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# NUCLEAR REGULATORY COMMISSION ISSUANCES

April 1997

UNITED STATES NUCLEAR REGULATORY COMMISSION

U.S. NUCLEAR REGULATORY COMMISSION

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Vol. 45, No. 4
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# NUCLEAR REGULATORY COMMISSION ISSUANCES

April 1997

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)

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

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Cite as 45 NRC 265 (1997) LBP-97-7

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**Charles Bechhoefer**, Chairman
**Dr. Jerry R. Kline**
**Dr. Peter S. Lam**

| | |
|---|---|
| **In the Matter of** | **Docket No. 50-160-Ren (ASLBP No. 95-704-01-Ren) (Renewal of Facility License No. R-97)** |
| **GEORGIA INSTITUTE OF TECHNOLOGY (Georgia Tech Research Reactor, Atlanta, Georgia)** | **April 3, 1997** |

The Licensing Board issues an Initial Decision that authorizes grant of a 20-year renewal of the operating license of the Georgia Tech Research Reactor.

## RULES OF PRACTICE: REQUIREMENTS OF DECISIONS

Merely because expert witnesses for all parties reach similar conclusions on an issue does not mean that the Licensing Board must reach the same conclusion. The significance of various facts is for the Board to determine, based on the record, and cannot be delegated to the expert witnesses of various parties, even if they all agree. The Board must satisfy itself that the conclusions reached have a solid foundation.

### LICENSING BOARDS: SCOPE OF REVIEW

A licensing board must do more than act as an "umpire blandly calling balls and strikes for adversaries appearing before it." *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), LBP-94-35, 40 NRC 180, 192 (1994), *citing Scenic Hudson Preservation Conference v. Federal Power Commission,* 354 F.2d 608, 620 (2d Cir. 1965).

### EVIDENCE: TESTIMONY OF GOVERNMENT OFFICIALS

Although the testimony of a public official working for a government agency may be entitled to a presumption (albeit rebuttable) that public officials are presumed to have performed their official duties in a proper manner, this presumption does not apply where the official is not operating in a traditional governmental capacity but rather as an official of a regulated entity operated by a government unit.

### RULES OF PRACTICE: STANDARD OF PROOF

Government entities have the same burdens in proving their cases in NRC licensing proceedings as private entities.

### MANAGEMENT ORGANIZATION: STRUCTURE

NRC regulations prescribe no particular managerial structure. The acceptability of a managerial organizational structure depends, in part, on the independence of operational and safety functions.

### MANAGEMENT ORGANIZATION: STRUCTURE

With respect to power reactors, interpretations of quality assurance requirements have led to mandatory separation of operational and safety functions. With respect to nonpower reactors, there is no regulatory requirement for any particular structure, and they vary considerably, so long as some form of independent safety review is maintained.

### MANAGEMENT ORGANIZATION: STRUCTURE

Where two forms of management organization are legally acceptable, a Licensing Board would need a strong record establishing the performance superiority of one (and safety deficiencies attributable to the other) to mandate a change.

## ENFORCEMENT ACTIONS: CRITERIA

A licensing board would only refuse to authorize a renewed license under the enforcement policy (i.e., based on violations) for reasons that were as serious as those that could lead to license revocation. Under NRC's enforcement policy, a series of Severity Level IV violations would not warrant license revocation.

## TECHNICAL ISSUES DISCUSSED

The following technical issue is discussed: Management organization.

## APPEARANCES

**Alfred L. Evans, Jr., Esq., Patricia Guilday, Esq., E. Gail Gunnells, Esq.,** and **Randy A. Nordin, Esq.,** Atlanta, Georgia, for Georgia Institute of Technology (Georgia Tech or Applicant).

**Ms. Glenn Carroll,** Decatur, Georgia, **Mr. Robert P. Johnson, III, Ms. Carol Stangler, Mr. Alvin Lenoir,** and **Ms. Danna Smith,** Atlanta, Georgia, for Georgians Against Nuclear Energy (GANE or Intervenor).

**Sherwin E. Turk, Esq., Colleen P. Woodhead, Esq.,** and **Susan S. Chidakel, Esq.,** for the NRC Staff.

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## INITIAL DECISION

This proceeding involves the application of Georgia Institute of Technology (hereinafter, Georgia Tech or Applicant) to renew its Facility License No. R-97 for the Georgia Tech Research Reactor (GTRR), also known as the Neely Nuclear Research Center (NNRC), located on the Georgia Tech campus in Atlanta, Georgia. Under the terms of the existing license, the GTRR is a heterogeneous, heavy-water moderated and cooled reactor authorized to operate at power levels up to 5 megawatts (thermal) for research and development activities. GT Exh. 19,[1] Staff Exh. 13. As set forth in the September 19, 1994 Notice of Opportunity for Hearing, 59 Fed. Reg. 49,088 (Sept. 26, 1994), the renewal would extend the expiration date of the license for 20 years, until June 6, 2014 (GT Exh. 19; Staff Exh. 13), in accordance with the Applicant's timely application for renewal dated April 19, 1994.[2]

For reasons set forth herein, we are approving the sought license renewal. We are also suggesting that Georgia Tech consider making certain changes in management organizational structure, although we are not imposing any formal conditions to this effect.

## A. Background

In response to a Notice of Opportunity for Hearing on the license-renewal application, published in the *Federal Register* of September 26, 1994 (59 Fed. Reg. 49,088), Georgians Against Nuclear Energy (hereinafter, GANE or Intervenor) on October 26, 1994, filed a timely petition for leave to intervene. This Licensing Board was established on November 18, 1994, to rule upon GANE's petition and preside over any evidentiary hearing that might result. 59 Fed. Reg. 60,849 (Nov. 28, 1994).

By our Memorandum and Order (Intervention Petition), dated November 23, 1994 (unpublished), we outlined applicable standards for both standing to intervene and contentions and, in accordance with 10 C.F.R. § 2.714(a)(3), established a date by which GANE could submit an amended petition. GANE's amended petition was timely filed on December 30, 1994. Georgia Tech and the NRC Staff each opposed GANE's supplemental petition, both as to standing and contentions.

We held a prehearing conference on January 31–February 2, 1995, in Atlanta, Georgia, to consider GANE's standing and its proposed contentions.[3] Following the conference, we issued a Prehearing Conference Order (Ruling on Standing and Contentions), LBP-95-6, 41 NRC 281 (1995). We determined that GANE had established its standing to participate and admitted two of its ten proposed contentions, one dealing with the adequacy of the Applicant's management and the other with physical security of the site during the 1996 Summer Olympic Games held in Atlanta, Georgia.

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[1] Georgia Tech Exhibits will be referenced as GT Exh. __.

[2] By virtue of its timely application for renewal, Georgia Tech in effect extended the expiration date of its current license until the Commission reaches a final determination on the renewal application. 10 C.F.R. § 2.109.

[3] *See* Notice of Prehearing Conference, dated January 12, 1995, published at 60 Fed. Reg. 3885 (Jan. 19, 1995).

The Applicant and Staff sought Commission review pursuant to 10 C.F.R. § 2.714a of our determination to grant GANE a hearing and admit two contentions. They each contested our admission of the two contentions, and the Applicant in addition challenged our finding of GANE's standing. During the course of that appeal, the Applicant, responding to several Commission inquiries relative to security at the Olympic Games, determined to remove all nuclear fuel from the site prior to the Olympic Games and not to replace it until after the Games. The Commission accordingly remanded the security contention to us for appropriate action (CLI-95-10, 42 NRC 1 (1995)), and we issued a Partial Initial Decision dismissing the contention as moot. LBP-95-19 (corrected), 42 NRC 191 (1995).

The Commission affirmed both our finding of GANE's standing and our admission of the management contention (Contention 9). CLI-95-12, 42 NRC 111 (1995). With respect to that contention, we held 13 days of evidentiary hearings, between May 20, 1996, and June 28, 1996 (Tr. 963-2552, 2614-3545).[4] With the agreement of all parties, the filing of proposed findings of fact and conclusions of law was delayed until after the conclusion of the Olympic Games. Proposed findings of fact and conclusions of law were filed by Georgia Tech, GANE, and the NRC Staff.[5] Reply findings and conclusions were thereafter filed by Georgia Tech.[6]

## B. Georgia Tech's Prefatory Comment

Georgia Tech initially takes the position that, based on the bottom-line positions of expert witnesses of all parties to the effect that the operation of the GTRR currently poses no undue risk to the health and safety of the public, no detailed findings of fact need be made by us. App. FOF at iii-xii. We disagree. The significance of various facts is for us to determine, based on the record, and cannot be delegated to the expert witnesses of the various parties, even if they all agree. We must satisfy ourselves that the conclusions expressed by expert witnesses on significant questions have a solid foundation. *Philadelphia Electric Co.* (Limerick Generating Station, Units 1 and 2), ALAB-819, 22 NRC

[4] In accordance with 10 C.F.R. § 2.715(a), we also heard oral limited appearance statements, once during the initial prehearing conference (Feb. 1, 1995) and twice during the hearing sessions (May 20 and 22, 1996).

[5] The Georgia Institute of Technology's Proposed Findings of Fact and Conclusions of Law, dated September 13, 1996 (App. FOF); Georgians Against Nuclear Energy Proposed Findings of Fact in Consideration of Application for Renewal of Facility License, dated October 11, 1996 (GANE FOF); NRC Staff's Proposed Findings of Fact and Conclusions of Law, dated October 25, 1996 (Staff FOF).

[6] The Georgia Institute of Technology's Reply to the Proposed Findings of Fact and Conclusions of Law of (1) GANE, and (2) The NRC Staff, dated November 8, 1996 (App. Reply FOF).

681, 741 (1985).[7] Moreover, the evidentiary record includes more than just expert witnesses' testimony. We must also assess the significance of information obtained from fact witnesses and documentary exhibits.

As another basis for not making detailed findings, Georgia Tech also has claimed that Dr. Ratib A. Karam, Director of the GTRR, is a public official working for a governmental agency and is entitled to a presumption (albeit rebuttable) that public officials are presumed to have performed their official duties in a proper manner. App. FOF, Prefatory Comment at xii, *citing United States v. Chemical Foundation, Inc.*, 272 U.S. 1, 14-15 (1926), and 31A C.J.S. *Evidence* § 146, at 318-22. This presumption does not apply where, as here, the government official is not operating in a traditional governmental capacity but rather as an official of a regulated entity operated by a governmental unit. Indeed, insofar as relevant here, government entities have the same burdens in proving their cases in NRC licensing proceedings as private entities. *See Tennessee Valley Authority* (Phipps Bend Nuclear Plant, Units 1 and 2), ALAB-506, 8 NRC 533, 544 (1978), establishing that no different regulatory standards would apply if the GTRR were operated by a private rather than a governmental entity.

We therefore reject Georgia Tech's suggestion that we need not make detailed findings on the many factual issues on which we took evidence. We turn now to our findings on the management contention, the single contention at issue.

### C. GANE'S Management Contention

GANE's Contention 9, as submitted in GANE's Amended Petition for Leave to Intervene, dated December 30, 1994, and as admitted by us in our April 26, 1995 Prehearing Conference Order (Ruling on Standing and Contentions), LBP-95-6, 41 NRC 281, reads as follows:

> *GANE contends that management problems at the GTRR are so great that safety for the public cannot be assured.* Safety concerns at the Georgia Tech reactor are the sole responsibility of Dr. R. A. Karam (SAR, Fig. 6.1, p. 157). Dr. Karam is the director who withheld information about a serious accident from the NRC (1987 cadmium-115 accident). The NRC was advised of the 1987 cadmium-115 accident by the safety officer at that time, who was later demoted, and left the GTRR operation claiming harassment. Since the incident, management has been restructured giving the director (Dr. Karam) increased authority, including increased authority over the Manager of the Office of Radiation Safety. Although the safety officer has line to higher-ups than the director, since he/she works for the director on a day-to-day basis, the threat of reprisal would be a huge disincentive to defying the director.

[7] Stated another way, a licensing board must do more than act as an "umpire blandly calling balls and strikes for adversaries appearing before it." *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), LBP-94-35, 40 NRC 180, 192 (1994), *citing Scenic Hudson Preservation Conference v. Federal Power Commission*, 354 F.2d 608, 620 (2d Cir. 1965).

The Nuclear Safeguards Committee which has theoretical oversight of the GTRR operations has a distinct flaw in having no concern with health issues. The Office of Radiation Safety Manager is sought for its knowledge of law more than its knowledge of health physics. (SAR, Sec. 6.1, p. 156-159).

During the course of the hearing, upon a demonstration of good cause for the delay, GANE added several other discrete items as examples of poor management.

The Applicant, Staff, and GANE each presented witnesses and each also relied on documentary evidence. We will identify these witnesses and the relevant documentary evidence in conjunction with our discussion of specific aspects of the contention.

