succinct references that on the basis of Board review, the agreements involved were found in the public interest. *See, e.g., Radiation Oncology Center at Marlton* (Marlton, New Jersey), LBP-96-4, 43 NRC 101, 102 (1996); *North American Inspection, Inc.* (P.O. Box 88, Laurys Station, Pennsylvania 18059), ALJ-86-2, 23 NRC 459, 460 (1986). It would appear that, in enforcement cases, the weight provided the Staff's position has uniformly resulted, without more, in Board acceptance of the agreements.

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[7] *In the Matter of New York Shipbuilding Corp.,* 1 AEC 842, 844 (1961).

The rationale for such judgments may be grounded on the merited understanding that, in the end, the Staff is responsible for maintaining protection for the health and safety of the public and in the absence of evidence substantiating challenges to the exercise of that responsibility, the Staff's position should be upheld. And the lack of such evidence appears to be the case here in considering approval of the agreement before us.

Nothing the Intervenors or State have propounded evidences a conclusion that, due to the terms of the agreement before us, the public's health and safety will not be protected. On the fundamental question of whether adequate funds will be available for decommissioning SFC's facility, GA in its response, submitted a Declaration of its Vice President–Administration conveying information that SFC is receiving revenue from ConverDyn and the latter firm is performing as expected under its agreement with SFC. Declaration, ¶¶9-11. The Staff, which the Declaration alleges has been receiving financial spreadsheets of ConverDyn's substantial performance to date, comments that the total projected revenues to SFC — approximately $72 million — should not be overlooked in the consideration of approving the agreement. And, the agreement itself cites that "based upon SFC's actual experience to date, General Atomics and SFC believe that SFC's net assets and revenues, as defined in the Settlement Agreement between the NRC Staff and SFC, will provide adequate capital resources to allow SFC to conduct its ongoing standby operations and to complete environmental remediation and decommissioning."

It is the opinion here that, in addition to the foregoing assurances concerning the likely availability of decommissioning revenues, an approval of settlement of the enforcement order should also receive our affirmation after weighing the consideration given to other factors in the public interest.[8] These factors concern the intensity of negotiations, the complexity of questions of law and fact in the proceeding placing its ultimate outcome in doubt, the value of an immediate recovery compared to the mere possibility of prevailing after protracted and expensive litigation, and the judgment of the parties concerning the fairness and the reasonableness of the settlement.[9]

Despite the concerns expressed by the Intervenors and State, the issue is not whether the matter before us presents the best settlement that could have been obtained. Our obligation instead is merely to determine whether the agreement is "within the reaches of the public interest." *United States v. Gillette Co.*, 406 F. Supp. 713, 716 (1975). Here, the Staff and GA negotiated the terms of this agreement over a period of 10 months, a fact that supports a recognition by the parties of the seriousness of resolving the litigative differences involved; both

[8] *See Sequoyah Fuels Corp.* (Gore, Oklahoma Site), CLI-94-12, 40 NRC 64, 71 (1994).

[9] A leading case in settlement proceedings where similar factors were delineated. *Gottlieb v. Wiles*, 11 F.3d 1004, 1014 (1993).

signatories, in the agreement and responding briefs, cite the complexity of the legal and factual issues between them and the heavy financial and manpower resources required if the proceeding continues; in comparing the value of GA payments under the agreement against the possibilities of ultimately prevailing in the litigation, the Staff recognizes the risk of not receiving any funds if either its unique legal theory of holding GA liable as a *de facto licensee* does not prevail or the Company's finances are depleted as a competitive business entity; and finally, both parties, in consideration of the total circumstances of the controversy, assent to the fairness and reasonableness of the agreement. From the terms of the agreement and the briefs submitted by the signing parties, it is clear that the interests of the public have not been neglected in the document before us. It requires our approval.

There is no necessity for this opinion to discuss extensively the arguments raised by the opposing parties: no basis exists for concluding that NRC's regulatory requirements for funding decommissioning will not be met; the trust fund called for in the agreement has been instituted and its provisions made public; any judicial review here of the Staff-SFC settlement agreement is now beyond this Board's jurisdiction; consideration of GA's license responsibilities at its facilities in San Diego, California, or anywhere else, does not bring the Staff review of such matters within our jurisdictional boundary; this Board has no jurisdiction to consider impacts that the agreement's provisions might have in regard to cleanup requirements of the Environmental Protection Agency; both the Intervenors and State have had their interests acknowledged by being allowed to participate in this proceeding and to express their concerns; and finally, although the current financial estimates of SFC's decommissioning costs, if different from those previously submitted to the Staff, may have some bearing on the Staff's determinations leading up to the agreement, they have no bearing on the Board's responsibility in approving the agreement itself. On its part, the Staff has acknowledged that these estimates may have increased and, nevertheless, the lowest figure mentioned in the agreement — $5.4 million — would still justify its acceptance.[10]

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[10]The dissent to this opinion by our colleague, like his similar dissent to the Board's approval of the Staff/SFC settlement agreement, requires additional information from the Staff to secure his concurrence herein. The majority declines to follow that course because it projects the Board's role as one of overseeing the Staff's function of assuring decommissioning funding of the Sequoyah facility. The information requested might be necessary at a trial on the merits. Here, the inquiry is inappropriate because it would, in our view, make us a participant in settlement negotiations.

The majority opinion recognizes that our role at a settlement stage is limited to a review that consideration has been provided to the public interest. The Board's approval of the agreement is not based on the merits of the 1993 Order but the merits of the agreement itself. The majority declines to intrude into the merits of the issues of the case because that would ignore the fact that a settlement is a compromise of the issues framed by the Order; it would invade an area of Staff responsibility; and would not give appropriate weight to the Staff's position as required by 10 C.F.R. § 2.203. It is apparent that the Staff is now willing to forego claims for total funding

(Continued)

In light of the foregoing, the settlement agreement is approved.
It is so ORDERED.

THE ATOMIC SAFETY AND
LICENSING BOARD

James P. Gleason, Chairman
ADMINISTRATIVE JUDGE

Jerry R. Kline
ADMINISTRATIVE JUDGE

Rockville, Maryland
November 5, 1996

Dissenting Statement by Bollwerk, J.

Previously, when the Sequoyah Fuels Corporation (SFC) and the NRC Staff sought Licensing Board approval of their proposed agreement to settle this litigation as between them, I declined to consent and filed a separate statement. In that statement, I delineated several matters about which I needed additional information before I could make the requisite "public interest" finding pursuant to 10 C.F.R. § 2.203. *See* LBP-95-18, 42 NRC 150, 156-59 (1995) (separate statement of Bollwerk, J.), *petition for review granted,* CLI-96-3, 43 NRC 16 (1996). Now, more than a year later, I find myself in the same position relative to the pending settlement agreement between General Atomics (GA) and the Staff. Below, I outline my central concerns about the GA/Staff agreement and the questions I would seek to have answered.

The Staff's October 1993 enforcement order was rooted in three premises:

(1) The existing sources of revenue for cleanup of the SFC Gore, Oklahoma facility consist of (a) the ConverDyn agreement that, while estimated to result in revenues totaling no more than $72 million, was "based on inherently speculative assumptions" such that it did not provide the requisite "reasonable assurance" of adequate decommissioning funding under 10 C.F.R. § 40.36; and (b) $17 million from other sources.

(2) Based on SFC estimates of the cost of its preferred decommissioning alternative (which had not been approved by the Staff), decommissioning costs would run at least $86 million,

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assurance based on its theory that GA is a *de facto licensee* in exchange for limited but guaranteed contributions from GA. No conclusions can or should be reached from that decision that decommissioning of SFC's facility has been abandoned or threatened. That matter is not before us and to suggest it might be is to provide little, if any, weight to the Staff's position in the settlement as we are directed by NRC's regulations to do.

> but there was "uncertainty" over these preliminary projected costs such that the estimated total of $89 million from the ConverDyn agreement and other sources was "unlikely to be sufficient" to cover the costs of decommissioning the SFC facility if the NRC imposed additional requirements.
>
> (3) In light of SFC's apparent inability to cover the total costs of decommissioning, to obtain the necessary "reasonable assurance" under section 40.36, it was necessary to make GA — as the parent corporation exercising "de facto" control over SFC's day-to-day business — liable for any shortfall in decommissioning funds.

58 Fed. Reg. 55,087, 55,089, 55,091-92 (1993). In toto, the order was an apparent attempt by the Staff to ensure the "public interest" was protected by providing the requisite reasonable assurance that the *total* decommissioning costs for SFC's Gore, Oklahoma facility would be covered by those entities purported to have regulatory responsibility for such costs.

As I understand the terms of the present settlement, the Staff now has forsworn its quest to make GA the general (and seemingly unlimited) guarantor of decommissioning funding for the SFC facility and has instead chosen to settle for a specific (but limited) contribution. The apparent theory behind this decision to compromise is that, with all the uncertainties, difficulties, and expense involved in this litigation and the financial problems of GA (about which the Board has no direct knowledge), the settlement the Staff and GA have arrived at is the "best bargain in the public interest." NRC Staff's Reply to Intervenors' Opposition to Joint Motion for Approval of Settlement Agreement Between NRC Staff and General Atomics and to the State of Oklahoma's Response to NRC Staff's and General Atomics' Joint Motion for Approval of Settlement Agreement (Oct. 11, 1996) at 9 [hereinafter Staff Reply].

Before I can accede to this formulation of what serves the "public interest," at a minimum I need a fuller understanding about the implications of the settlement agreement's terms on the "public interest" as the Staff framed it in its original enforcement order by its reliance on the need for "reasonable assurance" under 10 C.F.R. § 40.36. *See* LBP-95-18, 42 NRC at 159 n.6. I would, therefore, pose the following questions to the Staff as the proponent of that order and, under 10 C.F.R. § 2.203, the party whose "position" must be given "due weight":

1. As was noted in the Staff's 1993 enforcement order, it was estimated by SFC that decommissioning costs for its Gore, Oklahoma facility would total at least $86 million. What is the Staff's present best estimate of the total costs of decommissioning the facility?

2. As was also noted in the Staff's 1993 enforcement order, it was estimated the ConverDyn agreement would result in revenues of no more than $72 million available to pay decommissioning costs and there would be $17 million from other sources to pay such costs. In light of developments since 1993, what is the Staff's present best estimate of (a) the maximum revenue that will be generated for facility decommissioning work under the ConverDyn agreement; and (b) the amount that

> would be available for such work from other sources (not including funds generated by the proposed GA/Staff settlement agreement)?

With the Staff's responses to these questions,[1] and any additional relevant information that GA or the Intervenors might provide when given a chance to comment on the Staff's answers, I believe the Board would have a much clearer understanding of whether, and to what degree, the proposed settlement agreement impacts on the "public interest" in seeing that there is "reasonable assurance" of adequate funding for facility decommissioning.[2]

In addition, borrowing from the medical profession and its well-established principle, as embodied in the Hippocratic Oath, that one should strive to "do no harm," to ensure the GA/Staff agreement contains nothing that would have an adverse impact on the "public interest," I would make a third inquiry:

> 3. If the total funds available for decommissioning under the SFC and GA settlement agreements with the Staff ultimately are insufficient to cover the total costs of decommissioning the Gore facility (a) what, if any, additional cleanup mechanisms are available to complete decommissioning (e.g., Superfund); and (b) if there are additional cleanup mechanisms, would anything in the provisions of the GA/Staff settlement agreement have an adverse impact on GA's liability, if any, under those cleanup mechanisms?

Finally, so that the record before the Board is clear, I would seek information on two other, albeit less central points:

> 4. Under paragraph two of the settlement agreement, GA was to request an IRS opinion regarding the tax status of the settlement trust fund "immediately" following execution of the agreement. *See* NRC Staff's and [GA's] Joint Motion for Approval of Settlement Agreement (July 11, 1996), attach. 1, at 6-7. To the best of the Staff's knowledge, what is the status of the GA request for an IRS determination and when is an IRS determination expected?
>
> 5. Under paragraph eight of the settlement agreement, if amounts borrowed by SFC from GA pursuant to certain "Lines of Credit" are not repaid by December 31, 1998, then GA is permitted to delay for 1 year payment to the trust fund of one-

[1] Although, as the majority observes, *see* 44 NRC at 257-58, in responding to the concerns of Intervenors Native Americans for a Clean Environment and the Cherokee Nation the Staff commented that the earlier projection of $72 million in revenues from the ConverDyn agreement "should not be ignored," Staff Reply at 10, with the only representations about the validity of this projected revenue figure in the settlement agreement attributed to GA and SFC, *see* NRC Staff's and [GA's] Joint Motion for Approval of Settlement Agreement (July 11, 1996) attach. 1, at 4, I would seek the Staff's explicit views about the soundness of that estimate.

[2] The majority suggests that because the Board's role "is limited to a review that consideration has been provided to the public interest," seeking this information would result in an improper intrusion into the "merits" of the staff's enforcement order. 44 NRC at 258-59 n.10. I find my proposed inquiry wholly consistent with the Board's authority under 10 C.F.R. § 2.203 to make its *own* judgment, albeit while "according due weight to the position of the staff," about whether the agreement is in the public interest such that no further adjudication of the issues in the proceeding is warranted.

half of the amounts otherwise due no later than December 31, 1998. *See id.* attach 1, at 8. Because the "Lines of Credit" in question apparently relate to a separate Environmental Protection Agency administrative order, *see id.* attach. 1, at 3-4, why does their repayment have an impact on payments due under this settlement agreement between GA and the Staff?

With this information, I might well be in a substantially better position to determine, relative to the GA/Staff settlement accord, where the "public interest" lies. Without it, I am not prepared to approve their agreement.