### *1. Historical Record of Management*

In order for us properly to assess GANE's management contention, it is necessary to review the management deficiencies extending at least as far back as early 1987, upon which the contention is based, and the partial and complete shutdowns that occurred in 1987-1988. We will then examine the record of management after restart to determine whether, as GANE contends, substantial management deficiencies still persist (*see* LBP-95-6, 41 NRC at 299) or, as Georgia Tech and the Staff assert, the deficiencies have been adequately remedied.

#### *a. Management Record Leading to Shutdown*

##### (1) INSPECTION REPORT 87-01

Our review of the Applicant's managerial deficiencies that undergird GANE's contention must initially focus upon the NRC Staff's inspection findings in early 1987, as presented by the NRC Staff's Panel A, as well as by NRC Inspector Anne Rebecca Long, testifying on behalf of GANE.[8] As reflected by the current record, the earliest of those inspections citing management deficiencies was conducted by Inspector Long on February 9-23, 1987, and is documented in Inspection Report (IR) 50-160/87-01[9] (GANE Exh. 21).

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[8] Ms. Long was called as a witness by the Staff in response to GANE's request, as directed by this Licensing Board. The Board had determined, in accordance with 10 C.F.R. § 2.720(h)(2), that Ms. Long's "view of the facts . . . can reasonably be expected to differ significantly from views likely to be presented by the inspectors on NRC's witness panels." Third Prehearing Conference Order, LBP-96-8, 43 NRC 178, 181 (1996).

[9] Inspection reports (IRs) related to nuclear reactor licensees are generally issued in numerical sequence each year, designating the facility's docket number followed by the IR number. For simplicity, references in this opinion to NRC IRs will omit the GTRR docket number (50-160) from the IR number.

Inspector Long testified that, prior to this inspection, the NRC had received allegations concerning the GTRR (to the effect of an unreported power excursion and a report that the reactor had been running without a licensed operator at the controls) and she was instructed by her acting supervisor to include these allegations in her routine inspection but not to reveal to Georgia Tech that the allegations had been received. Tr. 1444, 1446, 1449-50, 1549 (Long). IR 87-01 concluded that the power excursion occurred but was not a violation (GANE Exh. 21, Report Details, at 27-29). The Staff referred the other allegation to Georgia Tech for investigation after determining there was a lack of evidence to pursue its own investigation. Tr. 1449-50 (Long); Staff Exh. 9.

Inspector Long documented six Severity Level IV violations[10] in IR 87-01, with numerous examples given for several of the violations, specifically: (1) failure to provide or utilize procedures (seven examples); (2) failure to control experiments as required by the Technical Specifications (TS) (four examples); (3) failure to perform a weekly heat balance surveillance; (4) failure to receive prior NRC approval for a change made to the facility's Technical Specifications; (5) failure to comply with the requalification program for annually documenting performance of operators under simulated emergency conditions for 1984, 1985, and 1986; and (6) failure of the Nuclear Safeguards Committee (NSC) to perform its review and audit functions as required (four exampl

Following Georgia Tech's responses dated May 25, 1987, and July 15, 1987, to the IR and Notice of Violations (NOVs), the Staff withdrew the last two violations and some examples of the others. Georgia Tech initiated corrective actions for the remaining violations. Staff Panel A, ff. Tr. 1740, at 9, 10-12; GANE Exh. 21, Enclosure 1 (Notice of Violation); GANE Exh. 23.[11]

Beyond the specific violations identified, the Staff advised Georgia Tech that it was "concerned about a programmatic weakness in implementation of Technical Specification requirements." GANE Exh. 21, Letter to Georgia Tech transmitting NOV and IR 87-01, at 1. The Staff testified that, "collectively, the violations provided substantial evidence of a lack of management oversight." Staff Panel A, ff. Tr. 1740, at 13.[12]

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[10] At the time, NRC categorized violations in Severity Levels I to V, as follows: Severity Level I and II violations are of very significant regulatory concern. In general, violations that are included in these severity categories involve actual or high potential impact on the public. Severity Level III violations are cause for significant concern. Severity Level IV violations are less serious but are of more than minor concern; i.e., if left uncorrected, they could lead to a more serious concern. Severity Level V violations are of minor safety or environmental concern. 10 C.F.R. Part 2, Appendix C (revised as of January 1, 1988); Staff Panel A, ff. Tr. 1740, at 12.

[11] Inspector Long would have preferred to escalate the six Level IV violations into more severe Level III violations, but she did not pursue the formal steps to appeal the classification and indicated that she was satisfied with IR 87-01. Tr. 1344-47, 1394-95 (Long).

[12] Reflecting the Staff's elevated level of concern, the cover letter was signed by the Director, Division of Reactor Projects, one level of management higher than normal. Staff Panel A, ff. Tr. 1740, at 13-14.

Inspector Long brought to the attention of Region II management (specifically, Mr. Albert F. Gibson, Director of the Division of Reactor Safety, Region II, from 1985 to the present, and Mr. Malcom Ernst, then Deputy Regional Administrator of Region II) her dissatisfaction with NRC's withdrawal of two of the violations and portions of two others set forth in IR 87-01. Tr. 1405, 1406-07, 1582 (Long). Mr. Gibson subsequently agreed that the violations should not have been withdrawn. But no further action in this regard was taken against Georgia Tech, inasmuch as, by that time, further inspections had been undertaken, an order modifying the GTRR license had been issued, and an enforcement conference with Georgia Tech had been scheduled. Staff Panel A, ff. Tr. 1740, at 13-14; Staff Exh. 19.[13]

### (2) INSPECTION REPORT 87-03

The next significant inspection, carried out on April 7-10, 1987, by a Radiation Specialist in the Emergency Preparedness and Radiological Protection Branch, produced many apparent violations, including a failure to label a container of radioactive material, failure to perform radiological surveys (two examples), failure to wear protective clothing as required by procedure (two examples), failure to wear required dosimetry, failure to implement Health Physics (HP) monitoring as required by a Radiation Work Permit, failure to obtain review and approval of experiments (two examples), failure to complete the Experimenter's Checklist as required by procedure (two examples), failure to respond to a criticality alarm, and failure to survey radiation levels during handling of a pneumatic transfer device containing an irradiated sample. Although the Applicant had itself discovered several of these failures, adequate corrective actions were not taken. *Id.* at 16; IR 87-03 (GANE Exh. 31).

Based on an unusually large number of apparent violations, the Staff held an enforcement conference on May 4, 1987, at which violations identified earlier that year in IRs 87-01 and 87-02 were also addressed. Staff Panel A, ff. Tr. 1740, at 16; Tr. 1764 (Collins); *see* GANE Exh. 31, at 1; Tr. 1529-30 (Long). At the enforcement conference, documented in IR 87-06 (GANE Exh. 30),[14] Georgia Tech outlined actions to improve management oversight and self-identification of problems, including a possible reorganization to place the radiation protection or health physics (HP) function under the authority and responsibility of the NNRC Director and the possible merger of the campus-wide Radiation Safety

[13] The next inspection of the GTRR, covering radiation controls and environmental protection, identified two further violations, one Level IV and one Level V. IR 87-02 (GANE Exh. 35). For these violations, the Applicant proposed corrective actions acceptable to the Staff. Staff Panel A, ff. Tr. 1740, at 15.

[14] Although the inspection giving rise to IR 87-03 was not conducted by Ms. Long, she was present at the enforcement conference which additionally considered practices uncovered in the inspection that Ms. Long had conducted.

Committee with the Nuclear Safeguards Committee (NSC). Staff Panel A, ff. Tr. 1740, at 16-17, 18.

NRC Region II issued five Severity Level IV violations based on IR 87-03. The Staff further noted that these violations, and the violations described in the NOVs accompanying IRs 87-01 and 87-02, raised concerns about the Applicant's management control and involvement in implementation of Georgia Tech's programs for radiation protection, reactor operations, and control of experiments. The Staff asked Georgia Tech to respond in a comprehensive way to the indications of management control problems by indicating the corrective actions it had taken or planned to take, and to describe how it planned to improve the working relations between the HP and reactor operations groups:

> in addition to the need for corrective action regarding the specific matters identified in the enclosed Notice, please address the root cause for the violations and the corrective actions you have taken or propose to correct the programmic deficiencies in the operation of your facility. *Particular attention should be given to how you will improve working relations between health physics and operations and adherence to written procedures by personnel at the facility.*

GT Exh. 8; GANE Exh. 31 (emphasis supplied); Staff Panel A, ff. Tr. 1740, at 17; Tr. 1767-68 (Collins).

In addition, the Staff noted that the Applicant had inappropriately expressed concern at the enforcement conference that its employees had reported safety concerns directly to the NRC, without providing GTRR managers an opportunity to resolve perceived or actual safety problems. The Staff acknowledged that the most effective way to resolve such issues is to have them brought directly to line management, and encouraged the Applicant to promote the type of working conditions in which employees feel their concerns will be appropriately addressed. However, the Staff reminded Georgia Tech that its employees had the right to provide information directly to the NRC, under section 210 [211] of the Energy Reorganization Act, as implemented by 10 C.F.R. § 50.7. GANE Exh. 31, at 2; Tr. 1531-32 (Long).

In its June 15, 1987 reply to the NOV, the Applicant identified difficulties in communications and coordination of work activities between the reactor operations and health physics groups at the GTRR, and continuing quarrels between the two groups, as the cause for several of the violations. The Applicant also noted that the HP group had identified problems and violations of NRC requirements but had not communicated them to the Director. The Applicant mentioned a proposed corrective action for these difficulties as a reorganization, under consideration for about a year, that would require the Manager of the

Office of Radiation Safety (MORS) to report to the NNRC Director. Staff Panel A, ff. Tr. 1740, at 17.[15]

#### (3) THE JULY 1, 1987 MANAGEMENT REORGANIZATION

Historically, the next matter of significance to management[16] was the reorganization that was implemented in July 1987. Georgia Tech's reasons for the reorganization are described later in this Decision (*infra,* pp. 309-10). Suffice it to say here that the NOV emanating from IR 87-03 (GT Exh. 8; GANE Exh. 31) issued on May 26, 1987, little more than a month prior to the reorganization (and included five Severity Level IV violations, together with NRC's expression of concern about Georgia Tech's management control and involvement in programs for radiation protection, reactor operations, and control of experiments).

Georgia Tech made its reorganization effective July 1, 1987, although it had failed to seek a license amendment from NRC.[17] By letter dated August 6, 1987, however, the Applicant belatedly submitted a license amendment request proposing to amend the GTRR organizational structure. Staff Panel A, ff. Tr. 1740, at 21; Staff Panel C, ff. Tr. 3171, at 12. (This proposed amendment, as well as several that followed, are discussed in greater detail *infra,* at p. 305.)

Shortly after the July 1987 reorganization, on August 19, 1987, a significant incident occurred at the reactor — the cadmium-115 spill (after the irradiation of a topaz crystal). The spill was not discovered by the NRC Staff until a December 16, 1987 inspection by Inspector George B. Kuzo. Staff Panel A, ff. Tr. 1740, at 19. This accident, including any reporting to NRC that might have been required, is discussed in detail *infra,* at pp. 284-85. We note here only that, contrary to the claim in GANE's contention, the accident occurred after, not before, the management restructuring and thus cannot be viewed as a cause for the restructuring.

The July 1987 reorganization caused considerable animosity and hard feelings at the GTRR, particularly among the HP staff which was then headed by Mr. Robert M. Boyd — whose title was changed from Radiation Safety Officer (RSO) to Manager, Office of Radiation Safety (MORS), and who thereafter was required to report to Dr. Karam, the NNRC director, in whose hands the responsibility for radiation safety had been placed. GT Exh. 6 (Figure 1); GANE

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[15] The NRC Staff later discovered that the Applicant had undertaken a management reorganization without receiving a license amendment or NRC authorization to do so. *See* IR 87-08 (Staff Exh. 12) and Testimony of Staff Panel C, ff. Tr. 3171.

[16] The next inspections, documented as IRs 87-04, 87-05, and 87-07, produced one deviation but nothing of significance to management of the GTRR. (IR 87-06, GT Exh. 7, was a report of the May 4, 1987 enforcement conference referenced above.) Staff Panel A, ff. Tr. 1740, at 18-19.

[17] This action was identified as an apparent violation in IR 87-08, but no violation issued because Georgia Tech had previously advised NRC that it was considering a reorganization. Tr. 1792-93 (Collins).

Exhs. 42, 43. Even prior to the reorganization, Dr. Karam, who had become Director in 1983, had attempted to assuage the group animosities by bringing the HP and operations personnel together socially. At his own expense, he invited the entire staff to Christmas luncheons in 1983, 1984, 1985, 1986, and 1987. He also started recognizing birthdays with brief office parties. Karam, ff. Tr. 2723, at 23. However, Dr. Karam opined that, notwithstanding these efforts, the reorganization had produced further problems and had not ameliorated the existing situation. Tr. 2773 (Karam).

Thus, Dr. Karam testified that, within 3 months of the reorganization, a number of incidents occurred at the NNRC which led him to believe that someone on the GTRR staff was engaged in "dirty tricks" or deliberate acts to damage the facility or impair its ability to function. These acts included damage to an expensive liquid scintillation counter, the erasure of floppy diskettes containing important data, the theft of two cases of batteries, placement of a bag of human feces in a staff refrigerator, and slashing of a large container of algicide causing the contents to spill on the floor. More significantly, in September 1987, a 500-watt light bulb above the 20-foot-deep Cobalt Storage Pool was smashed,[18] causing glass fragments to fall into the pool where they could interfere with the water filtration system; and three safety switches in the cobalt storage area were turned off at the same time, thereby disabling the associated safety alarms which were required under certain conditions to avert human exposure to lethal cobalt radiation. Karam, ff. Tr. 2723, at 31-33. Although there had been hostilities at the NNRC prior to the reorganization, Dr. Karam characterized these incidents as more serious than any that had occurred previously. Tr. 2785, 2786 (Karam).

Dr. Karam believed that the act of turning off the three Cobalt Pool switches was extremely serious from a safety standpoint, and was consistent with sabotage. Accordingly, he consulted with the Campus Police Chief (who also served as Deputy Chairman of the NSC) who suggested the use of a polygraph test. Dr. Karam then discussed polygraph testing with the entire NNRC staff in late September 1987; all (including Mr. Boyd) agreed to take the test, except for the two HP technicians in Mr. Boyd's unit — whose response was, "see our lawyer." Karam, ff. Tr. 2723, at 33-34; Tr. 2786, 2788 (Karam).

The two HP technicians' resistance to taking the polygraph examination caused Dr. Karam to wonder if they had been involved in these incidents. In the following two months, with hostilities between the HP and operations units continuing, it seemed to Dr. Karam that the two HP technicians' work performance was declining, that they were "disgruntled," that their attitude

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[18] Georgia Tech is authorized under its state license to possess a specified quantity of cobalt-60, which it stores in a "Cobalt Pool" under approximately 20 feet of water. Incidents concerning the cobalt-60 storage are not within our jurisdiction to resolve, except insofar as they may also pertain to the reactor itself.

bordered on insubordination, and that this could affect nuclear safety. Karam, ff. Tr. 2723, at 34-35; Tr. 2789-90 (Karam). Dr. Karam spoke about this situation to Dr. Stelson, who stated that he had heard similar statements about the HP staff from the NRC. Karam, ff. Tr. 2723, at 35; Tr. 2791 (Karam).[19]

On December 9, 1987, Dr. Karam advised Dr. Stelson that he believed the situation had deteriorated to the point that nuclear safety was involved, and in his opinion the HP staff should be replaced as quickly as possible with interim personnel. Karam, ff. Tr. 2723, at 36; Staff Exh. 25, at 14.[20] Dr. Stelson suggested waiting until January 1988, when a new Associate Director was expected to join the staff. They then agreed to speak to Dr. Bernd Kahn, Chairman of the NSC, about the situation. Dr. Kahn suggested that an assessment be obtained from an industrial psychologist prior to taking the contemplated personnel actions, to which they agreed. Drs. Karam and Stelson then engaged Dr. R. Michael O'Bannon, an industrial psychologist, and asked him to perform this assessment. Karam, ff. Tr. 2723, at 36; GT Exh. 10, at 1, 4.

#### (4) INSPECTION REPORT 87-08

The NRC inspection that commenced on December 16, 1987, conducted by Inspector George B. Kuzo, led to the identification of numerous violations in the areas of operations and health physics related to the cadmium spill and resulted in the NRC's issuance of the January 20, 1988 Order suspending reactor experiments. These events further degraded Dr. Karam's confidence in the HP staff — whom he also believed had provided damaging (and arguably inaccurate) information to the NRC (*see* note 20, *supra*, explaining that we have an inadequate record to resolve whether reports to Inspector Kuzo played any part in the proposed discharges of the two HP technicians). Following the

[19] Dr. Karam also stated that the two HP technicians were adversely affecting Mr. Boyd's decisiveness and effectiveness; and he believed that removing the two HP technicians would help to eliminate the strife at the facility. Tr. 2773-74 (Karam). In contrast, Mr. Boyd believed that the University's reason for replacing the HP staff was vindictiveness on the part of Dr. Stelson, due to Mr. Boyd's having closed down a (state-licensed) hot cell operation in early 1987, causing the loss of a $4000 contract. Tr. 2181 (Boyd), Tr. 2474-77 (Karam).

[20] Dr. Karam's recommendation to replace the two HP technicians was made one week before the commencement of Mr. Kuzo's inspection on December 16, 1987, thus supporting Georgia Tech's assertion that their discharge was based upon the HP-operations conflict and the HP technicians' conduct, rather than on a belief that they had reported problems to the NRC during Mr. Kuzo's inspection. Tr. 3490, 3491 (Karam). *See* Staff Exh. 25, at 14-15. However, the discharges were not announced or put into effect until after Inspector Kuzo's inspection, lending some credence to GANE's view that the discharges could have been motivated (at least in part) by advice given to NRC rather than Georgia Tech. *See* OI Report 2-88-003 (GANE Exh. 33), Synopsis, at 6. A Federal District Court apparently agreed, finding that one factor in the discharges was their report to NRC inspectors (in December 1987) of the August 1987 cadmium spill. *Millspaugh v. Karam*, Civil Action No. 1:88-cv-312-ODE (N.D. Ga. 10/31/91 (slip op. at 24-25, 27-28), *aff'd per curiam sub nom. Sharpe v. Karam*, 976 F.2d 744 (11th Cir. 1992) (Staff Exhs. 25, 26; Tr. 3457-58 (Karam). There is an insufficient record for us to resolve this question and, given its occurrence almost 10 years ago, we need not do so.

NRC's inspection "exit interview" on January 22, 1988, Dr. Karam concluded that removal of the HP staff should be expedited. Karam, ff. Tr. 2723, at 42-43, 44; Staff Exh. 25, at 24-27.

At about the same time, Dr. O'Bannon performed his psychological assessment of the GTRR organization and, in February 1988, reported to Dr. Karam. GT Exh. 10; Staff Exh. 25 at 17. Dr. O'Bannon concluded that Mr. Boyd's management of the HP unit was weak, that the level of hostility between the HP and operations units was too great and too entrenched to be repaired, that the HP staff showed a defiant attitude with no desire to correct the situation, and that one of the HP technicians (Mr. Millspaugh) was likely to have been involved in the "dirty tricks" referred to above. Karam, ff. Tr. 2723, at 37-38; Tr. 3197 (Karam).

Dr. O'Bannon recommended that the entire HP staff be removed from the NNRC and assigned elsewhere, and that a new manager of the HP staff be appointed to replace Mr. Boyd. Karam, ff. Tr. 2723, at 37; GT Exh. 10, at [unnumbered] 4. NRC Inspector Kuzo confirmed, based on the number of violations issued for poor performance by the HP group, that the group had problems necessitating some sort of remedial action. Tr. 1898 (Kuzo).

On February 11, 1988, Dr. Karam handed letters to the two HP technicians, Messrs. Paul Sharpe and Steven Millspaugh, advising each that his "employment will be terminated on February 25, 1988." On February 15, 1988, however, prior to the effective date of the proposed discharges, following discussions with counsel, Dr. Stelson "rescinded" the discharges, pending a hearing; and the HP technicians were thereafter reassigned to other duties outside the NNRC. Staff Exh. 25, at 20-21; Tr. 3198 (Karam).[21]

In IR 87-08, Mr. Kuzo identified significant reactor operations and radiation safety issues that required further NRC attention. Therefore, during the period of January 14-22, 1988, Region II management dispatched a special inspection team (which included Inspector Kuzo) to review selected GTRR program areas. The inspection team found numerous examples of failures to follow or to have adequate procedures to implement the Technical Specifications (TS), and/or violations of 10 C.F.R. Part 20 health physics requirements associated with the August 1987 experiment and the resulting Cd-115 contamination event. Staff Panel A, ff. Tr. 1740, at 19-20; OI Report No. 2-88-003 (GANE Exh. 33). These deficiencies involved both operational and health physics issues related to the pre-experiment review and calculation of dose rate levels for the topaz and cadmium container, as well as HP issues related to post-accident radiation surveys and evaluation of personal exposures. Tr. 1778 (Kuzo).

[21] At the time of this hearing, Mr. Millspaugh was still working for Georgia Tech (although not at the reactor). Tr. 3200-01 (Karam).

In general, the inspection findings identified continuing poor performance by Georgia Tech personnel regarding routine operations and HP activities. Details of these findings will be reviewed later, in connection with the cadmium-115 accident description, but particularly noteworthy was Georgia Tech's failure, by the time of the inspection (some 4 months after the accident), to have conducted a complete and thorough evaluation of the cadmium-115 contamination incident or to have implemented corrective measures to prevent recurrence during future experiments. Staff Panel A, ff. Tr. 1740, at 20.

Georgia Tech's failure to evaluate the incident and to implement corrective actions by the time of the inspection were perceived to indicate a lack of management involvement and control of operations and HP activities — which had been consolidated under the Director's control in the July 1987 reorganization. *Id.* at 20-21; Tr. 1835 (Fredrickson); Tr. 3219-20 (Karam). The lack of management involvement and control identified in IR 87-08 was considered by the NRC Staff to be detrimental to the safety of the facility. Tr. 1782 (Collins, Fredrickson, Gibson, Kuzo).

During this inspection, NRC Staff members also determined that working attitudes between HP and operations had continued to deteriorate, and informal training rather than procedures were used for many routine tasks. Operations personnel appeared satisfied with the NNRC Director's management efforts, but HP personnel indicated that the Director was involved too much in day-to-day health physics activities to the detriment of those HP activities. (At the same time, the Applicant added an NNRC Deputy Director, which NRC Region II viewed as a positive development because the individual selected had an operations background and had not been involved in the prior conflict between the HP and operations staffs; and because establishment of this position would assist the Applicant in improving its procedures and training. Staff Panel A, ff. Tr. 1740, at 21; Tr. 1888-91 (Fredrickson).)[22] IR 87-08 concluded that there had been no significant improvement in the Applicant's performance since the May 1987 enforcement conference and that the management control problem continued. Staff Panel A, ff. Tr. 1740, at 21; Staff Exh. 12, at 1-2.[23]

Particularly troubling to the NRC Staff were certain findings it reached concerning the surveys and bioassay performed by Georgia Tech HP personnel

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[22] The Staff was not concerned that this individual later resigned from the facility, or that the position has been vacant from April 1992 until the present, because (a) there has been no degradation in Georgia Tech's performance since the Deputy Director resigned; (b) the position was most needed to assist in resolving the problems that existed at that time (involving revisions to procedures, programs to ensure regulatory compliance, and the functioning of the organization), and those problems have since been resolved; and (c) there was no licensing or TS requirement for the position. Tr. 1891 (Fredrickson); Tr. 2981-84 (McAlpine).

[23] This inspection also raised concerns over the Applicant's proposed organizational change which, the NRC inspectors learned during this inspection, had been implemented on July 1, 1987, without the prior issuance of a license amendment. Staff Panel A, ff. Tr. 1740, at 21; Tr. 1792-94 (Collins); Tr. 1839 (Fredrickson). *See* p. 276, *supra*.

in response to the cadmium-115 contamination event. (These findings are reviewed, *infra,* in our discussion of the accident.[24]) Technical inadequacies also were identified in this inspection regarding personal contamination surveys and bioassays performed for the operator (Mr. William Downs) involved in the contamination event. Staff Panel A, ff. Tr. 1740, at 23-24; Tr. 1800, 1802, 1803-05 (Kuzo). (These inadequacies are addressed, *infra,* in our discussion of Mr. Downs.)

In IR 87-08, the Staff also determined that the Applicant had not conducted adequate surveys and analyses of possible airborne contamination in August 1987, after the incident occurred. Staff Exh. 12, at 7, 9; Tr. 1884, 1885-86. The survey results reviewed by the NRC included the August 24, 1987 memorandum to Dr. Karam from HP technician Paul Sharpe, who had served as the Decontamination Supervisor. Staff Panel A, ff. Tr. 1740, at 22-23. As we will review under the cadmium-115 incident, *infra,* that memorandum is not pertinent to our Decision here, except to the extent that it may relate to Georgia Tech's current policy concerning reports to the NRC.