ATTACHMENT

SETTLEMENT AGREEMENT

THIS AGREEMENT is made by and between the Staff of the United States Nuclear Regulatory Commission ("NRC Staff") and General Atomics (the "Company"), to wit:

WHEREAS, on October 15, 1993 the NRC Staff issued an Order to Sequoyah Fuels Corporation ("SFC") (58 Fed. Reg. 55087, October 15, 1993) (the "October 15 Order"), relating to the funding of the site decontamination and decommissioning of the facilities located in Gore, Oklahoma that are licensed under NRC License No. SUB-1010, Docket No. 40-8027 (the "SFC Facility"); and

WHEREAS, the NRC Staff also issued the October 15 Order against SFC's third-tier parent company, General Atomics, alleging *inter alia,* that General Atomics and SFC were jointly and severally responsible for: (1) Providing funding to continue remediation of existing contamination at the SFC Facility site, regardless of whether the facility continued to operate or not; (2) Providing financial assurance for decommissioning in accordance with the requirements of 10 C.F.R. § 40.36; and (3) Providing an updated detailed cost estimate for decommissioning and a plan for assuring the availability of adequate funds for completion of decommissioning, in accordance with the requirements of 10 C.F.R. 40.42(c)(2)(iii)(D); and

WHEREAS, the October 15 Order does not allege, and the NRC Staff has not asserted, either that General Atomics caused any contamination which may exist at the SFC Facility, or that General Atomics has otherwise engaged in any form of activity that is wrongful or dangerous to the public health and safety; and

WHEREAS, on November 2, 1993, General Atomics and SFC filed separate answers to the October 15 Order requesting that it be rescinded, or in the alternative, that a hearing be held on it; and

WHEREAS, an administrative enforcement proceeding is now being conducted before an Atomic Safety and Licensing Board (the "Board") in Docket No. 40-8027-EA ("Administrative Proceeding"), and General Atomics and the NRC Staff are parties in that proceeding; and

WHEREAS, General Atomics has consistently and specifically denied that the Nuclear Regulatory Commission ("Commission") has jurisdiction over it with regard to the matters set forth in the October 15, 1993 Order; and

WHEREAS, General Atomics commenced a civil action in the U.S. District court for the Southern District of California (the "California Civil Action") challenging the Commission's exercise of jurisdiction over it; and

WHEREAS, the California Civil Action was appealed to the United States Court of Appeals for the Ninth Circuit and that Court has ruled that the action is premature because of the absence of a Final Order by the NRC in the Administrative Proceeding; and

WHEREAS, on July 26, 1995, the Commission published a Final Rule, "Clarification of Decommissioning Funding Requirements" (60 Fed. Reg. 38,235, July 26, 1995) that if applied to SFC, would require that on the effective date of the rule, November 24, 1995, SFC provide financial assurance of decommissioning funding for the Sequoyah Facility using one of the methods provided for in 10 C.F.R. § 40.36(e); and

WHEREAS, General Atomics and SFC commenced separate civil actions in the United States Court of Appeals for the Ninth and Tenth Circuits challenging the lawfulness of the new final rule, and the civil actions are now consolidated in the Ninth Circuit; and

WHEREAS, on August 24, 1995, SFC and the NRC Staff filed a joint motion with the Board seeking the Board's approval of a Settlement Agreement by and between SFC and the NRC Staff which would, subject to the terms of that agreement, obligate SFC to devote all of its net assets and net revenues to the completion of the decommissioning of the SFC Facility in accordance with the requirements of the Commission, the U.S. Environmental Protection Agency ("EPA"), and any other state or federal agency with jurisdiction, until the NRC Staff determines that such decommissioning has been satisfactorily completed; and

WHEREAS, by its Memorandum and Order of October 26, 1995, the Board formally approved the proposed Settlement Agreement between the NRC Staff and SFC; and

WHEREAS, in connection with an August 3, 1993 Administrative Order on Consent ("RCRA Consent Order") issued by the EPA (U.S. EPA Docket No. VI-005-(h) 93-H) and agreed to by SFC for the environmental remediation of the SFC Facility, General Atomics voluntarily agreed to continue to make funds available to SFC as loans under certain Revolving Promissory Notes (in the amounts of $2,500,000.00 and $4,500,000.00 respectively) and for the purpose of supporting SFC in its efforts to provide financial assurance regarding the availability of funds to implement the RCRA Consent Order; and

WHEREAS, based upon SFC's actual experience to date, General Atomics and SFC believe that SFC's net assets and net revenues, as defined in the Settlement Agreement between the NRC Staff and SFC, will provide adequate capital resources to allow SFC to conduct its ongoing standby operations and to complete environmental remediation and decommissioning; and

WHEREAS, the NRC Staff and General Atomics understand and acknowledge that many of the issues raised by the October 15 Order are complex and likely to require the continued expenditure of significant manpower and financial resources by each party if they are to be resolved through litigation; and

WHEREAS, the NRC Staff and General Atomics understand and acknowledge that it is in the public interest to avoid the dissipation of their financial resources and manpower in litigation, particularly since it is in the public interest that General Atomics retain the financial capability to meet certain other decommissioning obligations which are not disputed and which are not within the scope of the Administrative Proceeding or the jurisdiction of the Board; and

WHEREAS, General Atomics believes that the mere existence of the October 15 Order has adversely and significantly affected the credit rating of General Atomics and its ability to engage in its regular business activities, irrespective of the lawfulness or the merits of the Order; and

WHEREAS, the business of General Atomics has been dependent upon government contracts, especially U.S. Department of Energy contracts for the development of the Company's Gas Turbine-Modular Helium Reactor (GT-MHR) and for its nuclear fusion research program; and

WHEREAS, funding for continued development of the GT-MHR has now been terminated by Congress; and

WHEREAS, General Atomics believes that funding for the Company's fusion research program in FY 1996 was reduced by approximately thirty percent (30%) by Congress from the FY 1995 level; and

WHEREAS, General Atomics asserts that it has sustained significant financial impairment since the NRC Staff issued the October 15 Order; and

WHEREAS, the NRC Staff and General Atomics have engaged in negotiations seeking an amicable resolution of the issues raised by the October 15 Order because they recognize that the public interest will be served and that certain advantages and benefits may be obtained by each of them through settlement and compromise of the controverted matters now pending.

NOW, THEREFORE, in consideration of the mutual promises made herein, the NRC Staff and General Atomics agree as follows:

1. Within ninety (90) days of the execution of this Settlement Agreement, General Atomics shall establish a trust fund ("Fund") for the benefit of the NRC,

into which General Atomics shall deposit a total of $9,000,000.00 in accordance with the schedule set forth in Annex "A," attached hereto. The governing trust fund agreement provided by General Atomics and approved by the NRC shall be structured, to the extent applicable, consistent with the model trust fund agreement set forth in NRC Regulatory Guide 3.66. The trust fund agreement shall provide that the trustee ("Trustee") shall make payments from the Fund as the NRC shall direct or in accordance with procedures approved by the NRC. *Provided, however,* that until such time as the Internal Revenue Service renders an opinion which is unqualified in any material respect (1) that all of the payments to the Fund by General Atomics are deductible when made for Federal income tax purposes, whether the Fund is deemed a "qualified settlement fund" as that term is used in the Internal Revenue Code, or otherwise constitutes a fund regarding which such payments by General Atomics are deductible when made under the Internal Revenue Code, and (2) that payments into the Fund are not taxable to SFC or General Atomics until such amounts are actually paid from the Fund to SFC if in fact so paid, General Atomics shall not be required to deposit into the Fund in excess of $5,400,000.00, and shall make deposits totalling such amount in accordance with the schedule set forth in Annex "B," attached hereto.

2. General Atomics shall, immediately following the execution of this Settlement Agreement, seek the above-described opinion from the Internal Revenue Service regarding the trust fund established pursuant to Paragraph 1 of this Settlement Agreement. At such time as General Atomics receives the opinion, it shall promptly transmit a copy of it to NRC.

3. This Settlement Agreement constitutes full settlement of the NRC Staff's claims against General Atomics with respect to the October 15 order.

4. General Atomics shall have no control over the management of either the Fund or the funds deposited therein. Any principal and interest of the trust will be distributed pursuant to the terms of the trust instrument as established by General Atomics and approved by the NRC Staff.

5. General Atomics further agrees that the two officers of General Atomics who currently serve on SFC's Board of Directors will resign from that board no later than June 30, 1997.

6. General Atomics further agrees that subsequent to the execution of the Settlement Agreement and no later than ten (10) days after the establishment of the Fund, it will pay into the Fund the sum of $600,000.00. Except as the terms of paragraph 1 above expressly provide to the contrary, no further payments shall be required of General Atomics until there is final agency action regarding the Settlement Agreement, unless there is no final agency action by December 15, 1997, in which circumstances General Atomics will pay an additonal $1,200,000.00 into the Fund within sixty (60) days following December 15, 1997.

7. General Atomics further agrees that in the event that the Settlement Agreement is finally approved by the Commission or an order approving it becomes final agency action and all appeals of such action have been exhausted without success by December 31, 1996, it will make payments into the Fund in the amounts and on the dates specified in either Annex "A" or Annex "B," whichever annex is most appropriate under the provisions of paragraph 1 above. If the Settlement Agreement is finally approved by the Commission, or an order approving it becomes final agency action, and all appeals of such action are exhausted without success after December 31, 1996, but before December 31, 1997, General Atomics agrees to make sufficient payments to the Fund to ensure that the payment schedule set forth in either Annex "A" or Annex "B," whichever is most appropriate under the provisions of paragraph 1 above, is made current no later than 120 days after final agency action.

8. It is the intent of the parties to this Settlement Agreement that as they are paid down by SFC, the Revolving Promissory Notes from SFC to General Atomics for amounts borrowed pursuant to the Lines of Credit extended by General Atomics, will be reduced to the sums still owed and extinguished as to excess borrowing capacity as the sums are paid by SFC to General Atomics. It is the further intent of the parties that the Promissory Notes will be fully extinguished no later than December 31, 1998. Notwithstanding anything in paragraphs 1-7 above or Annexes "A" or "B" to the contrary, in the event that all amounts borrowed by SFC pursuant to the Lines of Credit have not been repaid by December 31, 1998, General Atomics shall have the option of delaying until December 31, 1999, the payment to the Fund of one-half of the amounts otherwise due no later than December 31, 1998.

9. The parties hereto agree that in the event that (a) the Commission does not approve the Settlement Agreement, or (b) the Commission's final approval of the Settlement Agreement is reversed or otherwise set aside by a court of law, all funds which have been paid under this Settlement Agreement, together with all earnings thereon, shall be returned to General Atomics by the Trustee no later than sixty (60) days after a Commission or judicial decision disapproving the Settlement Agreement. All other matters shall likewise return to the status quo which existed prior to the execution of the Settlement Agreement.

10. General Atomics further agrees not to take any action that an independent observer would reasonably conclude will interfere with the ability of SFC to carry out the NRC-SFC Settlement Agreement which was approved by the Board on October 26, 1995.

11. General Atomics further agrees to cooperate fully with the NRC Staff in explaining the terms of this Settlement Agreement to the public, the Board, the Commission, and/or any court of competent jurisdiction. In this context, and because of the potential effect upon General Atomics' competitive position within the marketplace, the NRC Staff agrees that the office of Public Affairs of

the NRC ("OPA") has represented to the NRC Staff that before OPA issues any news release describing the terms of this settlement, it will confer with General Atomics to confirm the accuracy of any statements of fact which it proposes to include in the news release.

12. The NRC Staff and General Atomics agree that the obligations assumed by the Company in this Settlement Agreement represent a good faith, voluntary and major effort to settle the matters relating to the October 15 Order. As a consequence of this effort, the NRC Staff agrees to permanently rescind the October 15 Order insofar as it applies to General Atomics and accepts the terms of this Settlement Agreement in lieu of these provisions of the October 15 Order that are directed to General Atomics. Subject to the provisions of Paragraphs 13 and 14 below, the NRC Staff further agrees to forbear from taking any enforcement or other action against General Atomics or its current, former, or future officers, directors or employees (relating to their actions in their official capacities), (a) based upon any alleged requirement to provide funds for the decommissioning of the SFC Facility or to provide financial assurance for the decommissioning of the SFC Facility, whether such requirement arises under any current NRC regulations or under any future regulation that might alter, redefine or clarify the currently applicable requirements, or (b) based upon the facts alleged in the October 15 Order and/or those reasonably known by the NRC Staff that are related to the subject matter of the October 15 Order.

13. The NRC Staff further agrees that it will not assert against General Atomics in the future, and "*de facto* licensee" or other claims which are similar to those asserted in the October 15 Order, and which are based upon (a) the performance by General Atomics' personnel of any of the audit or other oversight responsibilities required by the license issued to SFC by the Commission, (b) the reasonable exercise by General Atomics' officers and management personnel of the business judgment referred to in paragraph 15 below, and (c) the nature of the degree of ownership by or of General Atomics of any of its parent, subsidiary or affiliated companies.

14. Notwithstanding any provisions in this Settlement Agreement to the contrary, nothing herein shall limit the NRC Staff's ability to take appropriate action to enforce General Atomics' compliance with this Settlement Agreement, or to take appropriate enforcement action based upon (a) future conduct by General Atomics which is materially different from that described in the October 15 Order, in paragraph 13 above, or which is reasonably known by the NRC Staff on the date this Settlement Agreement is entered into, (b) material information that is not currently available to or reasonably known by the NRC Staff, or (c) evidence that any representation in this Settlement Agreement is incomplete or inaccurate in a material respect. The NRC Staff and General Atomics acknowledge that the terms and provisions of this Settlement Agreement, once approved by the Board, shall be incorporated by reference into an order issued

by the Board, as the term "order" is used in subsections (b), (i) and (o) of Section 161 of the Atomic Energy Act of 1954, as amended (the "Act"), 42 U.S.C. § 2201, and shall be subject to enforcement pursuant to the Commission's regulations and Chapter 18 of the Act, 42 U.S.C. § 2271 *et seq.*

15. Nothing in this Settlement Agreement shall limit the right and obligation of the officers and management personnel of General Atomics to fully exercise their best judgment in the management of the Company.

16. The NRC Staff and General Atomics understand and acknowledge that this Settlement Agreement is the result of a compromise and shall not for any purpose be construed as an admission of the facts alleged or conclusions of law drawn in the October 15 Order, as an admission of the alleged joint and several responsibilities of General Atomics included in Section VII.A and other sections of the October 15 Order, or as an admission by General Atomics of any violation of 10 C.F.R. § 40.36, 10 C.F.R. § 40.42, or any statute, regulation, license condition, or other regulatory requirement.

17. The NRC Staff and General Atomics further agree that no inference adverse to either party shall be drawn based upon the parties having entered into this Settlement Agreement.

18. The NRC Staff and General Atomics further agree to file a joint motion requesting that the Board approve this Settlement Agreement, pursuant to the Commission's regulations in 10 C.F.R. § 2.203. Upon approval of this Settlement Agreement by the Board, without any substantive modification by the Board, the NRC Staff and General Atomics agree that they will not appeal the Board's approval or otherwise seek judicial review of such approval. If this Settlement Agreement is not approved by the Board, or if this Settlement Agreement is approved by the Board but is modified by the Board in a manner which either party believes to be a substantive modification, or any body or court to which the Board's approval is appealed reverses such approval or affirms the approval but modifies the Settlement Agreement is a manner which either party believes to be substantive, either the NRC Staff or General Atomics may void the Settlement Agreement by giving written notice to the other party within ninety (90) days of such action by the Board, body or court, unless such 90-day period is extended by written agreement of both parties. The NRC Staff and General Atomics agree that under such circumstances and upon request they will negotiate in good faith to resolve differences which are the result of such substantive modification.

19. This Settlement Agreement shall become effective upon execution, and is revocable only upon a failure of the Board to approve it or upon the action of the Commission, another agency of the U.S. Government, or any other body or court of law which has jurisdiction to review and approve or disapprove the Settlement Agreement and which disapproves it or any substantive part of it.

IN WITNESS WHEREOF, the NRC Staff and General Atomics have caused this Settlement Agreement to be executed by their duly authorized representatives on this 10th day of July, 1996.