In IR 87-08, the NRC Staff rejected Dr. Karam's reliance on the August 1987 air sample analysis. Staff Exh. 12 (Report Details at 9). The Staff also questioned the reliability of Dr. Karam's January 1988 analysis of air filter samples. GT Exh. 11; Staff Exhs. 27, 28; Tr. 2511-[illegible] (Boyd); Tr. 3423-25, 3441, 3444-50, 3465, 3472-74 (Karam). Again, we need not resolve this dispute. We recognize that there were certain deficiencies in the sampling techniques and procedures used in 1987-1988 but, as discussed later, those techniques and procedures do not persist, and those used today appear to be adequate. (For further elaboration of these matters, and to the extent relevant to our determination here, see our description of the Cd-115 accident, *infra.*)

**(5) SHUTDOWN ORDERS**

On January 20, 1988, the NRC issued an "Order Modifying License, Effective Immediately," which suspended all further irradiation experiments. Staff Exh. 13; Staff Panel A, ff. Tr. 1740, at 25. The Order stated that the Applicant's actions after the May 1987 enforcement conference had not been sufficient to address the management control problems, which continued. The order described the specific operations and health physics violations related to the

[24] At the hearing, there was considerable difference of opinion between the Staff and Georgia Tech concerning whether there had been adequate sampling of the contamination from the cadmium-115 incident and whether adequate records were available to evaluate the extent and levels of contamination. *Cf.* Staff Panel A, ff. Tr. 1740, at 22-24, and Tr. 1796-97, 1799, 1800, 1802, 1803-05, 1884, 1885-86, 1906 (Kuzo) *with* Karam, ff. Tr. 2723, at 40, 43-45, and Tr. 3206, 3433, 3439 (Karam). We need not resolve these questions here, however, inasmuch as the Applicant eventually took steps to improve its sampling procedures and techniques and the Staff has accepted the current procedures as adequate. Tr. 1791 (Fredrickson).

August 1987 contamination event, and it stated that Georgia Tech had failed to complete a thorough review of the event regarding its cause(s) and had not taken any corrective measures to prevent recurrence during future experiments. The order required Georgia Tech to cease utilization of the reactor facility for any irradiation experiments until the following requirements were met:

(1) assessment of management controls over facility operations;

(2) review of records for similar occurrences and identification of root causes;

(3) assessment of personnel exposures during the contamination and decontamination;

(4) review of facility health physics and operating procedures for inadequacies;

(5) identification and scheduling of corrective actions;

(6) development and implementation of a training program; and

(7) submission of the results of these assessments and reviews to the NRC for review.

Staff Panel A, ff. Tr. 1740, at 25.

On February 15, 1988, the President of Georgia Tech directed the immediate suspension of all reactor operations pending adequate resolution of all safety questions. Karam, ff. Tr. 2723, at 45-46. An NRC enforcement conference was held with Georgia Tech management on February 23, 1988. During this conference, the NRC representatives presented their view that a serious management problem existed at the NNRC, which was not limited to the facility's health physics organization. These representatives also expressed concern as to whether certain recent changes made at the facility, involving the replacement of HP personnel and the addition of an operator, would really solve the principal problems; and they stated that Georgia Tech management needed to provide an expectation of excellence by direction and example. Staff Panel A, ff. Tr. 1740, at 25-26; Tr. 1806 (Fredrickson). The NRC representatives also criticized the Applicant's failure to coordinate survey data collection related to the cadmium incident and to thoroughly investigate the incident and evaluate its seriousness. Georgia Tech was advised that its lack of regulatory sensitivity and its communications with the NRC did not compare favorably with other major research reactors located in NRC Region II. Staff Panel A, ff. Tr. 1740, at 26.

During the enforcement conference, Georgia Tech's President stated that he had decided the reactor would not restart until the Applicant and the NRC were both convinced that operations and health physics activities could be safely conducted. The Applicant also presented an NNRC action plan to the NRC. *Id.*

On March 17, 1988, based on Georgia Tech's self-initiated shutdown of the facility and its commitment to conduct an independent evaluation of the nuclear reactor program, the NRC Staff issued a Confirmatory Order Modifying License

(Staff Exh. 14). This order set out additional conditions that had to be met prior to restart of the reactor — specifically, (a) Georgia Tech was to submit a written identification of the root causes of problems that could impact safe operations of the reactor, and (b) the President of Georgia Tech was to submit to the NRC a written description of the corrective actions taken to resolve the problems, as well as the reasons he believed the facility should be allowed to restart. Staff Panel A, ff. Tr. 1740, at 26-27; Staff Exh. 14 (GT Exh. 15).

*b. The Cadmium-115 Accident*

In our review of the management history leading to shutdown, we referred to the cadmium-115 incident that occurred in August 1987 but was not discovered by the NRC Staff until December 1987. This was mentioned by GANE in both its contention and its FOF as a primary example of mismanagement. We now turn to this accident in detail.

As set forth earlier, GANE's management contention asserts in part that Dr. Karam is the Director who withheld information about a serious accident from the NRC — the 1987 Cd-115 accident. According to GANE, the NRC allegedly was advised of the accident by the RSO at that time (Mr. Boyd) who was later demoted and left the GTRR operation claiming harassment. We decide here whether the Director in fact withheld information from the NRC or retaliated against the RSO for reporting information to the NRC about the Cd-115 accident.

The Cd-115 accident occurred at the GTRR in August 1987, almost 10 years ago. When the Staff learned of the accident (in December 1987), it responded vigorously by conducting special inspections at GTRR, issuing orders to Georgia Tech, and finally issuing a civil penalty in November 1988.[25] We do not adjudicate the correctness of the Staff findings or actions in dealing with the incident in the 1987-1988 time period. The basic facts of the incident and Staff responses are undisputed. Some details not now material to license renewal remain in dispute between the Staff and Georgia Tech but they are not essential to our decision and we do not resolve them.

The event itself is material to license renewal at GTRR now only because the Director of the GTRR (Dr. Karam) at the time of the event remains Director now. At the hearing we permitted GANE the opportunity to demonstrate that the Director's actions taken at the time of the Cd-115 accident were not conducive to safety at the time and were part of a pattern of unsafe behavior which continues to the present day. We earlier made clear to GANE that, even if the Director

[25] Four violations were evaluated collectively as Severity Level III. A $5000 civil penalty was imposed — a base penalty of $2500 that was escalated 100% (i.e., doubled) because of Georgia Tech's prior poor performance and failures to take prompt corrective action to deal with the management control problems. Staff Panel A, ff. Tr. 1740, at 35-36; Tr. 1852-53, 1855 (Fredrickson, Collins).

made mistakes in the past, that would not be material to license renewal unless the behavior went substantially uncorrected to the present. Tr. 1521-22.

### (1) SUMMARY DESCRIPTION OF THE Cd-115 ACCIDENT

On August 18, 1987, Mr. William Downs, a Senior Reactor Operator (SRO) at the GTRR, transferred an irradiated topaz crystal from a cadmium-lined aluminum container to a glass beaker on the top of the reactor. During the irradiation the cadmium liner had become radioactive by neutron capture. Several isotopes of cadmium including Cd-115 and Cd-109 were formed. Unknown to the operator, however, the cadmium metal liner had partially disintegrated during the irradiation, possibly because of heat exposure in the reactor. When he poured the topaz from the container into the glass beaker, radioactive cadmium particles from the partly decomposed cadmium liner escaped and were spilled on the top of the reactor. Karam, ff. Tr. 2723, at 39-40; Tr. 3201-04, 3429, 3437 (Karam).

Subsequently, radioactive particles were carried either by air currents or gravity from the top of the reactor to the reactor containment floor below. Whether radioactive particles were transported to other parts of the reactor building is a matter in dispute between the Applicant and the Staff. Records that could resolve the matter are nonexistent. Karam, ff. Tr. 2723, at 40; Tr. 2256, 2503 (Boyd); Tr. 3432-33 (Karam); Staff Panel A, ff. Tr. 1740, at 22.

A small amount of radioactive Cd-115 was found on the containment building floor in a routine survey the next day, August 19, 1987. Subsequent investigation on the same day showed radioactive contamination measured at 20 millirem per hour at the top of the reactor where the topaz transfer was conducted. Karam, ff. Tr. 2723, at 39-40; GT Exh. 11; Staff Exh. 25, at 9; Staff Exhs. 27, 28; Tr. 2255-56 (Boyd); Tr. 3420-21, 3423-24 (Karam). Decontamination efforts were initiated under the direction of the MORS (Mr. Boyd) who, in turn, delegated operational responsibility for assessment and decontamination to a health physics technician. Tr. 3421 (Karam). On August 24 the HP technician reported in a memorandum to the Director (GT Exh. 12) that decontamination efforts were concentrated on several specific locations in the reactor building.

The wording suggested that contamination was found at each of the locations that were decontaminated. Karam, ff. Tr. 2723, at 40; GT Exh. 12, at 1; Tr. 3432 (Karam). The Director suspected that the memo was deliberately misleading and that there was no contamination beyond the locations where it was first identified. Tr. 3205-06 (Karam). However, survey records which could settle the issue were inadequate. Staff Panel A, ff. Tr. 1740, at 22; Tr. 2503 (Boyd); Tr. 3206 (Karam). Subsequent surveys showed no contamination, although limited hot spots remained which were later decontaminated. Tr. 3207 (Karam).

The Director reported the radioactive release to the Georgia Tech Nuclear Safeguards Committee (NSC) whose chairman was the ex-officio RSO. Neither the Director nor the RSO thought it should be reported to NRC because they had concluded that the accident lacked sufficient safety significance to be reportable. The MORS (Mr. Boyd) agreed that the event was not reportable but urged his management to report to the NRC anyway as a matter of prudence. Karam, ff. Tr. 2723, at 40-41; Tr. 2198-99, 2253, 2259, 2436-37 (Boyd). His advice was not followed. Later, the NRC Staff investigated the event and after some uncertainty because of incomplete records concluded that the accident was not a reportable event under Georgia Tech's Technical Specifications or 10 C.F.R. Part 20. Staff Panel A, ff. Tr. 1740, at 24; Tr. 1784-86 (Kuzo).

GANE did not pursue this aspect of its contention in its FOF. Neither did it direct our attention to any facts of record that contradict or suggest a substantially different view of events summarized above. Our review of the record did not reveal any conflicting information.

Accordingly, the Board finds that, contrary to the contention, the Director of GTRR did not wrongfully withhold information from the NRC concerning a serious accident (the 1987 cadmium-115 accident). The accident did not create a serious radiation hazard and was not required to be reported to the NRC under either the reactor technical specifications or 10 C.F.R. Part 20.

#### (2) GANE'S CLAIM OF MISTREATMENT OF THE SAFETY OFFICER

GANE's contention on the Cd-115 accident also suggests that the safety officer suffered retaliation from his management after informing the NRC of the accident. Although it is somewhat ambiguous, we assume that GANE initially intended this part to refer to the MORS. However, GANE did not pursue this allegation in its FOF and the other parties did not discuss it either.

We heard testimony from the former MORS (Mr. Boyd) where he could have but did not make the claim that his reporting concerning the Cd-115 incident resulted in his later removal from duty at the reactor and his reassignment to work elsewhere in the University system. He concurred with his management that the accident was not reportable to the NRC. The former MORS has many grievances against the Director for other reasons and he harbors hard feelings to the present time for actions taken against him. Tr. 2233-47 (Boyd). The hard feelings are based on his demotion prior to the Cd-115 spill and his perception that management unfairly blamed him and his HP staff for the Cd-115 incident when, in fact, the original release of Cd-115 was due to the carelessness of the SRO. In this he apparently misunderstood the significance of the accident, which did not create a serious health risk to anyone but did reveal to the NRC that there were serious deficiencies in management and HP procedures and practice at the reactor. The Board would take a serious view of a substantiated attempt by management to limit the flow of information about reactor operations to the NRC, as alleged by the contention. However, the disgruntlement of the former MORS based on disagreement with management decisions is not an important factor in the licensing decision before us even if the Director was biased or unfair to the employee at the time.

We find that, contrary to the contention, the Director did not retaliate against the MORS (Mr. Boyd) for passing information on the Cd-115 accident to the NRC.[26] Job actions taken against the MORS were related to the Director's adverse perception of job performance by the MORS and the HP staff. This view was formed in an ongoing process that both unfolded before the Cd-115 incident and was exacerbated by the HP staff performance in the wake of the incident.

The contention also claims that management was restructured to give the Director more control over the MORS after the Cd-115 incident. Although it is true and undisputed that management was restructured and that the responsibilities of the MORS were reduced — see discussion at pp. 276, 309-10 of this Decision — this occurred in July of 1987, before the accident in August. Thus the restructuring was not linked in any manner to the Cd-115 incident and it could not have been motivated by retaliation of the Director against the MORS stemming from the Cd-115 incident.

Our findings in this section are narrowly constructed to respond to GANE's admitted contention. The contention as filed reflected considerable initial misunderstanding on the part of GANE. Contrary to the assertions in the contention, we find that the Cd-115 accident was not treated by the NRC as an accident having serious health and safety implications. The Director was not required to report it to NRC. The MORS was not demoted or removed from duty by reason of information he reported to NRC about the accident. The management restructuring at GTRR happened before the incident and was not linked to it. Nor is there any evidence that the incident in any way resulted from the restructuring.

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[26] The Board interprets the contention to mean "MORS" where it refers to "the safety officer" and we have structured the Decision accordingly. We heard extensive testimony on the personnel problems that were rampant at the reactor at the time and are aware that the NRC Office of Investigations concluded that there were allegations of retaliation against two HP technicians who were supervised by the MORS for giving information to the NRC but that there were no intentional, contrived violations of regulations and licensing requirements. Staff Panel A, ff. Tr. 1740, at 29-30; OI Report 2-88-03, GANE Exh. 33. *See* note 20, *supra*. The thrust of the OI Report, however, was that there was severe mismanagement at the reactor, a fact not in dispute in this licensing action. Although these were serious matters at the time they unfolded, they are not material to the licensing decision now before us without additional evidence that the mismanagement has continued to the present day.

### 2. *Management Record After Restart*

*a. Record of Violations*

Restart of the reactor was authorized by the NRC Staff on November 15, 1988. Staff Panel A, ff. Tr. 1740, at 39-40; Staff Exh. 16. GANE relies on various Staff inspection reports following restart to demonstrate that managerial problems persist and, accordingly, that Georgia Tech's license should not be renewed. We here consider the management record after restart as reflected in pertinent Staff inspection reports from the restart date until the close of the record.

From January 1989 through April 1996, thirty-one inspections were performed by the NRC Staff to review numerous aspects of the Applicant's operation and management of the facility. The areas inspected include operational and maintenance activities, design change functions, operator licenses, requalification and medical activities, procedures, fuel movement, surveillance, experiments, effluent and environmental monitoring, emergency preparedness, radiation protection, safeguards and security, as well as the Applicant's organizational structure and review/audit functions. Among these thirty-one inspections, no violations were found in eighteen inspections; and seventeen cited violations (Severity Levels IV and V) and seven noncited violations were found and documented in the remaining thirteen inspections. Staff Panel B, ff. Tr. 2813. A brief description of these violations is given below.

**(1) INSPECTION REPORT 89-02**

An operations inspection was conducted in July and August 1989, and was documented in Inspection Report 89-02 (GANE Exh. 61). Two Severity Level IV violations were identified:

1. failure to perform leak-rate testing in accordance with commitments, and
2. inadequate procedure to assure that any shim blade not fully inserted was withdrawn sufficiently to cause a negative trip when released into the core.

Staff Panel B, ff. Tr. 2813, at 14. Adequate corrective actions were taken by the Applicant, and this matter was closed by the Staff. *Id.* at 14-15.

**(2) INSPECTION REPORT 89-05**

A security inspection was conducted during September 1989, as documented in IR 89-05 (GANE Exh. 64). The following six Severity Level IV violations were identified:

1. failure to maintain assessment equipment in operable condition and failure to properly position assessment equipment,

2. failure to secure a controlled access barrier,

3. failure to maintain the alarm system in operable condition,

4. failure to change keys as committed,

5. failure to control keys as committed, and

6. failure to establish and maintain a safeguards event log.

*Id.* at 15. This excessively large number of violations caused the Staff to be concerned about weaknesses in the Applicant's procedures used to implement its physical security program, and escalated enforcement action was considered by the Staff. GANE Exh. 64, at 1; Tr. 3046-47, 3162-63 (McAlpine). Corrective actions were taken by the Applicant to address these violations, and they were found to be acceptable by the Staff. Staff Panel B, ff. Tr. 2813, at 15-17.

#### (3) INSPECTION REPORT 90-02

A health physics inspection was performed during June 1990, and was documented in IR 90-02 (GANE Exh. 55). One Severity Level IV violation and one noncited violation were identified:

1. failure to maintain a high radiation area locked as required in 10 C.F.R. § 20.203(c)(2), and

2. failure to perform a personal survey at the exit to a controlled area. (Noncited violation.)

Staff Panel B, ff. Tr. 2813, at 17. Appropriate corrective actions, which included procedural revisions, counselling and training the individuals involved, were taken by Georgia Tech to address these matters. *Id.* at 17-18; Tr. 2822, 2825, 2827-28, 2995-97 (Bassett, Mendonca).

#### (4) INSPECTION REPORT 91-04

An emergency planning inspection was conducted during September 1991 and was documented in IR 91-04 (GANE Exh. 58). Although various emergency planning exercise strengths were observed, GANE Exh. 58 (Summary at 1-2), Tr. 3143-44 (McAlpine), two noncited violations were noted:

1. Inadequate procedure for implementing the Emergency Plan notification requirements, and

2. Failure to perform a biennial review of the Emergency Plan as required.

Staff Panel B, ff. Tr. 2813, at 18. The Staff found that the Applicant took appropriate corrective actions concerning these violations. *Id.* at 19.

##### (5) INSPECTION REPORT 92-04

An emergency planning inspection was conducted during November 1992 and was documented in IR 92-04 (GANE Exh. 57). One Severity Level V violation was noted during this inspection: failure to have an adequate procedure for implementing certain emergency planning notification requirements (a repeat of the noncited violation noted in Inspection Report 91-04). Staff Panel B, ff. Tr. 2813, at 19. Appropriate corrective actions were taken by Georgia Tech to address this violation. *Id.* at 19-20.

##### (6) INSPECTION REPORT 93-02

A combined operations and health physics inspection was performed in September 1993 and documented in Inspection Report 93-02 (GANE Exh. 60). Three Severity Level IV violations were cited as a result of this inspection:

1. failure of the Nuclear Safeguards Committee (NSC) to conduct the biennial audit of the licensed operator requalification program as required by Technical Specifications (the Manager of the Office of Radiation Safety performed the audit; he was not a member of the NSC).
2. failure to follow procedures for conducting neutron surveys, for completing certain twice-weekly contamination control surveys, and for completing survey forms required for shipping radioactive material, and
3. failure to comply with 49 C.F.R. Part 172 requirements concerning the description of radioactive material being shipped and indicating a 24-hour emergency response telephone number on shipping documents.

Staff Panel B, ff. Tr. 2813, at 20. Appropriate corrective actions were taken by the Applicant concerning these matters, including a commitment that the NSC would thereafter perform the required audits, procedural revisions, and revision of the shipping forms. *Id.* at 20-21.

##### (7) INSPECTION REPORT 93-03

An emergency planning inspection was conducted during November 1993 and was documented in IR 93-03. One noncited violation was noted: failure to perform periodic testing of the criticality alarm system in accordance with procedure. The required monthly tests of the system were not performed during

May, June, and July 1993. Appropriate corrective actions were taken by the Applicant concerning this matter. Staff Panel B, ff. Tr. 2813, at 21-22.

#### (8) INSPECTION REPORT 94-01

An unscheduled inspection was conducted during March 1994, to follow up on an incident involving the failure of a Senior Reactor Operator (SRO), William Downs, to follow procedures that resulted in two disabled reactor scram functions. *Id.* at 22; Tr. 2860-61 (Mendonca); Tr. 2865 (McAlpine). This inspection was documented in IR 94-01 (GANE Exh. 59). One noncited violation with two examples was identified:

1. failure to complete the actions required by the checklist for startup of the reactor on February 15, 1994 (a fuse was not replaced after it had been removed during a training session, as the checklist required), and
2. failure to complete the actions required by the checklist during shutdown of the reactor on February 11, 1994 (three electrical jumpers had not been removed).

Staff Panel B, ff. Tr. 2813, at 22; Tr. 2862 (Bassett, Mendonca). These incidents were classified as noncited violations because the disabled scram functions were not required under the Technical Specifications (TS) for safe operation of the reactor, since they generally provide equipment protective functions, and credit is not taken for them in accident mitigation in the Final Safety Analysis Report. Staff Panel B, ff. Tr. 2813, at 22; Tr. 2863-64, 3155 (McAlpine, Bassett).

Following the incident, the Applicant took corrective actions which included reviewing the incident, temporarily suspending the SRO's reactor operating duties, and establishing a panel to further investigate the incident and the SRO's operating history to recommend what further actions should be taken. The Applicant's panel evaluated the technical performance of the SRO with respect to the incident of February 15, 1994, as well as the SRO's historical performance, and determined that, because of the SRO's lack of diligence to safety and poor past performance, the suspension of the SRO should remain in effect until there was an obvious change in attitude and a commitment to follow procedures. The SRO subsequently terminated employment at the facility in June 1994. Staff Panel B, ff. Tr. 2813, at 22-23; Tr. 2800-02, 2804 (Karam); Tr. 2865-66 (McAlpine). See further discussion of Mr. Downs, *infra*, pp. 292-95.

#### (9) INSPECTION REPORT 94-02

A health physics inspection was conducted during August 1994 and was documented in IR 94-02 (GANE Exh. 56). One violation (Severity Level IV) was cited: failure of the Applicant to make a proper evaluation of the extent of the radiation present following the annual neutron radiation survey performed August 11, 1994, which was required by procedure. Staff Panel B, ff. Tr. 2813, at 23. The Applicant subsequently took appropriate corrective actions concerning this matter. *Id.* at 23-24.

#### (10) INSPECTION REPORT 94-04

An emergency planning inspection was performed during October 1994 and was documented in Inspection Report 94-04. One noncited violation was noted: failure to submit emergency procedure changes to the NRC in accordance with section 10.4 of the Emergency Plan. *Id.* at 24. Adequate corrective actions were taken by the Applicant with respect to this matter. *Id.*

#### (11) INSPECTION REPORT 94-05

An operations inspection was conducted during December 1994 and was documented in Inspection Report 94-05 (GANE Exh. 63). One noncited violation was noted: failure to replace the charcoal cartridges every 2 weeks as required by Technical Specification 6.4.b(6). Staff Panel B, ff. Tr. 2813, at 24-25. Appropriate corrective actions were taken by Georgia Tech with respect to this matter. *Id.* at 25.

#### (12) INSPECTION REPORT 95-01

A health physics inspection was performed during February and March 1995 and the inspection results were documented in IR 95-01 (GANE Exh. 66). Two violations (one Severity Level IV and one Severity Level V) were identified:

1. reporting failures, by: (a) omission of some of the required data and providing inaccurate data in annual reports concerning liquid and gaseous radioactive effluents to the NRC for the years 1983, 1986, and 1988 through 1993, and (b) providing inaccurate information to the NRC in the 1994 Safety Analysis Report concerning continuous, automatic measurement and recording of meteorological data, and

2. failure to have a Nuclear Safeguards Committee (NSC) approved procedure to calibrate and operate the alpha/beta proportional counter.

Staff Panel B, ff. Tr. 2813, at 25. Appropriate corrective actions were taken by the Applicant with respect to the inaccurate reporting data, including the creation of a computer data base for gaseous and liquid discharges, and the correction of the inaccurate portions of the annual reports and FSAR. *Id.* at 25-26. Appropriate corrective actions also appeared to have been taken with respect to the failure to have an NSC-approved procedure, although verification of these corrective actions had not yet been completed and documented by the NRC Staff prior to the commencement of hearings in this proceeding. *Id.* at 26.

#### (13) INSPECTION REPORT 95-02

A security inspection was conducted during May 1995 and was documented in IR 95-02. One violation (Severity Level V) was identified: failure to submit material status reports within 30 days of March 31 and September 30 of each year as required by 10 C.F.R. § 74.13(a)(1). *Id.*; GANE Exh. 69; Tr. 3097 (Mendonca). Appropriate corrective actions were taken by the Applicant to resolve this matter. Staff Panel B, ff. Tr. 2813, at 26-27.

#### (14) SUMMARY

As stated earlier, none of these violations identified by the Staff in the period following restart was more serious than Severity Level IV, and the corrective actions taken by the Applicant were assessed to be adequate by the Staff. In addition, in none of the inspections from May 1995 through April 1996 were any violations identified, at least as reflected by the record herein. The Staff explicitly indicated that the decreasing frequency of violations with the passage of time was a factor it took into account in assessing the adequacy of management. Tr. 3151 (McAlpine, Mendonca). Therefore, collectively, the identified violations together with other inspection findings do not present a picture of serious management deficiency during the January 1989 through April 1996 period.

### *b. Employment History of William Downs*

One matter stressed by GANE as an example of poor management by Georgia Tech — "a glaring problem" — is the failure to take any action until 1994 against Mr. William Downs, an SRO at the GTRR from 1976 until June 1994. GANE FOF at 8. Mr. Downs was involved in several serious incidents at the reactor, two of which we have previously alluded to (i.e., the cadmium-115 incident of August 1987 and the disabled scram functions of March 1994). GANE claims that his employment history raises questions as to the adequacy of personnel management during this period of time.

Specifically, to rehearse the incidents involving Mr. Downs:

(a) February 1985 — Striking of Hot Cell Window with a wrench while manipulations were in progress. Mr. Downs explained that he struck the window accidentally during horseplay. Staff Exh. 22, Enclosure 2, at 1.

(b) January 1986–February 1987 — Failure to isolate sample line per procedure when performing monthly surveillance. IR 87-02. Mr. Downs explained that this procedure had little safety significance and that he violated it for convenience sake. However, he claims that, as of June 1988, he was strictly adhering to the procedure. Staff Exh. 22, Enclosure 2, at 1.

(c) 1986 — Failure to fill out or complete Experiment Schedule Forms or Experimenter's Checklists. IR 87-01. Mr. Downs admitted his error. He was counseled by the NNRC Director on procedural adherence after the NRC violation was issued. Staff Exh. 22, Enclosure 2, at 2.