FOR THE NUCLEAR REGULATORY COMMISSION:

Hugh L. Thompson, Jr.
Deputy Executive Director for Nuclear Materials Safety, Safeguards and Operations Support

FOR GENERAL ATOMICS:

John E. Jones
Senior Vice President

ANNEX "A"

DATE	AMOUNT TO BE PAID
No later than ten (10) days after the establishment of the Trust Fund	$1,000,000.00
December 31, 1996	$2,000,000.00
December 31, 1997	$2,000,000.00
December 31, 1998	$2,000,000.00
December 31, 1999	$ 200,000.00
December 31, 2000	$ 200,000.00
December 31, 2001	$ 200,000.00
December 31, 2002	$ 200,000.00
December 31, 2003	$ 200,000.00
December 31, 2004	$ 200,000.00
December 31, 2005	$ 200,000.00
December 31, 2006	$ 200,000.00
December 31, 2007	$ 200,000.00
December 31, 2008	$ 200,000.00
TOTAL	$9,000,000.00

ANNEX "B"

DATE	AMOUNT TO BE PAID
No later than ten (10) days after the establishment of the Trust Fund	$ 600,000.00
December 31, 1996	$1,200,000.00
December 31, 1997	$1,200,000.00
December 31, 1998	$1,200,000.00
December 31, 1999	$ 120,000.00
December 31, 2000	$ 120,000.00
December 31, 2001	$ 120,000.00
December 31, 2002	$ 120,000.00
December 31, 2003	$ 120,000.00
December 31, 2004	$ 120,000.00
December 31, 2005	$ 120,000.00
December 31, 2006	$ 120,000.00
December 31, 2007	$ 120,000.00
December 31, 2008	$ 120,000.00
TOTAL	$5,400,000.00

Directors' Decisions Under 10 CFR 2.206

DIRECTORS' DECISIONS

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Frank J. Miraglia, Jr., Acting Director

In the Matter of

ALL POWER REACTOR LICENSEES **November 6, 1996**

The Acting Director of the Office of Nuclear Reactor Regulation denies in part and grants in part a petition dated April 13, 1994, submitted to the Nuclear Regulatory Commission (NRC) by Mr. Paul M. Blanch (Petitioner) requesting that the NRC take immediate action with regard to all power reactor licensees concerning the potential failure of the fuel in spent fuel pools for all reactors in the United States.

Petitioner requested that the NRC immediately issue an information notice or other appropriate notification forwarding all information in its possession to all power reactor licensees regarding the potential failure of fuel in spent fuel pools and reminding licensees of their responsibilities to perform timely operability determinations. This request was granted in part based on issuance by the NRC of generic communications to licensees on failure of spent fuel.

Petitioner also requested that the NRC direct each licensee to immediately perform an evaluation of the potential failure of spent fuel in spent fuel pools to determine compliance with its current licensing basis. This request was granted in part based on evaluations performed by the NRC Staff of both the design and operational aspects of spent fuel pool storage issues for all operating reactors.

Finally, Petitioner seemed to suggest that the exercise of enforcement discretion by issuance of a Notice of Enforcement Discretion (NOED) may be appropriate concerning spent fuel pool issues raised in the petition. Based upon the review of the information provided in the petition, the NRC Staff has not identified any circumstance warranting the issuance of a NOED.

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

I. INTRODUCTION

By a petition submitted pursuant to 10 C.F.R. § 2.206 on April 13, 1994, Mr. Paul M. Blanch (Petitioner) requested that the U.S. Nuclear Regulatory Commission (NRC) take immediate action with regard to all power reactor licensees, concerning the potential failure of the fuel in the spent fuel pools for all reactors in the United States. Specifically, the Petitioner requested that the NRC: (1) immediately issue an information notice or other appropriate notification forwarding all information in its possession to all power reactor licensees regarding the potential failure of fuel in spent fuel pools, and reminding licensees of their responsibilities to perform timely operability determinations in accordance with their technical specifications and NRC Generic Letter 91-18; (2) direct each licensee to immediately perform an evaluation of this potential deficiency to determine compliance with its current licensing basis; (3) deny all requests for license amendments for the expansion of spent fuel pool capacity until these safety concerns are fully resolved[1]; and (4) after evaluation by each licensee, if the NRC determines that there is little or no risk to public health and safety, the NRC may issue a Notice of Enforcement Discretion which represents a determination by the NRC not to enforce an applicable technical specification or license condition.

As a basis for his requests, the Petitioner asserted that approximately 1 1/2 years before the petition was submitted, the NRC was informed of a potential substantial nuclear safety hazard at the Susquehanna Steam Electric Station (SSES) operated by Pennsylvania Power and Light Company (PP&L or Licensee) and that the NRC overlooked the need to inform utilities of this potential problem. The Petitioner claimed that this hazard involves a major design flaw such that, during a design-basis loss-of-coolant accident, the electrical power to the fuel pool cooling system would be turned off, resulting in loss of cooling for the spent fuel pool. Petitioner alleged that, as a result of the loss-of-coolant-accident, radiation levels in the reactor building would prohibit operators from entering the reactor building to restart the system. Petitioner claimed that, if cooling is not restored, the water in the spent fuel pool will boil, water will evaporate and, since the valves that must be opened to provide replacement water are located within the inaccessible reactor building, replacement water cannot be provided. Petitioner postulated that this would result in high onsite and offsite radiation

[1] This request by Petitioner is not within the scope of the 2.206 process as it does not request enforcement action as is more fully discussed in my letter transmitting this Director's Decision to Petitioner. Accordingly, it will not be further addressed in this Director's Decision.

levels and a failure of the spent fuel in the pool and a consequent release of massive amounts of airborne radioactivity outside of primary and secondary containment. Petitioner alleged further that the residual heat removal system could not cool the fuel pool under accident conditions, and that if replacement water could be provided, temperature and humidity conditions inside the reactor building would cause the emergency systems to fail, resulting in additional fuel failure and failure of the primary and secondary containment.

In a letter of May 5, 1994, the Director of the Office of Nuclear Reactor Regulation acknowledged receipt of the petition and denied the Petitioner's requests for immediate relief. In the acknowledgment letter, he informed the Petitioner that the remaining requests were being evaluated under section 2.206 of the Commission's regulations and that action would be taken in a reasonable time.

The NRC Staff's review of the issues related to spent fuel storage pool safety raised in the April 13, 1994 petition is now complete. As explained below, the NRC Staff has taken actions that, in part, address Petitioner's requests. A discussion of these issues and the NRC response to the Petitioner's requests follows.

II. DISCUSSION

On November 27, 1992, a report was filed pursuant to 10 C.F.R. Part 21 by two contract engineers at SSES, which notified the Commission of potential design deficiencies in spent fuel pool decay heat removal systems and containment systems at the Susquehanna Steam Electric Station. The report noted that, under certain conditions, systems designed to remove decay heat from the spent fuel pool would be unable to perform their intended function and that, due to concurrent plant conditions, it would not be possible for operators to place backup systems in service or that backup systems would also otherwise be unable to perform their intended function. The report contended that, under such conditions, the spent fuel pool could reach boiling conditions and that the adverse environment created by a boiling pool would render systems designed to remove decay heat from the reactor core and systems designed to limit the release of fission products to the environment unable to perform their intended function. The ultimate consequence of this condition would be the failure of fuel in both the reactor vessel and the spent fuel pool and a substantial release of fission products to the environment that would cause significant harm to the public health and safety.

The NRC Staff determined initially that the issues appeared to be of low safety significance because of the low probability that the necessary sequence of events would take place. Specifically, the NRC Staff observed that a loss-of-coolant accident followed by multiple failures of emergency core cooling systems would be necessary to achieve the adverse radiological conditions that would preclude operator actions to ensure continued adequate decay heat removal from the spent fuel pool. On this basis, the NRC Staff determined that immediate actions to ensure public health and safety were not warranted.

However, because of the complex nature of the issues raised in the Part 21 report, the NRC Staff undertook an extensive evaluation of the matter which continued from November 1992 to June 1995. The NRC Staff review process included information-gathering trips to the Licensee's engineering offices and to the SSES, public meetings with the Licensee, public meetings and written correspondence with the authors of the Part 21 report, and numerous written requests for information to the Licensee and corresponding responses. The Staff issued Information Notice 93-83, "Potential Loss of Spent Fuel Pool Cooling After a Loss-of-Coolant Accident," on October 7, 1993, which informed licensees of all operating reactors of the nature of the issues raised in the Part 21 report.

The NRC Staff reviewed and evaluated the plant design and expected operation of plant equipment with respect to the various event sequences described in the Part 21 report. The Staff also evaluated the response of plant equipment to a broader range of initiating events than was identified in the Part 21 report. For example, the Staff considered the safety significance of a loss of spent fuel pool decay heat removal capability resulting from loss of offsite power events, from seismic events, and from flooding events. The Staff considered the potential for such events to lead to spent fuel pool boiling sequences that could in turn jeopardize safety-related equipment needed to maintain reactor core cooling. The NRC Staff conducted both deterministic and probabilistic evaluations to fully understand the safety significance of the issues raised. In addition, the Staff evaluated the impact of certain modifications made by the Licensee during the course of the NRC Staff's review. Finally, the Staff examined issues associated with the design of the spent fuel pool cooling system to determine the extent to which the Licensee's design and operation met the applicable regulatory requirements.

The NRC Staff issued a draft safety evaluation addressing the issues raised in the Part 21 report regarding SSES for comment on October 25, 1994. After receiving comments from the Licensee, the authors of the Part 21 report and the Advisory Committee on Reactor Safeguards, the Staff issued a final safety evaluation regarding the issues raised in the Part 21 report for the SSES on June 19, 1995 (SSES SE).[2]

[2] Letter to R. Byram, PP&L, from J. Stolz, NRC, "Susquehanna Steam Electric Station, Units 1 and 2, Safety Evaluation Regarding Spent Fuel Pool Cooling Issues (TAC No. M85337), dated June 19, 1995.

In the SSES SE, the Staff documented the deterministic and probabilistic evaluations regarding the spent fuel pool issues raised in the Part 21 report and resulting conclusions. On the basis of the deterministic analysis of the plant as it was configured at the time the SSES SE was prepared, the NRC Staff concluded that systems used to cool the spent fuel storage pool are adequate to prevent unacceptable challenges to safety-related systems needed to protect the health and safety of the public during design-basis accidents.

On the basis of the probabilistic evaluation, the NRC Staff concluded that the specific scenario involving a large radionuclide release from the reactor vessel, which was described in the Part 21 report, is a sequence of very low probability. The NRC Staff's evaluation concluded that, even with consideration of the additional initiating events described above, "loss of spent fuel pool cooling events" represented events of low safety significance at the time the Part 21 report was submitted. However, the Staff also concluded that the plant modifications and procedural upgrades made during the course of the Staff's review, which included removal of the gates that separate the spent fuel storage pools from the common cask storage pit, installation of remote spent fuel pool temperature and level indication in the control room, and numerous procedural upgrades, provided a measurable improvement in plant safety and that these conclusions had potential generic implications. In summary, with regard to loss of spent fuel pool cooling events, the design of the SSES facility was adequate to protect public health and safety.

The Staff issued Information Notice 93-83, Supplement 1, "Potential Loss of Spent Fuel Pool Cooling After a Loss-of-Coolant Accident or a Loss of Offsite Power," to all power reactor licensees on August 24, 1995, in which the SSES SE was summarized. The information notice also described the Staff's plans to undertake an action plan to evaluate the generic concerns raised in the SSES SE and to address certain additional concerns arising from a special inspection at a permanently shutdown reactor facility.[3] The generic action plan, entitled "Task Action Plan for Spent Fuel Storage Pool Safety" (Task Action Plan) was issued on October 13, 1994, and included the following actions: (1) a search for and analysis of information regarding spent fuel storage pool issues, (2) an assessment of the operation and design of spent fuel storage pools at selected reactor facilities, (3) an evaluation of the assessment findings for safety concerns,

[3] On January 25, 1994, the licensee for Dresden Unit 1, a permanently shutdown facility, discovered approximately 55,000 gallons of water in the basement of the unheated Unit 1 containment. The water originated from a rupture of the service water system that occurred due to freeze damage. The licensee investigated further and found that, although the fuel transfer system was not damaged, there was a potential for a portion of the fuel transfer system inside containment to fail and result in a partial draindown of the spent fuel pool that contained 660 spent fuel assemblies. The NRC issued Bulletin 94-01, "Potential Fuel Pool Draindown Caused by Inadequate Maintenance Practices at Dresden Unit 1," on April 8, 1994, to all licensees with permanently shutdown reactors who had spent fuel stored in spent fuel pools. The NRC requested that such licensees take certain actions to ensure that spent fuel storage safety did not become degraded.

and (4) selection and execution of an appropriate course of action based on the safety significance of the findings.

As part of its review under the Task Action Plan, the Staff performed assessment visits to four operating reactors. The Staff also reviewed operating experience, as documented in Licensee Event Reports and other information sources, as well as in previous studies of spent fuel pool issues. Finally, the Staff gathered detailed design data for every operating reactor and analyzed these data to identify potential safety issues.

The NRC Staff completed its work under the Task Action Plan in July 1996. The Staff forwarded the results of its review to the Commission on July 26, 1996.[4] In the report, the Staff concluded that existing spent fuel storage pool structures, systems, and components provide adequate protection for public health and safety. Protection is provided by several layers of defense involving accident prevention (e.g., quality controls on design, construction, and operation), accident mitigation (e.g., multiple cooling systems and multiple makeup water paths), radiation protection, and emergency preparedness. Design features addressing each of these areas for spent fuel storage for each operating reactor have been reviewed and approved by the Staff. In addition, the limited risk analyses available for spent fuel storage suggest that current design features and operational constraints cause issues related to spent fuel pool storage to be a small fraction of the overall risk associated with an operating light-water reactor.

Notwithstanding the findings resulting from the Task Action Plan, the NRC Staff reviewed each operating reactor's spent fuel pool design to identify strengths and weaknesses, and to identify potential areas for safety enhancements. The NRC Staff identified seven categories of design features that reduce the reliability of spent fuel pool decay heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The NRC Staff determined that these design features existed at twenty-two sites.

As the Staff has concluded that present facility designs provide adequate protection of public health and safety, possible safety enhancements will be evaluated pursuant to 10 C.F.R. § 50.109(a)(3). The analyses for possible safety enhancement backfits will consider whether modifications to the plant design to address the plant-specific design features identified by the NRC Staff could provide a substantial increase in the overall protection of public health and safety and whether such modifications could be justified on a cost-benefit basis.

The NRC Staff also identified three additional categories of design features that may have the potential to reduce the reliability of spent fuel pool decay

[4] Memorandum to the Commission, from J. Taylor, "Resolution of Spent Fuel Storage Pool Action Plan Issues," dated July 26, 1996.

heat removal, increase the potential for loss of spent fuel coolant inventory, or increase the potential for consequential loss of essential safety functions at an operating reactor. The NRC Staff preliminarily determined that these design features existed at eleven sites. However, the Staff has insufficient information at this time to determine whether backfits pursuant to section 50.109(a)(3) are warranted. For plants identified as having design features in these three categories, the NRC Staff will gather and evaluate additional information prior to determining whether to require any backfits.