(d) March 1986–November 1986 — Failure to wear dosimetry and protective clothing in areas requiring their use. IR 87-03. Mr. Downs could not recall any failure to wear dosimetry or protective clothing when they were required. Staff Exh. 22, Enclosure 2, at 2; Enclosure 3, at Event 5.

(e) 1986 — Failure to log Initial Conditions and Equilibrium Conditions per Procedure 2000, "Reactor Operation" on frequent occasions, as well as numerous missing/incomplete log entries. IR 87-01. Mr. Downs responded that, during 1986, there were three instances where the Initial Critical Data (ICD) stamp was not completely filled out. On two of these occasions, a reactor scram occurred within 2 minutes of reaching power, and he had no opportunity to fill out the log. On the other occasion, he put the ICD stamp in the logbook out of sequence and forgot to go back and cross it out after completing his log entries. The ICD stamp was filled out after being restamped at the proper time. Mr. Downs stated that he would pay more attention to this procedure in the future but would also bring to management's attention a deficiency in the procedure. Staff Exh. 22, Enclosure 3, at Event 6.

(f) February 1987 Power Excursion from 300 kW to approximately 2 MW while power was supposed to be stabilized during conduct of Beam Port operations. IR 87-01. Mr. Downs asserted that he believed he reacted in a safe manner, in that the time between the power excursion and his actions was not excessive. He blamed the event on a stuck power level indicator. However, the Staff observed that there were other indicators and the event took place over a period of approximately 10 minutes and was not terminated until radiation monitors alarmed. Staff Exh. 22, Enclosure 2, at 2.

(g) August 1987–February 1988 Inadequate log keeping and control of an experiment resulting in the overexposure of a topaz experiment. Subsequent contamination event was due to poor HP practices and inadequate communications with facility management. Inconsistent information was provided to the NRC regarding post-spill activities, in particular the radiation monitoring of his residence. IR 87-08. This is the cadmium-115 incident that we have reviewed elsewhere in this Decision (*see* pp. 283-86, *supra*.)

(h) February 1994 Failure to follow procedures that resulted in two disabled reactor scram functions. IR 94-01 (GANE Exh. 59). (*See* p. 290, *supra*.)

The foregoing history of events indicates that, during the early years of Mr. Downs' service, there were a number of events that might have warranted personnel action against him and which motivated the Staff to have an enforcement conference with him on May 20, 1988. Following the conference, the Staff determined to take no enforcement action with respect to Mr. Downs' SRO license but advised him of its concern "over your lack of adherence to procedures, your lack of diligence in recording information in operating logs and experiment forms, and your casual attitude displayed during the August 1987 contamination incident." Staff Exh. 22, letter to Mr. Downs from J. Nelson Grace, Regional Administrator, Region II, dated June 17, 1988.

Mr. Boyd blamed Mr. Downs (at least in part) for the 1987 HP-Operations hostilities, which we have described earlier in this Decision. Mr. Boyd believed that the HP technicians were being unfairly singled out for the conflict, instead of Mr. Downs. He regarded Mr. Downs as demonstrating a hostile attitude toward health physics or to anyone telling him what to do, as showing a total neglect for complying with procedures, and as being subject to repeated bursts of anger. Tr. 2165-68 (Boyd).

Mr. Boyd recommended to Dr. Karam that Mr. Downs' services be terminated following the cadmium-115 incident. Dr. Karam agreed. He testified that Mr. Downs had been asked to take a geiger counter home to his apartment to check on radioactivity from the cadmium-115 incident but could not remember whether he (Downs) had done so. Tr. 2798-99 (Karam). Dr. Karam believed that Mr. Downs "somehow didn't forget, he was playing games" (Tr. 2799) and accordingly requested to Dr. Stelson that Mr. Downs be terminated. Apparently, Dr. Stelson believed that people forget many things and instead recommended a psychological examination, which Mr. Downs passed. *Id.*

Mr. Downs served satisfactorily until the incident involving disabled reactor scram functions occurred in February 1994. Tr. 2800 (Karam); Tr. 2866 (McAlpine). Following the incident, the Applicant took corrective actions, which we have earlier described (*see* p. 290, *supra*), leading to Mr. Downs' suspension[26a] and his subsequent termination of employment at the facility in June 1994. Staff Panel B, ff. Tr. 2813, at 23; Tr. 2800-02, 2804 (Karam).

Our evaluation of Mr. Downs' service indicates, as Mr. Boyd suggested, that his horseplay incident in February 1985, and the attitude it reflected, may have warranted the immediate termination of Mr. Downs' services as a reactor operator. Several later incidents, including the cadmium-115 incident, also may have warranted his termination, as Dr. Karam recommended. Management's failure to take action against Mr. Downs until February 1994 perhaps reflects poorly upon it (although not on Dr. Karam).

But the failure to take action earlier is not sufficient to disqualify management from acting under a renewed license. This is particularly so when the current Director of the facility sought (unsuccessfully) to take action following the cadmium-115 incident. Furthermore, none of the evidence — except perhaps a surmise by Mr. Boyd (Tr. 2169) — supports GANE's claim that Mr. Downs was not discharged because the reactor would have lacked sufficient personnel to operate and produce a monetary return. Dr. Karam had responsibility for producing a monetary return, and he in fact sought to terminate Mr. Downs' employment.

[26a] The 1994 incident raised concern in NRC Region II over Mr. Downs' lack of diligence and caused the Staff to consider whether Mr. Downs' SRO license should be suspended or revoked. Tr. 2869 (McAlpine), Tr. 2872 (Mendonca). The Staff, however, considered Georgia Tech's suspension of Mr. Downs to be responsible and appropriate. Accordingly, the Staff took no action on its own, pending the outcome of the Applicant's evaluation. Tr. 2872 (Mendonca).

*c. Intrusion by Fox TV Film Crew*

One example of alleged mismanagement relied on by GANE was based on events occurring after the initiation of this proceeding. In early October 1995 (Tr. 2621 (Carroll)), a film crew from the television series "A Current Affair" visited the Georgia Tech site and, with its camera rolling, made its way into the administration building which adjoins the reactor containment building. A filmed record of their "intrusion" or "incursion" (Tr. 2621 (Carroll)) (i.e., entry) into the reactor complex was broadcast by Fox Television on November 15, 1995, and personally videotaped from the broadcast by Ms. Glenn Carroll, GANE's representative in this proceeding. Tr. 2620-22 (Carroll), 2653.

On November 10, 1995, after the "intrusion" although prior to the broadcast, GANE sought to introduce a new contention concerning security of the facility based on the incident. We preliminarily considered this proposed new contention at a prehearing conference held in Atlanta, Georgia, on November 15, 1995 (the same day as the broadcast). GANE offered to submit a videotape of the program in support of the new contention. At the conference, GANE also described the incident as having management implications (Tr. 520). We dismissed the new contention without prejudice to its being refiled along with a discussion of the factors relevant to late-filed contentions. Second Prehearing Conference Order, dated November 29, 1995 (unpublished).

On January 1, 1996, GANE provided the videotape to the parties and resubmitted the incident as part of its management contention. By our Memorandum and Order (Telephone Conference Call, 5/15/96), dated May 16, 1996, LBP-96-10, 43 NRC 231, 233, and as reiterated at the hearing (Tr. 2617), we determined that the tape was relevant to the management contention. Thereafter, we admitted into evidence the video portion of the tape (GANE Exh. 54), along with limited portions of a transcript of the broadcast (GANE Exh. 53). Tr. 2677-98.[27]

GANE contends that the film crew's ability to intrude, unimpeded, into the reactor complex demonstrates inadequate ("sloppy") management on the part of the Applicant. *See, e.g.*, Tr. 2669-70 (Carroll). Although Ms. Carroll was not present at the site during the film crew's entry into the reactor complex, she had been informed that members of the film crew were dressed like students and that a small, concealed hand-held camera was used in the filming. Tr. 2651, 2654-56. Ms. Carroll stated that the film crew tried to open certain doors but found them to be locked, and that they did not get into the room where the radioactive cobalt is stored or into the reactor containment. Tr. 2656-57, 2658 (Carroll). She pointed out a sign they had filmed, indicating the presence of

[27] We determined that the part of the audio that was descriptive of various events on screen was relevant but that other comments of the narrator that attempted to characterize the events or to provide interpretive comments were inappropriate, at least in the absence of the narrator who could be cross-examined. Tr. 2617.

radioactive materials — however, she did not know if entry had been made into areas containing radioactive materials, or if the facility's security plan was breached;[28] and she did not identify any violation of a regulatory requirement. Tr. 2649-50, 2657-59, 2660-61 (Carroll).[29]

Upon receiving a report of this event, an NRC Region II safeguards inspector conducted an inspection of the facility on October 31–November 3, 1995; the results of that inspection are summarized in Inspection Report 95-04. No violations or deviations were identified in this inspection. GANE Exh. 65 (Summary at 2; Report Details at 1, 3). The inspector determined that the film crew toured interior and exterior areas of the NNRC that are not subject to control under the GTRR security plan — including hallways in the administration building, a stairwell leading to the visitors' observation window, the roof of the administration building, and a fenced storage yard. GANE Exh. 65 (Summary at 2; Report Details at 1). The film crew was videotaped challenging two security doors, which remained locked. No breach of security or the security plan was identified; and there was no indication that the television crew had unauthorized access to protected or radiation-controlled areas. GANE Exh. 65 (Summary at 1-2; Report Details at 1-2); Tr. 3058 (Mendonca); *see* Tr. 3511-12 (Karam).

The NRC safeguards inspector spoke with Georgia Tech personnel concerning this event, and verified that access controls, barriers, alarms, assessment capabilities, and response to alarms were in accordance with the GTRR security plan. The inspector subsequently viewed the television broadcast of the event on November 15, 1995, and determined that it contained no indication that the television crew had unauthorized access to protected or radiation-controlled areas. GANE Exh. 65 (Report Details at 2-3); Tr. 3061-62 (McAlpine). The videotape did not lead to the identification of any weaknesses in the Applicant's security program. Tr. 3068 (Mendonca).[30]

After the event occurred, the facility director discussed it with all NNRC staff and students. Notwithstanding the fact that no violations or deviations were identified as a result of this event, the Applicant subsequently revised

[28] In contrast, Dr. Karam stated that the signs that appear in the videotape are located *outside* secured areas in which radioactive materials were present, and that the film crew only entered a public building that was open to students who come and go to classes there. Tr. 3511-12.

[29] GANE was not permitted to have access to the security plan, although earlier it had sought such access. Ms. Carroll offered a "common sense" opinion that the facility security plan should utilize fences and barbed wire. Tr. 2661, 2665. Ms. Carroll's education and experience (consisting of a Bachelor of Arts degree in visual arts, and experience as an artist, typesetter, and graphics designer, Tr. 2665-67) do not qualify her to render an expert opinion on this subject. Moreover, undoubtedly because she would have had no reason to be granted access, Ms. Carroll has never seen a security plan for any nuclear reactor, and she did not know (nor could have known) what security measures are in place at any other research reactor. Tr. 2667-68.

[30] The videotape showed that one individual (whom GANE identified as a reactor operator) allowed the film crew to continue in its intrusion into the administration building, unimpeded. This individual was not remiss in this regard, since there is no requirement for him to have done anything to limit their access to that area. Tr. 3068 (Mendonca).

its security measures, by restricting access to the NNRC to require use of an existing coded key card reader or the presence of an authorized individual to open the front entrance to the facility;[31] also, additional patrols by the campus police, whose office is located across the street from the reactor facility, were put into effect. GANE Exh. 65 (Report Details at 3); Tr. 3263-64, 3513 (Karam).[32] Georgia Tech's voluntary institution of these additional security measures was over and above NRC requirements. The Staff would not have required the Applicant to take these actions. Tr. 3054-56 (McAlpine); Tr. 3069-70 (Mendonca, McAlpine).

Upon review of the evidence on this event, we agree with the Staff (Staff FOF at 108) that the Fox Television film crew's intrusion into the reactor complex does not reflect inadequate management by the Applicant.[33] To the contrary, the security plan appears to have worked as intended, in compliance with applicable regulatory requirements. Further, as observed by the Staff (*id.*), the Applicant's subsequent decision to upgrade its security measures beyond the requirements of the security plan may be viewed as demonstrating good managerial judgment. Thus, this matter does not provide grounds for denying or conditioning the license renewal application.

### d. *Hardware Issues*

As part of its claim of poor management, GANE asserted that the GTRR had operated for extended periods of time using equipment that needed repair. We turn to an analysis of these claims.

#### (1) THE BISMUTH BLOCK

GANE asserted that the continued existence of a water leak in the bismuth block is evidence of inadequate management at the reactor. GANE did not pursue its concerns in its proposed findings of fact and did not direct our attention to any part of the record that could support its assertion. Neither did Georgia Tech address the matter in its proposed findings. We therefore find that this is a matter no longer in controversy between Georgia Tech and GANE and, accordingly, adopt the proposed findings of the NRC Staff on this matter, as summarized below. Staff FOF, §§ 2.4.2.1-2.4.2.5, at 99-102. We set forth below

[31] The key card reader at the front door was in place previously, but was only used when the door was locked (i.e., from 5:00 p.m. to 8:00 a.m.). Tr. 3522, 3530 (Karam).

[32] In addition, a new fence has been installed at the facility, with an alarm that activates at the NNRC and the campus police station if the fence is cut, climbed, or shaken. Tr. 3513. This fence was installed in connection with the advent of the 1996 Olympic Games, but Dr. Karam indicated that Georgia Tech intends to keep it in place after the Games have concluded. Tr. 3522-23, 3525.

[33] Georgia Tech submitted no proposed findings regarding this event.

a brief summary of the testimony on the bismuth block and find that leaking coolant has no safety significance, and is not material to license renewal.

The bismuth block is part of a shield within a biomedical beam port at the reactor. Its purpose is to attenuate gamma rays and permit neutrons to pass through for use in experiments. The bismuth block is cooled by a water source independent of any source in the reactor. The cooling system is not part of an accident mitigation system at the reactor. In August 1983, heavy water was found leaking from the bismuth block. Water drained to the basement of the reactor building. The wet area was posted as potentially contaminated and the reactor was shut down. After analysis, the leak was sealed with a commercial radiator stop leak compound and reactor operations resumed. The bismuth block coolant was converted from heavy water to ordinary light water in 1983.

The seal was successful until 1989, when the leak reappeared. An attempted repair using "stop leak" and epoxy compounds did not succeed. The leak did not interfere with the block cooling function and radioactivity levels remained below regulatory limits. Rather than attempting further repairs of the leak, the Applicant installed an NRC-approved collection system to catch and store the leaking water. The collection system is now functioning and no running water has been observed, although the basement area is damp. The bismuth block leak has no health and safety implications. Since there is no safety function, the Applicant is permitted by NRC to use the bismuth block in its current condition. The bismuth block leak raises no concerns with respect to the license-renewal application.

We have reviewed the record and find no contrary evidence to that cited by the Staff and summarized above. Accordingly, the Board finds that the water leak in the bismuth block is not evidence of poor management at the reactor and is not material to our decision on license renewal.

### (2) FUEL-ELEMENT FAILURE

GANE has asserted that a fuel-element weld failure is evidence of inadequate management at GTRR because of failure to notify NRC. Neither GANE nor the Applicant addressed the matter in their proposed findings and the Board considers the matter no longer in controversy. The NRC Staff's uncontested Findings of Fact state that the Staff was notified both in writing and by telephone in September 1992. The weld failure was not a violation of NRC regulations or of the GTRR license. The affected fuel element was removed from the reactor and was placed in storage in the fuel pool. Staff FOF, §§ 2.4.3.1, 2.4.3.2, at 2. We find that this event has no public health and safety significance and does not present a concern with respect to license renewal.

### (3) ENVIRONMENTAL MONITORING

GANE asserts in its proposed findings that it "remains concerned about Neely management's ability to contain radiation from the environment and their ability to monitor the contamination that is occurring." GANE FOF at 10. GANE claims that Georgia Tech has been cited by NRC for errors and omissions in environmental monitoring data over a 10-year period from 1983 to 1993. The asserted errors include errors in math, gaps or blanks in data, absence of meteorological monitoring equipment for 10 years, and submission of the same windrose diagram repeatedly. *Id.*; IR 95-01 (GANE Exh. 66).

GANE asserts that in 1996 the Applicant was cited for failure to calibrate the GM gas monitor in timely fashion. It cites in support NRC IR 96-02 (apparently not offered into evidence). Although we cannot confirm that the NRC inspection report has been admitted to the record, nevertheless we find reference to calibration of a GM gas monitor cited in NRC IR 95-01 (GANE Exh. 66). It was left as an open item in that report (*id.* at 21). Thus, GANE's calibration assertion cannot be substantiated.

We note also that GANE cross-examined at length on issues related to environmental monitoring around the reactor using film badges and thermoluminescent dosimeters (TLDs). Tr. 2903-25. It did not pursue these matters further in its proposed findings of fact.

GANE's licensing concern appears to stem from reports of radiation levels above background, set forth in IR 93-02 (GANE Exh. 60). GANE asserts that there is a lack of reliable data as to what (radiation) the environment has received from operations at the NNRC and that it may never be known what the risk to the population is. GANE urges the Board to deny the license renewal to prevent the reactor from operating in its "broken-down, slip-shod fashion for another 20 years." GANE FOF at 10.

On review of IR 93-02, the Board finds that the Applicant was cited for violations as asserted in GANE's proposed findings. The inspection report, however, shows that no citation for a violation was more serious than Severity Level IV.

We adopt the NRC Staff's uncontested proposed findings on issues related to film badges and TLDs in this Decision. Staff FOF, §§ 2.4.4.1-2.4.4.4, at 103-04, to the effect that GANE's concern for environmental monitoring using film badges and TLDs does not involve possible violations of NRC regulations, inasmuch as Georgia Tech is not required by regulation or license condition to perform such monitoring. It does so under a commitment starting in 1966 in the SAR to place thirty monitoring devices in the environment around the reactor. Tr. 2915 (Mendonca). Georgia Tech used film badges for monitoring for many years but converted to TLDs in 1994 or 1995. *Id.*; Tr. 2919 (Mendonca). The use of film badges or TLDs is equally acceptable to the Staff and its approval of the Georgia Tech application is not dependant on which was chosen. Tr. 2924 (Bassett).

GANE expressed concern that environmental monitoring had unacceptable uncertainty because some film badges in the past showed false radiation doses which were attributable to physical damage from rain and heat. Tr. 2906 (Mendonca). This concern is laid to rest, however, by Applicant's testimony that all the badges were not affected and that the plant has other monitoring devices plus monitors required by technical specifications in place. Furthermore, the TLDs now in use are not subject to damage from heat and moisture. Tr. 2908 (McAlpine).

The Board concludes that even though some film badges in the past showed false positive radiation readings, there was sufficient redundancy in monitoring devices to preclude uncertainty in radiation measurements large enough to be significant to public health and safety. Our confidence is enhanced by the fact that the errors asserted by GANE result in false positive readings in which the monitoring device appears to detect radiation when none is detectable by unaffected devices. This type of error attracts notice and requires analysis. Tr. 2910-11. (Bassett). Thus, there is little likelihood that false positive error could lead to a failure to detect radiation emissions to the environment, if any actually occurred. The Board accordingly concludes that GANE's concerns for environmental monitoring based on the Applicant's use of either film badges or TLDs is not well founded and does not present a concern for licensing.

### 3. *Georgia Tech's Management Organization Structure*

At the heart of GANE's concerns over Georgia Tech's management is the organizational structure of that management. As described by GANE:

> The most unique aspect of the management of the Neely Nuclear Research Center at Georgia Tech, and the one that caused us the most trepidation about the facility to begin with, is the management structure which places the Director of the facility over the Manager of the Office of Radiation Safety [GANE FOF at 3].

#### a. *Applicable Standards*

The acceptability of a managerial organizational structure depends, in part, on the independence of operational functions and safety functions. NRC regulations prescribe no particular managerial structure, either for power reactors or research reactors. Staff Panel C, ff. Tr. 3171, at 9. With respect to power reactors, however, interpretations of quality assurance requirements have led to a mandatory separation of operational and safety functions. 10 C.F.R. Part 50, Appendix B.I; *see, e.g., Consumers Power Co.* (Midland Plant, Units

1 and 2), ALAB-152, 6 AEC 816, 817 (1973) ("those charged with the function of assuring the quality of particular work must be independent of the individual or group having direct responsibility for performing that work"). Given the absence of regulatory requirements for any particular organization or management structure for nonpower reactors, those structures vary considerably, so long as some form of independent safety review is maintained.

*b. Examples of Organizational Structures*

Although some variations among types of managerial structures for research reactors exist, essentially two forms of organization are considered acceptable.

The first, recommended by Georgia Tech consultant Dr. Nicholas Tsoulfanidis, by the current MORS, Dr. Rodney D. Ice, as well as by several GANE witnesses, is comparable to the organizational model for power reactors. The operational Director reports to a high-level official — the Dean of Engineering — whereas the Radiation Safety Officer reports to another high-level official — the Vice Provost for Research. Both the Dean of Engineering and the Vice Provost for Research in turn report to a higher level, the Office of the President. Tsoulfanidis, Prepared Testimony, ff. Tr. 1939, at Exh. GT-2. *See* Figure 1, p. 303, *infra. See also* GANE Exh. 42 (GTRR Organization Chart Before 7/1/87). This model is essentially what Georgia Tech utilized prior to the 1987-1988 reorganization.

The second, relied on by the Staff, is based upon the "American National Standard for the Development of Technical Specifications for Research Reactors," ANSI/ANS-15.1, which includes a section on administrative controls. That version, initially set forth in 1982 as ANSI/ANS-15.1-1982, includes a level 1 unit or organizational head; a level 2 reactor facility director or administrator reporting to level 1; a level 3 reactor or shift supervisor reporting to level 2; and a level 4 that consists of the operating staff reporting to level 3. Review and audit functions are performed at a level above the facility director and report to level 1 management. Radiation safety personnel report either to level 2 (the director/administrator of the facility) or to level 1 (unit or organizational head).

This type of organizational structure permits the Radiation Safety Officer to report either to a level above the operational director — in effect like the first plan recommended by Dr. Tsoulfanidis — or to the Director. If reporting to the Director, safety review functions are overseen by entities outside the line of operational functions, although the direct reporting remains within that line. A chart of the ANSI-approved structure, as revised in 1990, is set forth as Figure 2 on p. 304, *infra*.

Although the ANSI standards referenced above do not constitute regulatory requirements, the NRC Staff participated in their development and has encour-

Proposed Administrative Structure

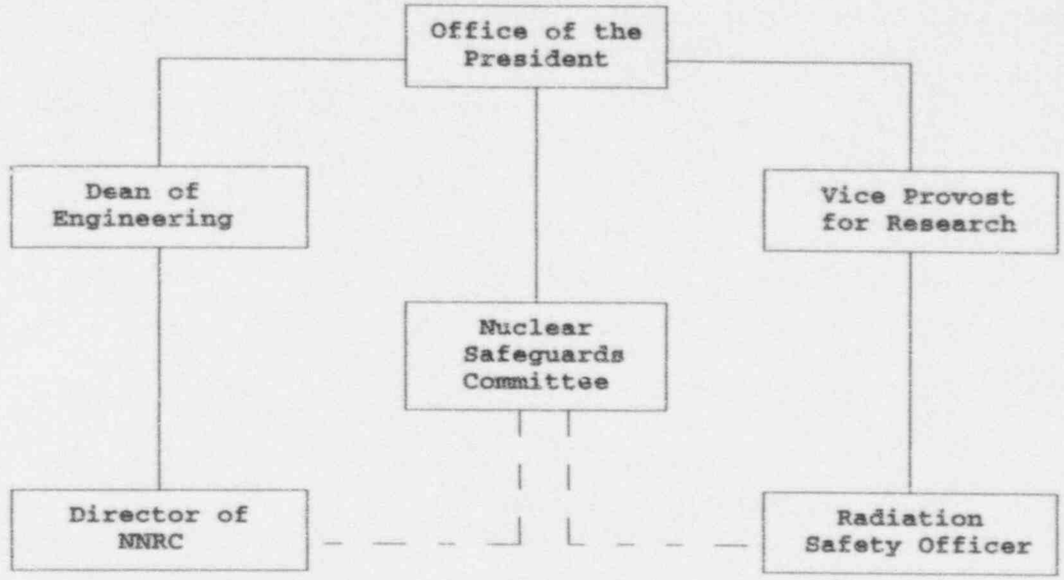


Figure 1. Derived from Tsoulfanidis, ff. Tr. 1939, at Exh. GT-2.