In addition to the plant-specific analyses described above for twenty-two sites, which will address certain design features, the NRC Staff plans to address issues relating to the functional performance of spent fuel pool decay heat removal, as well as the operational aspects related to coolant inventory control and reactivity control, for *all* operating reactors. The Staff plans to expand the proposed, performance-based rule for shutdown operations at nuclear power plants (10 C.F.R. § 50.67) to encompass fuel storage pool operations to address these performance and operational considerations.

The NRC Staff has sent the July 26, 1996 report to all licensees. For those licensees whose plants have one or more of the design features that warrant an analysis of possible plant-specific safety enhancements, the Staff has provided an opportunity for licensees to comment on (1) the accuracy of the NRC Staff's understanding of the plant design, (2) the safety significance of the design concern, (3) the cost of potential modifications to address the design concern, or (4) the existing protection from the design concern provided by administrative controls or other means. In developing a schedule and plans for conducting the plant-specific regulatory analyses, the NRC Staff will consider comments received from licensees.

III. RESPONSE TO PETITIONER'S REQUESTS

A. Issuance of Generic Communications to Licensees on Failure of Spent Fuel

The NRC Staff has issued three information notices on matters related to adequate decay heat removal from the spent fuel pool. Information Notice 93-83, "Potential Loss of Spent Fuel Pool Cooling After a Loss-of-Coolant Accident," was issued on October 7, 1993, and described the concerns raised in the November 27, 1992, Part 21 report. Information Notice 93-83, Supplement 1, was issued on August 24, 1995, to inform licensees of the results of the NRC review of the concerns at SSES. Information Notice (IN) 95-54, "Decay Heat Management Practices During Refueling Outages," was issued on December 1, 1995. It described recent NRC assessments of events at certain plants regarding licensee control of refueling operations and the methods for removing decay heat produced from the irradiated fuel stored in the spent fuel pool during refueling outages. In IN 95-54, the NRC Staff communicated to licensees that the plant-specific events described in IN 95-54 and the previous information notices illustrated the importance of assuring that (1) planned core offload evolutions, including refueling practices and irradiated decay heat removal, are consistent with the licensing basis, including the Final Safety Analysis Report, technical specifications, and license conditions; (2) changes are evaluated through the application of the provisions of section 50.59, as appropriate; and (3) all relevant procedures associated with core offloads have been appropriately reviewed.

As described in Section II, the NRC Staff also forwarded the July 26, 1996 report on spent fuel to all licensees. The NRC has determined that these generic communications to power reactor licensees are sufficient to provide licensees with information on spent fuel pool cooling issues.

Petitioner's request that the NRC issue an information notice or other appropriate notification forwarding all information in its possession to all power reactor licensees regarding the potential failure of fuel in spent fuel pools is granted to the extent that the NRC Staff has provided information on spent fuel storage safety issues by way of the generic communications and correspondence described above.

Petitioner's request that the NRC remind licensees of their responsibilities to perform timely operability determinations in accordance with their technical specifications is granted to the extent that the NRC has communicated to licensees the importance of conducting relevant spent fuel pool decay heat removal activities in accordance with technical specifications and other plant-specific applicable regulatory requirements in IN 95-54.

B. Licensee Evaluation of Compliance with the Licensing Basis

Petitioner requested that the Staff direct each licensee to immediately perform an evaluation of the potential failure of the fuel in the spent fuel pool to determine compliance with the current licensing basis. The NRC Staff examined the issue of the conformance of the existing plant design with the facility licensing basis in great detail for SSES.[5] As documented in the SSES SE, the NRC Staff concluded that neither operation of spent fuel pool cooling during design-basis accident conditions nor mitigation of the effects of a loss of spent fuel pool cooling during normal and design-basis accident conditions could be considered part of the SSES licensing basis with the exception of mitigation

[5] In the SSES spent fuel pool design review, the NRC Staff determined which regulations the Licensee was required to comply with. In addition, operational limitations were extracted from plant-specific licensing documents including the Final Safety Analysis Report, technical specifications, license amendments, and other docketed correspondence.

of loss of spent fuel pool cooling following a design-basis seismic event. In general, the NRC Staff's conclusion is based on the fact that, with respect to operation of the spent fuel pool cooling systems during normal and design-basis accident conditions, the SSES operating license safety evaluation report[6] (SER) did not cite the applicable General Design Criteria (GDC) (GDC 44 and GDC 61 in its entirety) as the basis for finding the system acceptable. With respect to the mitigation of the effects of a loss of spent fuel pool cooling during normal and design-basis accident conditions, in the SSES SE, the Staff found no evidence that it expected secondary containment systems to accommodate the added heat and vapor loads that would follow a sustained loss of spent fuel pool cooling for any design-basis event with the specific exception of a design-basis seismic event.

The NRC Staff's finding that mitigation of a loss of spent fuel pool cooling following a design-basis seismic event was part of the licensing basis was based on specific statements in the SER that acceptance of a nonseismic spent fuel pool cooling system was an acceptable deviation from GDC 2, based, in part, on the existence of an adequate standby gas treatment system. At the time of the original licensing review, the Staff did not attempt to extend the licensing basis for loss of spent fuel pool cooling following a design-basis seismic event to any other design-basis events.

During its review of spent fuel pool concerns at SSES, the NRC Staff raised its concerns to the Licensee regarding the ability to mitigate a loss of spent fuel pool cooling following a seismic event. As discussed in the SSES SE, the Licensee took certain actions, including implementing routine operation of the adjacent spent fuel pools in a cross-connected manner, that adequately addressed NRC Staff concerns. In summary, with regard to the spent fuel pool issues raised by Petitioner, SSES design and operation conform to the facility licensing basis.

As part of the Task Action Plan, the Staff considered on a generic basis the history of regulatory requirements related to spent fuel pools as they were applied in plant licensing activities. The Staff observed that such regulatory requirements evolved since the first nuclear power plants were licensed and observed that specific regulatory guidance on the design of spent fuel pool cooling systems was not issued until 1975 when the Standard Review Plan was issued, after construction permits for most currently operating reactors were issued. Because the regulatory requirements were not constant during the era when the Staff was conducting licensing reviews for the current generation of operating reactors, the Staff observed that approved designs varied from plant to plant. However, the Staff did conclude, based on information available during the recent review of spent fuel pool system design, that all operating reactors had design features for

[6] U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Susquehanna Steam Electric Station, Units 1 and 2," NUREG-0776, April 1981.

spent fuel storage (addressing accident prevention functions, accident mitigation functions, radiation protection functions, and emergency preparedness functions) that had been reviewed and approved by the Staff and that these facility designs were in compliance with the NRC requirements applied at the time of licensing.

Although the NRC Staff concluded that plants were in compliance with the NRC design requirements applied at the time of licensing, the NRC Staff also recently reviewed certain operating practices at all operating reactors to verify that the plants were being operated consistent with the plant design described in the licensing basis.[7] Specifically, the Staff reviewed refueling outage practices with regard to offloading irradiated fuel into the spent fuel pool. The Staff concluded on the basis of the information collected and reviewed and the specific licensee actions taken and commitments made during the course of this review, core offload practices are currently consistent with the spent fuel pool decay heat removal licensing basis for all plants or will be prior to the next refueling outage. However, during the course of the review, the Staff determined that nine sites (fifteen units) needed to perform evaluations or make modifications, pursuant to section 50.59 or 10 C.F.R. § 50.90, to ensure that their reload practices adhered to their licensing basis. This is an indication that these plants may have previously performed full core offloads inconsistent with their licensing basis.

The Staff has documented the details of its findings in recent NRC inspection reports for each of the nine sites. The Staff will take regulatory action, as appropriate, to address these potential operational nonconformances.

Petitioner requested that evaluations be performed of Petitioner's concern regarding spent fuel pool cooling by licensees to determine compliance with their licensing basis. This request is granted to the extent that the NRC Staff has performed evaluations of both the design and operational aspects of spent fuel pool storage issues for all operating reactors to the extent described above.

C. Issuance of Notices of Enforcement Discretion

The Atomic Energy Act of 1954, as amended (the Act), and the Energy Reorganization Act of 1974, as amended, give NRC the authority to take enforcement actions necessary to ensure compliance with certain provisions of those acts and with NRC regulations, orders, and licenses. Licenses include specified license conditions and facility technical specifications that are part of the license. The NRC's enforcement policy is published in NUREG-1600, "General Statement of Policy and Procedures for NRC Enforcement Actions," July 1995 (Enforcement Policy).

[7] Memorandum to the Commission, from J. Taylor, dated May 21, 1996.

The Enforcement Policy recognizes that, on occasion, circumstances may arise concerning a licensee's compliance with a Technical Specification Limiting Condition for Operation or with some other license conditions that would involve an unnecessary plant transient or the performance of plant testing that is inappropriate for the specific plant conditions. For such occasions, the Enforcement Policy provides a process, referred to as a Notice of Enforcement Discretion (NOED), by which the NRC Staff, upon request from the licensee, may choose not to enforce compliance with the applicable technical specifications or license conditions in limited circumstances. A NOED will only be issued if the NRC Staff is satisfied that the action is consistent with public health and safety.

In Request 4, Petitioner seems to suggest that the exercise of enforcement discretion by issuance of a NOED may be appropriate concerning spent fuel pool issues raised in the petition. As discussed in Section III.B, with regard to potential failure of fuel in spent fuel pools, the NRC Staff has determined that spent fuel pools contain design features that were reviewed and approved by the Staff. In addition, these facility designs have been found to be in compliance with NRC requirements applied at the time of licensing. Based upon the review of the information provided in the petition, the NRC Staff has not identified any circumstances warranting the issuance of a NOED. If a situation is presented to the Staff involving a request for a NOED, such a request will be considered in accordance with the Enforcement Policy.

IV. CONCLUSION

Based on the NRC Staff's evaluation described above, the NRC Staff has issued generic communications responsive to Petitioner's Request 1. In addition, the NRC Staff has reviewed the aspect of compliance of NRC-licensed facilities in the area of spent fuel pool design responsive in part to Petitioner's Request 2. To this extent, the petition is granted. With regard to Petitioner's Request 4, the NRC Staff has concluded that there has been no need for issuance of NOEDs regarding potential failure of fuel in spent fuel pools.

A copy of this Final Director's Decision will be placed in the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW, Washington, DC, and at the local public document room for all power reactor licensees.

A copy of this Final Director's Decision will also be filed with the Secretary of the Commission for review in accordance with 10 C.F.R. § 2.206(c) of the Commission's regulations. This Decision will become the final action of the

Commission 25 days after its issuance, unless the Commission, on its own motion, institutes review of the Decision within that time.

FOR THE NUCLEAR
REGULATORY COMMISSION

Frank J. Miraglia, Jr., Acting
Director
Office of Nuclear Reactor
Regulation

Dated at Rockville, Maryland,
this 6th day of November 1996.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Frank J. Miraglia, Jr., Acting Director

In the Matter of **Docket Nos. 50-335**
50-389
(License Nos. DPR-67
NPF-16)

FLORIDA POWER AND LIGHT COMPANY
(St. Lucie Nuclear Power Plant, Units 1 and 2) **November 18, 1996**

The Acting Director of the Office of Nuclear Reactor Regulation denies a petition dated June 12, 1996, filed with the Nuclear Regulatory Commission (NRC) pursuant to 10 C.F.R. § 2.206 by Thomas J. Saporito on behalf of himself and the National Litigation Consultants (Petitioners). The Petitioners requested the NRC (1) to issue a confirmatory order requiring that the Florida Power & Light Company (Licensee) not operate the St. Lucie Plant, Unit 1, above 50% of its power-level capacity, (2) to require the Licensee to specifically identify the "root cause" for the premature failure of the steam generator tubing, and (3) to require the Licensee to specifically state what corrective measures will be implemented to prevent recurrence of steam generator tube failures in all the steam generators in Unit 1 and Unit 2. The Petitioners' requests were based on assertions that (1) the Licensee's Unit 1 steam generator tubes have degraded to the extent that more than 2500 of the tubes have been plugged, (2) the Licensee has not identified the root cause for the premature failure of the steam generator tubing, (3) the Licensee will most likely experience similar tube ruptures on other generators at the station, and (4) the Licensee's "FSARs [Final Safety Analysis Reports] and the NRC's CFRs [*Code of Federal Regulations*] require that the integrity of the primary systems on Unit 1 and Unit 2 not be breached."

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

I. INTRODUCTION

On June 12, 1996, Mr. Thomas J. Saporito, Jr., on behalf of himself and the National Litigation Consultants (Petitioners), filed a petition with the U.S. Nuclear Regulatory Commission (NRC or Commission) pursuant to 10 C.F.R. § 2.206. The Petitioners requested the Commission (1) to issue a confirmatory order requiring that the Florida Power & Light Company (FP&L or Licensee) not operate St. Lucie Plant, Unit 1, above 50% of its power-level capacity, (2) to require the Licensee to specifically identify the "root cause" for the premature failure of the steam generator tubing, and (3) to require the Licensee to specifically state what corrective measures will be implemented to prevent recurrence of steam generator tube failures in all the steam generators in Unit 1 and Unit 2.

The Petitioners' requests are based on assertions that (1) the Licensee's Unit 1 steam generator tubes have degraded to the extent that more than 2500 of the tubes have been plugged, (2) the Licensee has not identified the root cause for the premature failure of the steam generator tubing, (3) the Licensee will most likely experience similar tube ruptures on other steam generators at the station, and (4) the Licensee's "FSARs [Final Safety Analysis Reports] and the NRC's CFRs [*Code of Federal Regulations*] require that the integrity of the primary systems on Unit 1 and Unit 2 not be breached."

The petition has been referred to my office pursuant to section 2.206 of the Commission's regulations. By letter dated July 8, 1996, an acknowledgment of receipt of the petition was sent to the Petitioners. In that letter, the Petitioners were informed that the NRC would take appropriate action within a reasonable time. I have completed my evaluation of the matters raised by the Petitioners and have determined that, for the reasons stated below, the petition is denied.

II. DISCUSSION

The NRC Staff's evaluation of the Petitioners' requests follows.

A. Issue a Confirmatory Order Requiring That the Licensee Not Operate Unit 1 Above 50% of Its Power-Level Capacity

In a meeting held at NRC Headquarters on July 3, 1996, the Licensee presented the inspection and repair history for the Unit 1 steam generator tubes.[1] The Licensee has performed fifteen inspections since commercial operation began in December 1976. For the most recent inspection, completed in June 1996, the Licensee inspected the full length of all active tubes using a bobbin coil.[2] In addition, the Licensee used a motorized rotating pancake coil[3] (MRPC) to inspect all expansion transition joints and drilled support intersections in the hot and cold legs, all free-span locations having bobbin coil indications,[4] and free-span tube regions in the upper two support areas in the hot legs. The inspection was based on the Electric Power Research Institute (EPRI) report "PWR Steam Generator Examination Guidelines," dated November 1992. Defective tubes having circumferential indications, axial indications, or volumetric indications[5] were plugged and removed from service.

Including tubes plugged during earlier outages, 2159 of 8519 tubes (25.3%) in the "A" steam generator and 1834 of 8519 tubes (21.5%) in the "B" steam generator have been plugged and removed from service. The Licensee performed an evaluation that showed that the plant could be safely operated at full power with the reduced reactor coolant flow resulting from the increased number of plugged tubes.[6] The NRC reviewed the Licensee's evaluation and concluded that it was acceptable and that the units could be operated at full power. The Staff's evaluation is documented in a safety evaluation dated July 9, 1996.

In the meeting on July 3, 1996, the Licensee presented a preliminary run-time analysis for Unit 1, which was used to determine the length of steam generator operation before the need for further tube inspections to ensure adequate tube integrity. The Licensee stated that the preliminary results of its analysis support a tube inspection interval of 15 months for the current Unit 1 cycle that started in July 1996. The Licensee also stated that in situ pressure testing of the steam generator tubes during the spring 1996 outage indicated that the most severely degraded tubes had adequate structural integrity and satisfied the safety margins in NRC's Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." On the basis of the results of the in situ pressure tests, the Staff concluded that adequate assurance of tube integrity existed to allow operation pending completion of the Licensee's run-time analysis. The NRC is

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[1] NRC Meeting Summary, Subject: "Steam Generator Inspection, Repair and Operating Issues — St. Lucie Unit 1," dated July 16, 1996.

[2] The bobbin coil is used for a general screening of tubes for indications of possible defects, while the motorized rotating pancake coil (MRPC) probe is used to further characterize bobbin coil indications. The MRPC is also used to inspect regions susceptible to circumferentially orientated degradation.

[3] *See* note 2.

[4] *See* note 2.

[5] Circumferential indications are crack-like indications orientated on the diameter of the tube. Axial indications are crack-like indications orientated on the long axis of the tube. Volumetric indications are areas of general reduction in tube-wall thickness with no specific orientation.

[6] FP&L Letter, "Thermal Margin and RCS Flow Limits," dated June 1, 1996.

currently reviewing the Licensee's analysis, which was submitted October 24, 1996.

The plant Technical Specifications for each of the units specify leakage limits for the reactor coolant pressure boundary, including steam generator tube leakage. If a tube leaks beyond the allowed limits, the unit must be shut down. The plant off-normal operating procedures for St. Lucie Units 1 and 2 also include criteria for shutdown based on EPRI TR-104788, "PWR Primary to Secondary Leak Guidelines," dated May 1995, which are more conservative than the limits in the plant Technical Specifications. Finally, if a tube fails, the plant's Emergency Operating Procedures contain the specific actions necessary for the operators to shut down and cool down the plant to mitigate the consequences of the event.

Thus, as required, the Licensee has implemented measures for both units to protect public health and safety in the unlikely event that tube integrity is compromised. These measures include a primary-to-secondary leakage monitoring program and emergency operating procedures. The leakage monitoring program provides early warning of tube leakage. The steam generator blowdown monitor and condenser air ejector monitor at each of the units continuously monitors the radioactivity level in the main steamline. A significant increase in the instrument readings, which would result from a relatively small tube leak, will cause an alarm to alert the operators to the change in radioactivity levels and potential tube leakage.

On the basis of the information submitted, the NRC Staff has concluded that the operation of the Unit 1 steam generators at full power poses no undue risk to public health and safety.

B. Require the Licensee to Specifically Identify the "Root Cause" for the Premature Failure of the Steam Generator Tubing

It is not clear how the Petitioners define "premature failure"; however, since there have not been any steam generator tube ruptures at St. Lucie Units 1 or 2, it is assumed the reference is to tube degradation. Many of the tubes in the Unit 1 steam generators have degraded as a result of corrosion and/or mechanical conditions. The root cause of tube degradation in steam generators is the interaction of water chemistry, thermal-hydraulic design, materials selection, fabrication methods, and operating conditions. The causes of tube degradation are well understood by the industry and are documented in the public record. The root causes for the St. Lucie steam generator tube degradations were presented to the NRC Staff in a meeting on August 27, 1986.[7]

[7] NRC Meeting Summary, Subject: "Summary of August 27, 1986 Meeting with FP&L and NRC Staff Regarding Steam Generator Tube Degradation Mechanism," dated September 12, 1986.

The Licensee has identified to the NRC modes of degradation that have affected the steam generator tubes in both St. Lucie Units 1 and 2 in its response of June 23, 1995, to NRC Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes," and in the meeting of July 3, 1996. The degradation modes identified include intergranular attack, stress-corrosion cracking and denting. Intergranular attack refers to localized attack at and adjacent to grain boundaries of tube material, with relatively little corrosion of the grains. Intergranular stress-corrosion cracking refers to cracking caused by the simultaneous presence of stress and a specific corrosive medium. Denting is the accumulation of corrosion products at the tube-to-tube support plate that causes plastic deformation of the tube. The Licensee has identified locations of these degradations in the tubes during the most recent steam generator inspection of St. Lucie Unit 1.[8] They include egg crate and drilled tube support plates, free spans, expansion transition regions, and sludge pile areas. In every case, the root cause of tube degradation can be attributed to material selection, water chemistry, fabrication methods, or residual stresses at the affected location.

The Staff concludes that the Licensee understands and has identified the root cause of tube degradation at St. Lucie Units 1 and 2.

C. Require the Licensee to Specifically State What Corrective Measures Will Be Implemented to Prevent Recurrence of Steam Generator Tube Failures in All the Steam Generators in Unit 1 and Unit 2

As previously discussed, degradation of the steam generator tubing is caused by the interaction of water chemistry, thermal-hydraulic design, materials selection, fabrication methods, and operating conditions. The Licensee has applied corrective measures in order to reduce the rate of tube degradation. For example, the rate of tube degradation may be reduced through improvements in water chemistry. The Licensee follows industry guidelines[9] on secondary water chemistry for both units, and these guidelines represent a significant improvement over the guidelines followed when Unit 1 began operating. The guidelines have stringent requirements and limitations on specific types and amounts of chemicals in the primary and secondary water to mitigate corrosion. Replacement steam generators having improved design, for example, better material selection and tube support configuration, have had much better operating experience than the earlier steam generators, such as those at St. Lucie. The Licensee plans to replace the Unit 1 steam generators in October 1997 with steam generators that incorporate these design improvements.

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[8] *See* note 1.
[9] FP&L Letter, "Generic Letter 95-03 Response," dated June 23, 1995.

The NRC Staff focuses on ensuring adequate tube integrity by requiring Licensee compliance with applicable regulations and Technical Specification requirements. The Staff uses its field inspections, meetings with the Licensee, and licensing reviews to ensure that the Licensee satisfies the regulations[10] and plant Technical Specifications as they apply to steam generator tube integrity and that appropriate inspection methods and repair criteria are used to address specific forms of degradation. Plant Technical Specifications define degraded and defective tubes, specify the scope of inspections and reporting requirements and set forth tube plugging criteria and limits for allowable leakage in the reactor coolant system. NRC regulations and plant Technical Specifications require that steam generator tube degradation be managed through a combination of inservice inspection, repair of tubes exceeding the plugging criteria in the plant Technical Specifications, primary-to-secondary leakage monitoring, and structural and run-time analyses to ensure that safety objectives are met. On the basis of the information provided by the Licensee in the meeting on July 3, 1996, and the Staff's onsite inspection, the Staff has concluded that the Licensee is in compliance with these requirements.

In summary, the Licensee's corrective measures to reduce the rate of steam generator tube degradation and continued compliance with NRC regulations and plant Technical Specification requirements provide reasonable assurance that steam generator tube integrity at St. Lucie Units 1 and 2 will be maintained.

III. CONCLUSION

On the basis of the fact that (1) the Licensee has performed adequate steam generator tube inspections that identified areas of degradation, (2) the Licensee has completed analyses and repairs of degraded tubes, (3) the Licensee's in situ pressure testing of degraded tubes indicated adequate structural integrity remains, (4) the Licensee is monitoring primary-to-secondary leakage on a continuing basis, and (5) the Licensee is complying with NRC regulations and plant Technical Specifications, I have concluded that a confirmatory order limiting St. Lucie Unit 1 to 50% of its power-level capacity is not warranted and that the Licensee has identified the root cause of tube degradation and implemented adequate corrective measures to provide reasonable assurance that steam generator tube integrity will be maintained at St. Lucie Units 1 and 2.

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[10]The NRC regulations that require steam generator tube integrity be maintained include 10 C.F.R. Part 50, Appendix A, General Design Criteria for Nuclear Power Plants, Criterion 1 — Quality Standards and Records, Criterion 14 — Reactor Coolant Pressure Boundary, Criterion 30 — Quality of Reactor Coolant Pressure Boundary, Criterion 31 — Fracture Prevention of Reactor Coolant Pressure Boundary, and Criterion 32 — Inspection of Reactor Coolant Pressure Boundary; 10 C.F.R. Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants; and 10 C.F.R. § 50.55a, which specifies codes and standards for nuclear power plants.

For the reasons previously discussed, no basis exists for taking any further action in response to the petition. As provided in 10 C.F.R. § 2.206(c), a copy of the Decision will be filed with the Secretary of the Commission for the Commission's review. 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

Frank J. Miraglia, Jr., Acting
Director
Office of Nuclear Reactor
Regulation

Dated at Rockville, Maryland,
this 18th day of November 1996.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Frank J. Miraglia, Jr., Acting Director

In the Matter of **Docket No. 50-309**

MAINE YANKEE ATOMIC POWER COMPANY
(Maine Yankee Atomic Power Station) **November 20, 1996**

The Acting Director of the Office of Nuclear Reactor Regulation (NRR) denies a petition filed with the Nuclear Regulatory Commission (NRC or Commission) by letter dated January 20, 1996, by Anne D. Burt on behalf of the Friends of the Coast — Opposing Nuclear Pollution (Petitioner), requesting that actions be taken regarding the Maine Yankee Atomic Power Station (Maine Yankee) operated by the Maine Yankee Atomic Power Company (the Licensee). The petition is denied based on the Acting Director's analysis of the technical issues, set forth in the Decision, which analysis showed no technical basis warranting granting the petition. Petitioner's requests for immediate action and for an informal hearing were denied by the Director, NRR, by letter dated May 13, 1996, for the reasons stated in that letter.

TECHNICAL ISSUES DISCUSSED

The following technical issues are discussed: Adequacy of containment design at or above originally authorized power level; Microfissuring of low-ferrite stainless steel weldments.

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

I. INTRODUCTION

By letter dated January 20, 1996, Ms. Anne D. Burt filed a petition with the U.S. Nuclear Regulatory Commission (NRC), pursuant to 10 C.F.R. § 2.206, on behalf of the Friends of the Coast — Opposing Nuclear Pollution (the Petitioner) requesting that actions be taken regarding the Maine Yankee Atomic Power Station (Maine Yankee), operated by the Maine Yankee Atomic Power Company (the Licensee). The petition requests that the Commission take expedited action to (1) suspend the operating license of Maine Yankee pending resolution of the petition; (2) examine and test by plug sampling — or other methods approved by the American Society of Mechanical Engineers — all large piping welds that may have been susceptible to microfissures at the time of construction; (3) reanalyze the Maine Yankee containment as one located in an area where seismic risk is not "low"; (4) reduce the licensed operating capacity of Maine Yankee to a level consistent with a flawed containment and/or flawed reactor coolant piping welds; (5) hold an informal public hearing in the area of the plant regarding the petition; and (6) place the Petitioner on service and mailing lists relevant to the group's interests in safety at Maine Yankee and intention to participate in all public forums opened by the NRC.

By letter dated May 13, 1996, the Director, Office of Nuclear Reactor Regulation (NRR), NRC, acknowledged the NRC's receipt of the petition, and, for the reasons stated in the letter, denied Petitioner's request for immediate action suspending the operating license or reducing the licensed operating capacity of Maine Yankee (Requests 1 and, in part, 4). In addition, for reasons stated in the May 13, 1996 letter, the Director denied the Petitioner's request for an informal hearing (Request 5). The Director also stated in the May 13, 1996 letter that the request that the NRC place Petitioner on service and mailing lists relevant to its interests in safety at Maine Yankee and its intention to participate in all public forums opened by the NRC (Request 6) was moot, as Petitioner's attorney had already been added to the Maine Yankee service list. In addition, the Petitioner was informed that NRC would review the petition in accordance with section 2.206 and issue a final decision within a reasonable time.

The remaining specific requests for NRC action in the petition dated January 20, 1996, i.e., Requests 2, 3, and 4 identified above, and the issues that Petitioner raised as their bases, are addressed in this Decision. For the reasons set forth below, Petitioner's remaining requests for action pursuant to section 2.206 are denied.

II. DISCUSSION

The NRC Staff has conducted a thorough evaluation of each of the two safety-related issues raised in the petition regarding the adequacy of the containment and reactor coolant welds. Each of the issues is addressed below.

A. Adequacy of Containment Design at or Above Originally Authorized Power Level

The Petitioner asserts that the containment is inadequate for operation at any power in excess of that authorized in the original license, and may be inadequate for the originally licensed power level because of insupportable original design acceptance criteria in that the Maine Yankee containment was designed and constructed without diagonal rods. The Petitioner states that

> the Atomic Energy Commission staff recommended to the commission that a license amendment permitting this type of construction be allowed, ". . . for this plant and this plant only due to low seismic risk." Early in 1979 the MYAPS was shaken by an earthquake of 4.2 magnitude and epicentered less than ten miles from the plant site. The NRC then ordered the shutdown of five nuclear power stations including MYAPS until piping and piping supports could be seismically qualified . . . ."

The Petitioner also states that there is no public record, however, that NRC reevaluated what Petitioner asserts is a marginally acceptable containment design at Maine Yankee before it granted license amendments to operate at increased power.

The Maine Yankee containment is a reinforced concrete structure. The original NRC operating license review determined that the seismic and thermal-hydraulic design of Maine Yankee's containment structure is adequate. (The construction permit for Maine Yankee was issued on October 21, 1968, and the operating license was issued on September 15, 1972.) With its petition of January 20, 1996, the Petitioner enclosed an NRC letter of January 22, 1971, in which the Staff asked the Licensee to submit additional information related to seismic shear stress, given that there are no diagonal seismic shear reinforcements in the containment wall. Low seismicity of the site was not a factor in the Staff's acceptance of the Maine Yankee containment design without diagonal seismic reinforcement bars. As described below, acceptance by the Staff of the adequacy of the seismic design was based on the results of stress analyses.

The earthquake for which Maine Yankee was originally designed — termed a Safe Shutdown Earthquake (SSE) — is based on a Housner design response spectrum with a zero-period peak horizontal ground acceleration of 0.10*g*. The five-plant shutdown that was ordered on March 13, 1979, was triggered by a finding of an error in a piping computer program, which led to the issuance of IE Bulletin No. 79-07, "Piping Stress Analysis of Safety-Related Piping" on April 14, 1979. The earthquakes that occurred near the plant site starting on April 18, 1979, at 02 hours and 34 minutes universal time, were not a factor in the five-plant shutdown that was ordered on March 13, 1979. As a consequence of the sequence of earthquakes that occurred near the plant in April 1979 and the occurrence of the January 9, 1982 magnitude $5\frac{3}{4}$ earthquake in New Brunswick, Canada, the Licensee undertook a seismic analysis program. This program included analyses and upgrading of certain plant components and a reevaluation of the seismic hazard. Thus, the results from the seismic analyses and upgrading program were instrumental in the Staff's conclusion that the existing seismic design for Maine Yankee remained adequate. However, following its review of the seismic hazard reevaluation, the NRC Staff determined that the appropriate characterization of the ground motion for any future analysis of the plant is a high-frequency peak ground acceleration of 0.18*g* anchoring the response spectrum obtained from NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," using the 50th percentile amplification factors.

Subsequently, in 1986, the Maine Yankee Plant underwent a seismic margin assessment program. The review-level earthquake used in the seismic margin assessment had a peak ground acceleration of 0.3*g*, which is much greater than the peak ground acceleration of the SSE. The seismic safety margin program included a review of the entire plant including analysis and upgrading of certain plant components, such as Main Control Board, Control Room Auxiliary Cabinets, Service Water Piping Support, and others. As a result of this reassessment, it was established that, with the upgrades implemented at the plant, the Maine Yankee Plant can be safely shut down during an earthquake with a peak ground acceleration of 0.27*g*.

In its report "Seismic Margin Review of the Maine Yankee Atomic Power Station" (NUREG/CR-4826, Vol. 2, dated March 1987), the NRC Staff also concluded that the overall seismic margin of the plant, including the containment, was well above the 0.18*g* value and, therefore, no upgrading of the seismic design was considered necessary. Further, in the Staff report "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants" (NUREG/CR-4334, dated August 1985), it is also noted that prestressed and reinforced concrete containment structures have a large seismic margin above the SSE-level earthquake.

Additionally, numerous tests and studies conducted since the operating license review of the Maine Yankee Plant, specifically on shear stress in biaxially cracked reinforced concrete without diagonal reinforcement bars, have led to the acceptance of specified allowable shear stress by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section

III, Division 2, CC-3421.5, for reinforced-concrete containment structures. An analysis of the Maine Yankee containment structure was conducted in December 1984 by the Licensee and submitted on the Docket as an attachment to letter MN-85-27, dated February 5, 1985. The results of the study indicate that the controlling peak ground acceleration value is 0.39*g* for the ASME Code-allowable tangential shear stress caused by the SSE loading in combination with design-basis internal pressure and dead loads. This provides additional confidence on the ruggedness of the Maine Yankee containment.

Based on the above, with regard to the Petitioner's concern about the adequacy of the Maine Yankee containment structural design for earthquakes (seismic), the Staff concludes that the Maine Yankee containment is satisfactory and has adequate margin. The NRC Staff has determined that the design of the Maine Yankee containment structure without diagonal reinforcement bars is supported by analysis and poses no undue risk to public health and safety. Accordingly, Petitioner's requests for NRC action based on the seismic design of the containment are denied.

B. Microfissuring of Low-Ferrite Stainless Steel Weldments

The Petitioner asserts that the Maine Yankee emergency core cooling system (ECCS), reactor coolant piping, and other large piping have not been adequately analyzed for materials degradation to ensure integrity at power operation in excess of the originally licensed power level or under accident conditions. The Petitioner states further that the Atomic Energy Commission's concern with "microfissures" in reactor coolant system welds led to the appointment of a task force, and prompted studies and reports in 1971 (before heightened awareness of embrittlement phenomena) that concluded that the microfissures would not propagate or grow under foreseeable conditions. The Petitioner asserts that large pipe welds next to the reactor vessel have endured 23 years of corrosion, stress, vibration, and radiation, and may fail, initiating a loss-of-coolant accident, or may be subject to thermal shock failure initiated by use of the ECCS.

In a safety evaluation dated February 25, 1972, the NRC Staff concluded that the low-ferrite stainless steel weldments in large piping at Maine Yankee are acceptable because the microfissures of the type and density found in the low-ferrite stainless steel weldments of the Maine Yankee facility do not significantly impair the strength and capability of the welds, and that removal of the welds and rewelding could introduce other problems of greater safety significance than those resulting from the presence of microfissures. This evaluation was based on information provided by Battelle Columbus Laboratories, Stone and Webster Engineering Corporation, and Dr. Ernest F. Nippes of Rensselaer Polytechnic Institute. Furthermore, the Maine Yankee reactor vessel meets the requirements of 10 C.F.R. § 50.61, "Fracture Toughness Requirements for

Protection Against Pressurized Thermal Shock." In addition, the large-diameter pipe welds attached to, or next to, the reactor vessel do not receive sufficient radiation to cause embrittlement. Finally, Type 316 stainless steel weld material, in which the microfissures were discovered, is resistant to corrosion in a PWR coolant environment, and the vibratory loads are insufficient to be a concern for large-diameter piping.

In a letter to the Petitioner dated May 13, 1996, the Staff stated that in order to determine if there is any long-term safety significance of the microfissures, the Staff will review the inservice inspection results for the welds identified as being susceptible to microfissures. The Staff has now completed its review of the inservice inspection test results for welds susceptible to microfissures. The Staff's review confirmed that no unacceptable indications have been observed during inservice inspection. In addition, pressure tests have not identified any leakage. These tests indicate that 23 years of plant operation have not caused the microfissures to grow to a size detectable by inservice inspection or through-wall leakage. Plug sample testing was performed by Battelle, Columbus Laboratories, on the primary coolant system low-ferrite welds (Reference: Battelle's report dated September 17, 1971, which was transmitted by the Licensee to the NRC by letter dated September 21, 1971). As part of the inservice inspection program in accordance with 10 C.F.R. § 50.55a(g), the Licensee has been performing and continues to perform ASME Code inspections of large piping welds that may have been susceptible to microfissures at the time of construction. Additional plug sample testing would not yield any pertinent additional information and is not needed.

On the basis of the above analyses, inservice inspection, and pressure test results, microfissures are not considered a long-term safety-significant issue for Maine Yankee. Accordingly, the Petitioner's remaining requests for NRC action based on asserted microfissures in large piping welds is denied.

III. CONCLUSION

As explained above, and as requested by the Petitioner, the Staff examined the adequacy of containment design and susceptibility of welds to microfissures. For the reasons stated above, no basis exists for taking any further action in response to the petition. Accordingly, no action pursuant to section 2.206 is being taken in this matter.

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

Frank J. Miraglia, Jr., Acting
Director
Office of Nuclear Reactor
Regulation

Dated at Rockville, Maryland,
this 20th day of November 1996.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Frank J. Miraglia, Jr., Acting Director

In the Matter of	**Docket Nos. 50-282** **50-306** **72-10**
NORTHERN STATES POWER COMPANY **(Prairie Island Nuclear Generating Plant, Units 1 and 2)**	**November 27, 1996**

The Acting Director of the Office of Nuclear Reactor Regulation denies a petition dated June 5, 1995, submitted to the Nuclear Regulatory Commission (NRC) by the Prairie Island Coalition Against Nuclear Storage (PICANS), now known as the Prairie Island Coalition, and the Nuclear Information and Resource Service (Petitioners) requesting that the NRC immediately suspend the operating licenses for Prairie Island Nuclear Generating Plant, Units 1 and 2, operated by the Northern States Power Company.

Petitioner presented four concerns. Prairie Island steam generators are suffering from tube degradation and may rupture unless proper testing is conducted and corrective actions are taken. The Prairie Island reactor vessel head penetrations have stress-corrosion cracks which, if not found and corrected, may result in a catastrophic accident involving the reactor control rods. Plans for unloading dry cask storage units in an emergency were not properly reviewed by the NRC and do not satisfy NRC requirements. Finally, the physical integrity of the Prairie Island crane requires physical testing and a safety analysis before future crane use following its handling of a heavy load for an extended period of time.

For the reasons explained in the Director's Decision, the Acting Director concludes that inadequate bases exist for granting Petitioners' request.

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

I. INTRODUCTION

On June 5, 1995, the Nuclear Information and Resource Service and the Prairie Island Coalition Against Nuclear Storage (PICANS), now known as the Prairie Island Coalition (Petitioners), filed 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 Nuclear Regulatory Commission (NRC) immediately suspend the operating licenses for Prairie Island Nuclear Generating Plant, Units 1 and 2, operated by Northern States Power Company (NSP or Licensee).

II. BACKGROUND

As a basis for their request, Petitioners presented four concerns which are summarized as follows: (1) The Prairie Island steam generators are suffering from tube degradation and may rupture unless proper testing is conducted and corrective actions are taken; (2) the Prairie Island reactor vessel head penetrations (VHPs) have stress-corrosion cracks which, if not found and corrected, may result in a catastrophic accident involving the reactor control rods; (3) plans for loading and unloading of dry cask storage units in an emergency, which include storage of irradiated components in the fuel transfer canal, were not properly reviewed by NRC and do not satisfy NRC requirements; and (4) the physical integrity of the Prairie Island crane used to lift the dry cask for Prairie Island's spent fuel requires physical testing and a safety analysis before future crane use following its handling of a heavy load for an extended period of time.

By a letter dated June 19, 1995, the Director of the Office of Nuclear Reactor Regulation (NRR) denied the Petitioners' request for immediate suspension of Prairie Island Units 1 and 2 licenses. The Director stated that the NRC Staff's review of the petition did not identify any safety issues warranting immediate action at the Prairie Island Nuclear Generating Plant. The Director also stated that the NRC Staff would issue a Director's Decision addressing Petitioners' concerns within a reasonable time.

PICANS submitted a letter to the Chairman of the NRC dated June 21, 1995, which reiterated the concerns raised in the petition and requested an evening public hearing within the vicinity of the Prairie Island facility. In a July 12, 1995 response, the NRC Staff informed PICANS that an evening public hearing was not warranted at that time but that the request would again be considered at the time of issuance of the Director's Decision.[1] PICANS was further informed that the concerns raised in the June 21, 1995 letter would be addressed in the Director's Decision.

On February 19, 1996, Petitioners filed an addendum to their petition raising further concerns regarding steam generator tube cracking and requested that Prairie Island Unit 1 not be allowed to return to operation until certain inspections of steam generator tubes were conducted. In a March 1, 1996 response, the Director of NRR denied Petitioners' request for action concluding that no safety issues warranting immediate action had been identified.

On March 13, 1996, Petitioners submitted another addendum to the petition raising additional concerns regarding steam generator tube cracking at Prairie Island and again requesting that the NRC require that Prairie Island Units 1 and 2 be placed in mid-cycle outages for the purpose of steam generator tube inspections. Petitioners further requested an informal public hearing if the NRC determined that such testing need not be conducted.

In an August 21, 1996 response, the Director of NRR concluded that the addendum did not raise any safety issues warranting immediate action and that an informal public hearing was not warranted at that time.

Petitioners' concerns are addressed below. In addressing these issues, I have considered the concerns expressed by the Petitioners in the letters of June 21, 1995, February 19, 1996, and March 13, 1996.

III. DISCUSSION

A. Steam Generator Tube Degradation

The steam generators used at pressurized water reactors (PWRs) are large heat exchangers that use the heat from the primary reactor coolant to make steam in the secondary side to drive turbine generators which generate electricity. The primary reactor coolant flows through tubes contained within the steam generator. As the coolant passes through the steam generator tubes, it heats the water (i.e., secondary coolant) on the outside of the tubes and converts it to steam which drives the turbine generators. Steam generator tubes made from mill-annealed alloy 600 have exhibited a wide variety of degradation mechanisms. Such material has been used in a number of steam generators at commercial nuclear facilities, including the steam generators at Prairie Island Units 1 and 2. These degradation mechanisms include mechanically induced (e.g., fretting wear, fatigue) and corrosion-induced (e.g., pitting, wastage, and cracking) degradation.

[1] For the reasons set out in the cover letter transmitting this Decision, the NRC Staff has again determined that an evening public hearing is not warranted.

Steam generator tubes constitute a significant portion of the reactor coolant pressure boundary. As a result, the structural and leakage integrity of the boundary is important in ensuring the safe operation of the plant. A loss of steam generator tube integrity has potential safety implications, as noted by the Petitioners, namely, (1) the loss of primary coolant which is needed to cool the reactor core and (2) the potential for leakage of radioactive fission products into the secondary system where their isolation from the environment cannot be ensured. As a result of the importance of this portion of the reactor coolant pressure boundary, NRC has regulations on maintaining the structural and leakage integrity of the steam generator tubes. The overall regulatory approach to ensuring that steam generators can be safely operated consists of the following:

(1) Technical specification requirements to ensure that the likelihood of steam generator tube rupture events is minimized, including
  (a) periodic inservice inspection of the tubing,
  (b) plugging or repair of tubing found by inspection to be defective, and
  (c) operational limits on primary-to-secondary leakage beyond which the plant must be shut down.

(2) Analysis of the design-basis steam generator tube rupture event to demonstrate that the radiological consequences meet 10 C.F.R. Part 100 guidelines.

(3) Emergency operating procedures for ensuring that steam generator tube rupture events can be successfully mitigated.

Steam generator tube degradation can be detected through inservice inspection of the steam generator tubes. These inspections are generally required by a plant's Technical Specifications which specify the frequency and scope of the examinations along with the tube repair criteria. In the 1970s, wastage (i.e., general tube wall thinning) and denting (mechanical deformation of the tube) were the dominant degradation mechanisms being observed. These degradation mechanisms were readily detectable with the bobbin coil inspection method and were effectively controlled or eliminated, in part, by improvements in water chemistry. Stress-corrosion cracking (SCC) emerged in the mid-1980s as the dominant degradation mechanism affecting the steam generator tubes. SCC can be oriented axially along the tube or circumferentially around the tube, or can consist of a combination of axial and circumferentially oriented cracks. SCC that has an axial orientation can be detected with a bobbin coil probe. The capabilities of the bobbin coil inspection method at detecting axially oriented cracks depend on such factors as the location of the cracking, interfering signals, and the data analysis procedures.

Circumferentially oriented SCC emerged as a significant problem affecting the industry in the late 1980s. The bobbin coil probe is generally insensitive to such cracking (i.e., circumferential SCC); as a result, locations susceptible to circumferential SCC may need to be examined with techniques other than the bobbin coil. Historically, probes such as the motorized rotating pancake coil (MRPC) probe have been used to detect circumferential SCC at locations susceptible to such degradation. Recently, more advanced probes (e.g., Zetec Plus-Point probe which contains a plus-point coil) have been used.

Deficiencies have been identified in certain utility inspection programs for detecting SCC, particularly circumferentially oriented SCC. Potential deficiencies include using inappropriate probes for inspecting locations susceptible to circumferential cracking, not optimizing the test methods to minimize electrical noise and signal interference, and not being alert to plant-unique circumstances (e.g., dents, copper deposits) which may necessitate special test procedures found unnecessary at other similarly designed steam generators or not included as part of a generic technique qualification.

Even though deficiencies in eddy-current inspection programs have been identified, operating experience indicates that steam generator tube integrity can be maintained at a plant when appropriate eddy-current data acquisition (including probe selection) and data analysis procedures are used, when the data analysts have been properly trained, when the intervals between inspections are determined based on the inspection findings, and when the operating environment of the steam generator tubes is controlled (e.g., water chemistry control). Adequate tube integrity has historically been achieved at plants through inservice inspections that involved the use of bobbin and MRPC probes. In some instances, operating intervals were shortened between inspections to ensure tube integrity.

Nevertheless, inspection findings at the Maine Yankee Atomic Power Station in 1994 and 1995 raised concerns that large circumferential cracks could develop over the course of an operating interval or that a large number of circumferential cracks may be present if a facility was not using appropriate inspection techniques. As a result of these inspection findings, the NRC Staff issued Generic Letter (GL) 95-03, "Circumferential Cracking of Steam Generator Tubes," on April 28, 1995, which (1) requested affected licensees to evaluate recent experience (including the Maine Yankee experience) concerning the detection and sizing of circumferential cracks and the potential applicability of this experience to their plants; (2) on the basis of the results of this evaluation, including past inspections and the results thereof, and other relevant factors, requested affected licensees to develop a safety assessment justifying continued operation until the next scheduled steam generator tube inspections were performed at their plants; and (3) requested that licensees develop and submit their plans for the next steam generator tube inspection as they pertain to the detection of circumferential cracks.

Subsequent to the issuance of GL 95-03, the Petitioners made the following requests with respect to steam generator tubes at Prairie Island Units 1 and 2:

Request (a) —

> That all steam generator tubes in Prairie Island Unit 2 be given a full length inspection utilizing the more comprehensive and proactive battery of tests employed at Maine Yankee during NSP's 1995 outage. Petitioners specifically demand that the Zetec Plus Point Probe and any state of the art, eddy current probe for corrosive cracking be employed at Prairie Island 2 during Outage 17 scheduled to end June 15, 1995.

Request (b) —

> That if the Zetec Plus Point Probe and any state of the art probe are not employed during the mid-June 1995 outage, then reactor Unit 2 be taken immediately off-line until such time these specific Zetec Plus Point Probe and any state of the art, eddy current probe for corrosion cracking are completed.

Request (c) —

> That Prairie Island Unit 1 immediately be placed into a mid-cycle outage to perform the NRC requested actions outlined in Generic Letter 95-03. In addition, all Unit 1 steam generator tubes be inspected through the use of the Zetec Plus Point Probe and any state of the art, eddy current probe for corrosion cracking.

NSP submitted its response to the generic letter for Prairie Island Units 1 and 2 by letter dated June 27, 1995. As discussed below, the information submitted provides no indication of an active circumferential crack mechanism at the Prairie Island units, nor does it suggest any significant concern regarding the potential for large, undetected circumferential cracks at these units.

The Prairie Island Unit 2 steam generators were last inspected in June 1995. This inspection included a 100%, full-length inspection with the bobbin probe. In addition, a 100% inspection was performed with a combined MRPC/Plus-Point probe from the hot-leg tube end to 3 inches above the tubesheet. Most row 1 and 2 U-bends were also inspected with the MRPC/Plus-Point coil. The bobbin probe is appropriate for performing the general-purpose, full-length inspection of the tubing because of its capability to detect flaw geometries exhibiting an axial component (e.g., corrosion thinning and wastage, mechanically induced wear, pitting, and axial cracks). The bobbin inspection was supplemented by inspections with a combined MRPC/Plus-Point probe to provide enhanced sensitivity to detecting cracks. These inspections encompassed the areas of axial crack activity with the bobbin coil probe and, in addition, the locations most vulnerable to circumferential cracking with the MRPC/Plus-Point coil.

NSP reports that the Prairie Island Unit 1 steam generators were last inspected in January 1996. This inspection included a 100% full-length inspection with the bobbin probe, except for rows 1 and 2 U-bends. Rows 1 and 2 U-bends were examined with MRPC/Plus-Point. All hot-leg tubes were examined with rotating probe technology (including Plus-Point) from the tube end to 6 inches above the top of the tubesheet. All sleeves were examined full length with the Plus-Point rotating coil.

In addition, NSP's response to the generic letter addressed, in part, each of five locations at which circumferentially oriented degradation has historically occurred in Westinghouse steam generators. These locations are places where there is significant axial stress associated with variations in tube geometry and include (1) tube expansion transition areas, (2) dented top-of-tubesheet locations in partial roll-expanded tubes (described below), (3) dented tube-to-tube support plate intersections, (4) small-radius U-bends, and (5) sleeve joints. Significant axial stress would contribute to the development of circumferential cracking.

Regarding the first and second categories, the tubes at Prairie Island are roll expanded over only the lower portion of the tubesheet depth (i.e., partial roll expansion). NSP reports that the incidence of circumferential cracks at expansion transitions where the tubes have received a partial-depth expansion has been negligible industrywide. For Prairie Island Unit 1, the 100% MRPC/Plus-Point inspection in the tubesheet regions in January 1996 did not find any circumferential indications in the in-service tubes. Similarly, for Prairie Island Unit 2, the MRPC/Plus-Point inspections in the tubesheet regions did not identify circumferential indications.

With regard to the third category, circumferential SCC at dented tube support plate intersections has only been reported at a limited number of plants. In addition, dented regions have exhibited both axial and circumferential SCC with axial SCC typically being the more frequently observed degradation mechanism. Axial SCC at dented locations can be detected with the bobbin probe. Although NSP has not reported performing MRPC or Plus-Point examination at the support plates, it has examined 100% of these locations using a bobbin probe and has not reported any axial cracking. Not detecting any axial cracking gives confidence that widespread circumferential SCC is not occurring.

Regarding the fourth category, SCC in the small-radius (row 1 and some row 2) U-bends has been extensive in Westinghouse steam generators. This cracking has been predominantly axial, with only isolated instances of nonaxial cracks. NSP reports that the small-radius U-bends are routinely inspected with the MRPC. In January 1996, the Licensee inspected 100% of rows 1 and 2 U-bends on Prairie Island Unit 1 with the MRPC/Plus-Point and found no indications. The June 1995 inspections at Prairie Island Unit 2 with the MRPC/Plus-Point probe looked at the majority of small-radius U-bends and found one axial and no circumferential indications.

Regarding the fifth category, during the January 1996 inspection in Unit 1, all in-service and new sleeves were examined full length with Plus-Point.

Indications were found in the upper sleeve weld region of sixty-one ABB Combustion Engineering welded tubesheet sleeves. These indications were characterized as single or multiple circumferential indications or volumetric indications. All of these sleeved tubes with circumferential indications were removed from service by sample removal and/or plugging. The volumetric indications were evaluated and indications located within the pressure boundary were plugged. No sleeves are installed in Unit 2. Sleeves were installed in Unit 1 to address forms of tube degradation (e.g., axial cracking and intergranular attack) other than circumferential cracking.

In response to the large number of indications identified in the upper sleeve welds of ABB Combustion Engineering welded tubesheet sleeves during the January 1996 Unit 1 outage, the NRC Staff held discussions and meetings with the Licensee to determine the root cause of the indications. NSP pulled five sleeve/tube samples during the outage to perform metallurgical analysis on and determine the root cause of the indications. Four of the removed tubes contained circumferential indications and one contained a volumetric indication. NSP started up Unit 1 on March 3, 1996, and committed to perform a mid-cycle outage to perform additional inspections unless the results of the metallurgical analyses from the pulled sleeves indicated that additional inspections would not be warranted.

ABB Combustion Engineering performed the metallurgical examinations, with third-party review by the Electric Power Research Institute. The results showed that the sleeve weld indications were not service induced. Instead, they were original fabrication flaws that were the result of faulty cleaning of tube surfaces prior to welding. The examinations of the tube samples revealed that the sizes of the flaws were such that the structural integrity of the welds was not compromised. None of the flaws showed any indication of having propagated in service. Since the indications were not service induced, the NRC Staff agreed that a mid-cycle outage to perform further inspections was not necessary.

ABB Combustion Engineering is currently revising its topical report on sleeving to incorporate improved cleaning techniques prior to installation of sleeves, in order to prevent such flaws in the future. NSP plans to submit an amendment to the NRC for review to adopt the revised ABB Combustion Engineering topical report prior to installation of CE sleeves.

After GL 95-03 was issued, additional information from inspections performed at Maine Yankee and the destructive examination of several tubes removed from Maine Yankee became available. This additional information appears in NRC Information Notice 95-40, "Supplemental Information Pertaining to Generic Letter 95-03, 'Circumferential Cracking of Steam Generator Tubes.' " This information led to the conclusion that the tubes with the largest indications at Maine Yankee continued to exhibit adequate structural integrity at the time they were found. This was attributable, in part, to the crack morphology as discussed in the Information Notice. As a result, adequate tube structural integrity was ensured for the operating interval between inspections, even though the MRPC probe, rather than the Plus-Point probe, was used during the earlier inspections.

As mentioned above, the safe operation of the steam generators is ensured by performing inspections and repairing defective tubes, limiting the operational leakage through the steam generators, analyzing a design-basis steam generator tube rupture event to demonstrate acceptable radiological consequences, and having appropriate emergency operating procedures in place. As discussed above, the Staff believes that the inspection probes used during the May 1994 and June 1995 outages at Prairie Island Units 1 and 2, respectively, were adequate to provide reasonable assurance of tube integrity. In addition, NRC requires an operational leak rate limit to provide reasonable assurance that, should a leak occur during service, it will be detected and the plant will be shut down in a timely manner before rupture occurs and with no undue risk to public health or safety.

Therefore, on the basis of (1) the fact that appropriate steam generator tube inspections have been performed, (2) monitoring of primary-to-secondary leakage is being conducted, and (3) the fact that appropriate emergency operating procedures are in place, the NRC Staff has concluded that the Petitioners' request for the shutdown of Prairie Island Units 1 and 2 until full-length tube inspections are completed using the Zetec Plus-Point probe and any state-of-the-art eddy-current probe should be denied.

B. Vessel Head Penetration (VHP) Cracking

The Petitioners contend that the VHPs at Prairie Island Units 1 and 2 are likely to have stress-corrosion cracks which, if not found and corrected, may result in a catastrophic accident involving reactor control rods. The Petitioners also contend that VHPs in PWRs in France, Belgium, Switzerland, and Sweden are cracking and that French data indicate that the cracking mechanism will not necessarily produce a detectable leak prior to a break that would initiate a serious accident. The Petitioners further contend that failure of a VHP could cause the ejection of a control-rod drive mechanism (CRDM), resulting in a loss of control of the reactor and/or a serious leak that could not be isolated and thereby could induce a loss-of-coolant accident. The Petitioners request immediate, full inspection of all VHPs in Units 1 and 2 for cracking using state-of-the-art eddy-current testing. The Petitioners also request that NRC immediately suspend the operating licenses of both units until the VHPs are inspected.

This same issue has been the subject of a recent Director's decision under 10 C.F.R. § 2.206 issued by the Director of NRR. *See All Pressurized Water Reactors,* DD-95-2, 41 NRC 55 (1995). There, the NRC Staff concluded, after reviewing the information referred to by that Petitioner, that the likelihood of the formation of circumferential cracks is small, the likelihood of forming small axial cracks is higher, and that leaks would develop before catastrophic failure of a VHP would occur. This would result in the deposition of boric acid crystals on the vessel head and surrounding area that would be detected during surveillance walkdowns. The Petitioners contend that this conclusion is not supportable as French data indicate that the cracking mechanism will not necessarily produce a detectable leak prior to a break that would initiate a serious accident.

The NRC Staff's review of the French data does not support the Petitioners' contention that a crack would not be detected due to leakage prior to catastrophic failure. Topical reports submitted to and reviewed by the NRC Staff indicate that cracks in the CRDM VHPs would need to grow well above the reactor vessel head before reaching a critical size that would lead to the catastrophic failure of a CRDM VHP. The portion of the crack above the head would leak well before the critical size is reached.

The circumferential crack at the French reactor was very small relative to the size flaw that would jeopardize structural integrity. Furthermore, the circumferential crack initiated from the exterior of the VHP which is more susceptible to circumferential cracking. This situation occurred after a small axial throughwall crack leaked. Thus, it is expected that leakage would be detected long before significant circumferential cracking could occur. Of the numerous VHP inspections in Europe, Japan, and the United States, no additional cases of circumferential cracking have been observed. The members of the Westinghouse, Babcock & Wilcox, and Combustion Engineering Owners Groups through Nuclear Energy Institute submitted acceptance criteria for both axial and circumferential cracking to the NRC for review and approval. The acceptance criteria were partially accepted by the NRC Staff. The criteria for axial cracking were accepted as proposed. The criteria for circumferential cracking were rejected. Any circumferential cracks found must be reported to the NRC Staff for disposition. If VHP cracking violated the above acceptance criteria, the NRC Staff would review the Licensee's plan for monitoring or repair of the crack.

Finally, a foreign reactor developed extensive circumferential cracking in VHPs as a result of two major demineralizer resin ingress events in the early 1980s. The NRC Staff issued a request for additional information to NSP on September 25, 1995, to determine if any similar resin ingress events had occurred at Prairie Island. The Licensee responded to the NRC Staff on October 24, 1995, that there have been no resin ingress events at Prairie Island.

The NRC Staff has closely monitored VHP cracking experience in the U.S. and abroad and has reviewed extensive evaluations of VHP cracking. The evaluations and operating experience indicate that it is highly unlikely that significant circumferential cracks could develop and that there is significant margin between the flaw sizes that would result in detectable leakage and the flaw sizes that would jeopardize structural integrity. Thus, the Staff has concluded that VHP cracking is not a safety concern at this time. To ensure that VHP cracking continues to be properly monitored and controlled, the NRC is in the process of preparing a Generic Letter requesting addressees to describe their program for ensuring the timely inspection of PWR CRDM VHPs and other VHPs. This letter was issued for public comment on August 1, 1996.

Accordingly, the requests made by the Petitioners for the shutdown of the Prairie Island units and inspection of the VHPs with enhanced inspection techniques is denied. As explained above, the NRC Staff has concluded that no substantial health and safety issues have been raised by the Petitioners.

C. Unloading of Dry Cask Storage Units

Spent fuel discharged from a reactor core is stored on site in a spent fuel pool prior to transfer to the U.S. Department of Energy (DOE) for final deposition. Typically, one-third of a reactor core is discharged every refueling outage (approximately every 18 months in the case of each of the Prairie Island units). The Licensee concluded several years ago that it would reach maximum capacity in its spent fuel pool in 1994, prior to availability of a DOE repository for storage of spent fuel. To support the need for continued storage of spent fuel at the reactor site, the Licensee applied to NRC for a license to store spent fuel in an onsite independent spent fuel storage installation (ISFSI). NRC issued Materials License No. SNM-2506 to NSP on October 19, 1993, for receipt and storage of spent fuel at the ISFSI on the site of the Prairie Island Nuclear Generating Plant. Materials License No. SNM-2506 allows NSP to use the TN-40-type casks for storage at its ISFSI. The TN-40, a metal cask system, is designed to store forty PWR spent fuel assemblies in each cask. Dimensions of the cask (with protective cover) are 202 inches high with an outside diameter of 103.5 inches. A loaded TN-40 storage cask weighs 109.3 metric tons.

On April 28, 1995, a public meeting was held in Red Wing, Minnesota, to present NRC inspection findings related to dry cask storage activities at the Prairie Island plant. Questions were raised by members of the public as to how the Licensee would unload the spent fuel in a dry storage cask, if it became necessary, i.e., would there be enough empty fuel racks in the spent fuel pool to accommodate unloading of the cask.

In a letter to the NRC dated May 3, 1995, the Licensee submitted a plan for unloading the TN-40 cask in response to the questions raised at the April 28, 1995 meeting. In that letter, the Licensee stated that some of the fuel racks in the spent fuel pool contain nonfuel-bearing components, which could be relocated to a temporary location in the fuel transfer canal. Alternatively, it may be possible for the components to be stored temporarily in the TN-40 cask, should it become necessary to unload a cask. In the latter case, even though the TN-40 cask being returned to the spent fuel pool may no longer be qualified to hold spent fuel, it quite possibly could still safely hold irradiated nonfuel-bearing components.

The Petitioners raised issues concerning compliance with 10 C.F.R. § 50.59 and the need to make changes to Technical Specifications in order to use the fuel transfer canal for nonfuel-bearing components under the Licensee's plan. Petitioners also stated that section 50.59 requires a safety analysis and amendment to the operating license with a public hearing whenever a change occurs in Technical Specifications for spent fuel pool and reactor transfer canal use. Petitioners further stated that a safety analysis is essential when a Technical Specification change occurs.

The need for a change to the Technical Specifications and the process to be followed under section 50.59 are two separate, but related, issues. With regard to the Prairie Island Technical Specifications, the plan proposed by the Licensee in its letter of May 3, 1995, for dealing with the need to unload a cask, would not involve a change to Technical Specifications because Technical Specifications do not address use of the fuel transfer canal nor do they address movement of nonfuel-bearing components within the spent fuel pool. Prairie Island's Technical Specification 3.8 specifies operating limitations associated with fuel-handling operations and core alterations only. Further, the fuel transfer canal is not classified as a reactor safety system. The fuel transfer canal provides no protection for the reactor, nor does it mitigate the consequences of a postulated accident to the reactor. The fuel transfer canal is a component of the fuel storage and fuel handling systems, which is considered a plant auxiliary system rather than a reactor safety system. As use of the fuel transfer canal in the Licensee's plan does not involve a change to the Technical Specifications, an amendment for this reason would not be required and the opportunity to request a public hearing with regard to a Technical Specification change would, therefore, not arise.

With regard to section 50.59 of Title 10 of the *Code of Federal Regulations,* that provision allows a Licensee to make changes to its facility and procedures as described in the Final Safety Analysis Report (FSAR) without prior approval from NRC, provided a change in Technical Specifications is not involved (which, as described above, is met in this instance) and an unreviewed safety question does not exist. Before moving the nonfuel-bearing components to temporary storage racks in its fuel transfer canal, NSP would need to determine if this use of the transfer canal changes the facility or procedures as described in the FSAR. If NSP determines that a change has been made to the facility or procedures as described in the FSAR, then a safety evaluation pursuant to section 50.59 is required to be performed by the Licensee. If a Technical Specification change were needed (not the case as discussed above), or an unreviewed safety question existed, NRC review and approval would be required. Otherwise, the Licensee could make the modifications without prior NRC approval. Licensees submit a list of modifications that were performed under section 50.59 without NRC approval to NRC annually.

The Licensee did not fail to comply with the requirements of section 50.59 by presenting a plan for retrieval of fuel from a cask, which included an option to place nonfuel-bearing components in the fuel transfer canal. At the time a cask unloading is deemed necessary, the Licensee can evaluate the specific modifications needed to implement the plan and determine whether section 50.59 is applicable.

When applying for the license, NSP performed an accident analysis, in its Safety Analysis Report, as required by NRC regulations.[2] In its Safety Evaluation Report dated July 1993, the NRC Staff reviewed the Licensee's accident analysis and determined that "Dose equivalent consequences, from a single cask, to any individual, from direct and indirect radiation and gaseous activity release after postulated accident events, are less than the 50 mSv (5 rem) limit established in 10 CFR 72.106(b)." Additionally, in its Environmental Assessment, dated July 28, 1992, the NRC Staff assessed the accident dose at the Prairie Island site boundary as: "a small fraction . . . of the criteria specified . . . , " and found that: "These doses are also much less than the Protective Action Guides established by the Environmental Protection Agency (EPA) for individuals exposed to radiation as a result of accidents . . . ." Because it has been shown that the dose equivalent from a single cask to any individual from postulated accident events is not in excess of the levels required for taking protective actions to protect public health, the NRC Staff considers that a time-urgent unloading of the TN-40 cask is not a likely event.

Even if such an unlikely accident occurred and the Licensee determines that corrective actions may need to be taken to maintain safe storage conditions, options are available. This may include returning the cask to the auxiliary building and/or the spent fuel pool for repairs. Once the cask is in the spent fuel pool, it does not necessarily have to be unloaded to maintain safe storage conditions. In addition, the Licensee may have other options available to cover this unlikely contingency, including temporary storage of spent fuel in a spare storage cask or use of an existing certified transportation cask. The Licensee would have time to consider these and other available options in such an unlikely event.

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[2] The Licensee analyzed accidents classified as Design Events III and IV, as described in ANSI/ANS 57.9, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)." Design Event III consists of that set of infrequent events that could reasonably be expected to occur during the lifetime of the ISFSI. Design Event IV consists of the events that are postulated because their consequences may result in the maximum potential impact on the immediate environs. Included among the scenarios considered under Design Event IV was a loss of confinement barrier leading to an immediate release of radioactivity.

Petitioners also raise an issue concerning the necessity to offload both the entire reactor core and a TN-40 cask simultaneously. NRC has no requirement for licensees to maintain the spent fuel capacity to offload the entire core at once. Prairie Island normally offloads only one-third of the core during refueling outages. If NSP determines the need to offload the entire core during a refueling outage, NSP can install temporary fuel racks in the cask laydown area in the spent fuel pool. Therefore, a cask could not be unloaded for the short time that temporary racks are installed in the cask laydown area. The Staff does not view this as a problem for two reasons. First, the probability that a cask would require unloading at the same time a full-core offload is in process is extremely small. Second, in the event it became necessary to unload a cask, fuel could be placed back into the reactor vessel and the temporary fuel storage racks could be removed. As discussed above, time-urgent unloading of a TN-40 cask is extremely unlikely. The cask could then be unloaded after the cask laydown area was cleared of the temporary fuel storage racks.

In addition to ensuring that a TN-40 cask could be unloaded if necessary, the Licensee's plan also provides assurance with regard to spent fuel retrievability. Subpart F of 10 C.F.R. Part 72 provides general design criteria for ISFSIs and monitored retrievable storage installations. Section 72.122 sets overall requirements and 10 C.F.R. § 72.122(l) provides for retrievability of the fuel and states: "Storage systems must be designed to allow ready retrieval of spent fuel or high-level radioactive waste for further processing or disposal." The NRC Staff concluded in a May 5, 1995 letter to the Licensee that the ability to unload a TN-40 cask if necessary in accordance with the Licensee's plan would satisfy this fuel retrievability provision.

Finally, Petitioners state that the wrong NRC department reviewed and approved NSP's plan for retrievability of irradiated fuel. The Office of Nuclear Material Safety and Safeguards (NMSS) is responsible for licensing and regulating all issues under Part 72, including issues related to the design requirements for ISFSIs. Therefore, NMSS is the correct NRC office to review whether the Licensee's plan met section 72.122(l). As discussed above, the Licensee's plan does not involve a Technical Specification change. Accordingly, NRR review of such a change would not be required. If, upon implementing its plan, the Licensee determined that a safety evaluation pursuant to section 50.59 was required, NRR review and approval would be required only if an unreviewed safety question existed.

With regard to the requests made by the Petitioners, there is no basis for suspending NSP's operating licenses for the Prairie Island units until a safety analysis is completed, reviewed, and approved by NRC, and until NSP's licenses are amended and public hearings have been held. If NSP plans to implement a specific plan to utilize the fuel-transfer canal which changes the facility or procedures as described in the FSAR, then an evaluation pursuant to section

50.59 would be required at that time, which would not require prior NRC approval unless an unreviewed safety question exists or a change to Technical Specifications is required.

D. Auxiliary Building Crane

Petitioners contend that a recent incident at Prairie Island on May 13, 1995, involving the crane used to lift the dry cask for Prairie Island's ISFSI, requires physical testing and safety analysis before future crane use. The incident resulted in the crane holding the 123.75-ton cask above the surface of the reactor pool for 16 hours. The Petitioners assert that the incident could have caused metal fatigue within the crane's structure and the cables attached to the crane. Also, Petitioner Prairie Island Coalition asserts in its June 21, 1995 letter to the Chairman of the NRC that the crane, its cable, and its cable mechanisms were not designed to withstand holding nearly a maximum load for 16 hours.

The Prairie Island auxiliary building crane was upgraded in 1992 in accordance with the provisions of Topical Report EDR-1(P), "Ederer Nuclear Safety-Related Extra Safety and Monitoring (X-SAM) Cranes." The crane is designed and tested in accordance with the NRC Staff's guidance as outlined in NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

The Staff evaluated the design of the auxiliary building crane and the lifting device for the cask as part of its review of the dry cask ISFSI. This crane system is designed so that a single failure will not result in the loss of the capability of the system to safely retain the load (this design is known as single-failure proof). The crane is designed to handle a rated load of 125 tons and is capable of raising, lowering, and transporting occasional loads, for testing purposes, of 25% higher than the rated load without damage or distortion to any crane part. All parts of the crane that are subjected to dynamic strains, such as gears, shafts, drums, blocks, and other integral parts, have a safety factor of five (i.e., they are designed to lift five times the design-rated load). The hook has a design safety factor of ten and was subjected to a 200% overload test followed by magnetic particle inspection prior to initial operation. Protection against wire rope wear and fatigue damage are ensured by scheduled inspection and maintenance. The special lifting device used for cask movement is designed to support six times the weight of the fully loaded cask and was subjected to a 300% overload test by the manufacturer. The lifting device undergoes dimensional testing, visual inspection, and nondestructive testing every 12 months (plus or minus 25%).

A single-failure-proof crane, such as the crane at Prairie Island, that has become immobilized by failure of components while holding a load, is able to hold the load or set the load down while adjustments or repairs are made. Safety features and emergency devices permit manual operation to accomplish this task. Two separate magnetic brakes are provided as well as an emergency drum band brake. Each magnetic brake provides a braking force of at least 150% of rated load. The emergency drum brake ensures that the load can be safely lowered even if power is lost to the crane. Because of the large design margins and the ability to withstand a failure of any single component, the NRC Staff does not postulate a load drop from a single-failure-proof crane.

After the incident on May 13, 1995, the Licensee temporarily removed the crane from service for testing. The Licensee and the crane vendor performed testing on the crane to analyze the event and ensure that the crane was operable. The Licensee's analysis of the May 13, 1995 incident found the problem to be an improperly calibrated load cell (a load cell is a device that measures the load being lifted by the crane and provides input to an overload-sensing device). It was determined that the actual load was less than what was being sensed by the overload-sensing device. The function of the overload-sensing device is to stop the operation of the crane when the load reaches a predetermined value. This prevents loading the crane beyond its rated load by maintaining loads within the design working limit, thereby maintaining safety and the physical integrity of the crane system.

Since the design-rated load of the crane was not exceeded during the incident, there is no reason to assume that the crane cannot continue to operate safely. Even if the rated load had been exceeded, an analysis would be needed to determine how much the rated load was exceeded and if that amount is significant. When cranes are built, manufacturers conduct proof tests at a load above rated load. The proof test for this crane was 25% higher than the 125-ton design-rated load for the main hoist (i.e., the proof test was 156.25 tons).

With regard to the Petitioners' comment about metal fatigue, metal fatigue is a condition that results from cyclic stress. Cyclic stress is produced by repeated loading and unloading. The crane is designed to handle all loading and unloading cycles during the life of the plant, including construction and operating periods. A single static (constant) load such as the load in question, does not produce the cyclic stress that causes metal fatigue. The Petitioners' contention that it was never contemplated that the Prairie Island polar crane hold a load of 123.75 tons inches above the surface of the reactor pool for 16 hours is incorrect. The contemplated failure mechanism of a single-failure proof crane is to hold the load safely at any location until the load can be safely moved. Because of the large design margins, the length of time that a design-rated load (or a load less than design rated) is on the hook of a single-failure-proof crane is inconsequential.

With regard to cable and cable mechanisms (also known as the reeving system and lifting devices), the crane is provided with a balanced dual reeving system with each wire rope capable of supporting the maximum critical load (if a load being held by a crane can be a direct or indirect cause of release of radioactivity, the load is called a critical load). The hydraulic load-equalizing system allows transfer of the load to the remaining rope, without overstressing it, in the event of a failure of one rope. Protection against wire rope wear and fatigue damage are ensured by scheduled inspection and maintenance.

In conclusion, NRC agrees with the Licensee in its determination that the cause of the incident was an incorrectly calibrated load cell. This cause was documented in NRC Inspection Report 95-006, issued June 27, 1995. NRC has determined that the Licensee met the design and testing requirements established in industry standards for the control of heavy loads such as a dry storage cask, that the overload-sensing device worked as designed, and that no safety issue was involved in the Licensee's use of the auxiliary building crane and associated cask handling equipment to move the cask. Therefore, the Petitioners' requests for suspension of NSP's licenses for the Prairie Island units until physical testing and safety analyses can be performed on the crane are denied.

IV. CONCLUSION

Petitioners requested an immediate suspension of NSP's licenses for Prairie Island Units 1 and 2 until corrective actions of potentially hazardous conditions would be taken by NSP and NRC with regard to issues identified in the petition. The institution of a proceeding in response to a request for action under section 2.206 is appropriate only when substantial health and safety issues have been raised. *See Consolidated Edison Co. of New York* (Indian Point, Units 1, 2, and 3), CLI-75-8, 2 NRC 173, 176 (1975), and *Washington Public Power Supply System* (WPPSS Nuclear Project No. 2), DD-84-7, 19 NRC 899, 923 (1984). I have applied this standard to determine if any action is warranted in response to the matters raised by the Petitioners. Each of the claims by the Petitioners has been reviewed. The available information is sufficient to conclude that no substantial safety issue has been raised regarding the operation of Prairie Island Units 1 and 2. Therefore, I conclude that, for the reasons discussed above, no adequate basis exists for granting Petitioners' requests for immediate suspension of NSP's licenses for Prairie Island Units 1 and 2.

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 this regulation, 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

Frank J. Miraglia, Jr., Acting
Director
Office of Nuclear Reactor
Regulation

Dated at Rockville, Maryland,
this 27th day of November 1996.