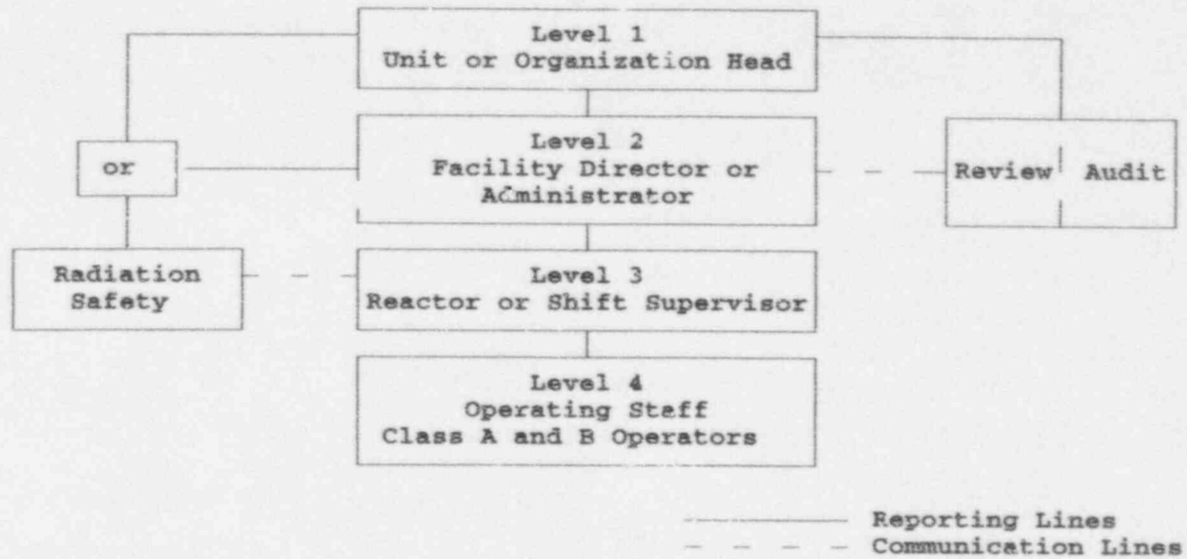


Figure 2. ANSI/ANS-15.1-1990, derived from Attachment 2 to Testimony of Staff Panel C, ff. Tr. 3171.

aged research reactors to follow them, at least in general outline. The two witnesses who comprised the Staff's Panel C, which dealt with this subject, were Messrs. Alexander A. Adams and Marvin M. Mendonca, former and current project managers for the GTRR.

Mr. Adams serves as the NRC's alternate representative to American Nuclear Society (ANS) Consensus Committee N-17, "Research Reactors, Reactor Physics and Radiation Shielding," is the NRC's representative to ANS subcommittee ANS-15, "Operation of Research Reactors," and represents the NRC on the working group for several individual American National Standards Institute (ANSI)/ANS standards pertaining to research reactors, including the working group for ANSI/ANS-15.1, "The Development of Technical Specifications for Research Reactors," which includes guidance on organizational issues. For his part, Mr. Mendonca has conducted training courses on research reactor inspection and regulation issues related, *inter alia*, to organizational, review, and audit functions, and serves as the NRC's representative on various standards committees associated with research reactors. Panel C, ff. Tr. 3171, at 1-6, 9, 12. We find Messrs. Adams and Mendonca to be well qualified to address the differing management structures in use at research reactors and the adequacy of the management structure currently used by Georgia Tech.

Under the 1987-1988 reorganization, Georgia Tech abolished the Office of Radiological Safety and established a new Office of Radiation Safety as a unit of the NNRC. Mr. Robert M. Boyd (the former RSO) became the MORS and commenced reporting to the facility director, Dr. Karam, as did operational personnel. In turn, the organization chart indicated the Director would report to the Vice President for Research, who would report to the President. At the same time, Dr. Bourne (the interim President) appointed Dr. Kahn to serve as the Chairman of the new Nuclear Safeguards Committee (NSC), which replaced two former committees (Nuclear Safeguards and Radiation Protection). Staff Panel C, ff. Tr. 3171, at 12-13; Tr. 2178, 2215 (Boyd).

In addition, Georgia Tech requested changes to the Technical Specifications for the NSC, including changes in the requirements for membership, quorum, areas of expertise, maximum number of members permitted to be from the GTRR staff, and the scope of the NSC's review and approval responsibilities. The proposal showed that the NSC (with the NSC Chairman also holding the title of RSO) would report to the NNRC Director, with communication to the Office of the President. Staff Panel C, ff. Tr. 3171, at 12-14.[34]

[34] An organizational flow chart prepared at that time showed arrows leading to Dr. Karam (the Director) from the NSC, the MORS, and the President, creating the impression that the President and NSC would henceforth report to Dr. Karam. Tr. 2484-85 (Boyd). The flow chart's indication that the NSC and President would report to Dr. Karam was disapproved by the NRC Staff, and was revised by the University President. The unrevised version was also adversely commented upon by Mr. Boyd in this proceeding. Tr. 2484-85.

The NRC Staff performed an initial review of the amendment request after it was submitted, and found certain aspects of Georgia Tech's proposal to be problematic; the Staff then communicated several questions to the Applicant. *Id.* at 14. The more significant issues related to the proposal's failure to conform to the recommendations contained in ANSI/ANS-15.1,[35] by (1) having the NSC report to the facility Director rather than to level 1 management, (2) providing too few review and audit functions for the NSC, (3) not specifying the minimum number of NSC members, and (4) not prohibiting NNRC staff members from being a majority of the required quorum of the NSC. *Id.* at 14-15.

The Applicant then submitted a revised organizational chart for the GTRR TS, which addressed the Staff's questions. In the revised organization, the NSC would report to level 1 management (Office of the President) and would communicate with the NNRC Director. Also, the MORS would report to the NNRC Director for supervision and administrative reporting but would report to the NSC on safety and safety policy matters. *Id.* at 15.[36] In addition, the Applicant revised its proposed amendment to expand the scope of the review and audit responsibilities of the NSC to activities generally suggested by ANSI/ANS-15.1, and it withdrew its proposal to delete the requirement that no more than a minority of the NSC members would be from the GTRR staff. *Id.* at 15-17, 18.

The management structure adopted following the reorganization in 1987-1988, and similar to that currently in place at the GTRR, is similar to the second model, with the MORS reporting directly to the Director of the GTRR, although also reporting safety concerns to the Nuclear Safeguards Committee (NSC). As set forth in Figure 3, p. 307, *infra.*

According to the Staff, both organizational forms work, with about 35% of research reactors having the radiation safety functions reporting directly to the facility director (like the GTRR) and the others reporting either to a higher level or to a different chain of command. Tr. 3175 (Mendonca).

### c. *GANE's Challenge to the Structure*

GANE claims that, under a structure where the MORS reports directly to the Director, (1) the MORS lacks sufficient independence to conduct his duties,

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[35] *See* discussion at p. 310, *infra.*

[36] Mr. Boyd similarly noted that certain aspects of the July reorganization were clarified by the University President in February 1988, in a memorandum and general faculty meeting. First, the President indicated that the NSC was to report to the University President; second, the MORS was responsible, under a revised organizational chart, to report safety problems to the NSC (as well as to the facility director) — and if the MORS was not satisfied with how safety problems were being treated by others, he was to inform the President or Vice President for Research of those matters. GANE Exh. 47, at 1; GANE Exh. 46. This latter statement responded to Mr. Boyd's concern that his reporting line to the NSC had been eliminated under the July reorganization. Tr. 2259, 2277, 2403-06, 2410-11 (Boyd); GANE Exhs. 46, 47.

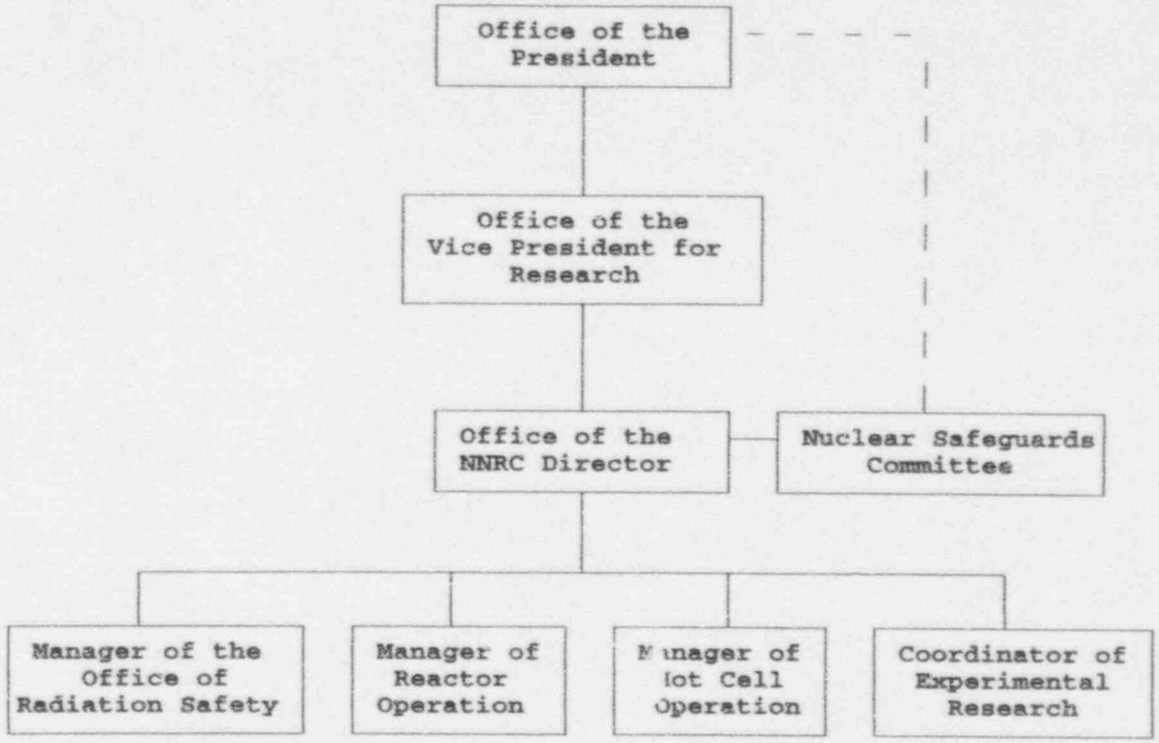


Figure 3. Derived from GT Exh. 6, NNRC Reorganization Chart.

(2) the NSC has an inadequate concern for safety, and (3) too much authority is concentrated in the Director (currently Dr. Karam). GANE in particular relies for these claims upon two of its witnesses who had been former radiation safety officers at the GTRR — Dr. Brian Copcutt and Mr. Robert Boyd. But in support of the superiority of an organization that has separate chains of command for the director and the radiation safety officer, GANE also points to the opinions of Dr. Rodney Ice, the current MORS, and Dr. Nicholas Tsoulfanidis, an expert witness presented by Georgia Tech.

Specifically, Dr. Copcutt served as MORS from July 1990 to November 2, 1990 (GANE Exhs. 1, 13). His letter of resignation to Dr. Karam, dated October 8, 1990 (GANE Exh. 13), cited extensively by GANE (GANE FOF at 4), states that it is "impossible for me to work effectively within the structure of the radiation safety program at Georgia Tech." Dr. Copcutt goes on to state in the letter that the MORS "lacks sufficient operational freedom to adequately conduct the radiation safety program" and that the health physics staff (which nominally reports to the MORS) appears to be "under the dual control" of the MORS and the facility Associate Director. He concludes that "I cannot, in good conscience, take responsibility for a program whose priorities I cannot set and in which I must compromise my professional judgments."[37]

Mr. Robert M. Boyd, who served as Radiological Safety Officer at Georgia Tech from 1973 until the reorganization in 1987, as MORS from 1987 to 1988, and who served (simultaneously) as Radiological Safety Officer at Georgia State University from 1973 until his retirement in 1995 (Boyd, Professional Experience, ff. Tr. 2122, at 1-2), even more strongly stressed in his testimony the superiority of dual reporting chains. He characterized the current form of organization, with the MORS reporting to the facility Director, as "the fox guarding the hen house" and called the decision to change to such a structure "a mistake — it was a mistake in my view, improper" (Tr. 2175 (Boyd)).

Mr. Boyd conceded, however, that the management structure in place was "not so serious as to say that the safety of the public cannot be assured" (Tr. 2396 (Boyd)). He added that he "did not consider the present organizational structure to constitute an immediate health hazard" (*id.*).

Dr. Ice, who has been MORS since 1992, with over 29 years of practical experience and published research in health physics and who is a health physicist and a teacher and advisor on radiation safety (Ice, ff. 1992, at 2, 5), also favored having the MORS not subject to the supervisory control of the Director. He explained:

[37] The letter also objects to alleged suggestions from the Director and Associate Director that he should not, in the future, "document observed regulatory violations or proposed program improvements." We have dealt with these allegations elsewhere in this Decision.

I think in an effective organization for radiation safety, executive management should be involved in the oversight in the scenario, so I think there should be a clear path between the radiation safety officer and executive management. . . . organizationally, and from an operational standpoint, I would love to see a cleaner relationship between safety and operations, a pure distinction between the two.

Tr. 2000-01 (Ice).

Finally, Dr. Tsoulfanidis, a consultant for the Applicant and, since 1975, the Radiation Safety Officer for the University of Missouri–Rolla (where he also serves as a professor of Nuclear Engineering and the Assistant Dean for Research in the School of Mines and Metallurgy (Tsoulfanidis, ff. Tr. 1939, at 2)), expressed the view that the present administrative structure of the Radiation Safety Program "seems to work fine and there is no evidence of any kind that safety is compromised." He recommended a structure with dual lines of authority (set forth as Figure 1, above) for the following reasons:

[T]he present reporting method has the potential for errors, omissions and abuse, particularly if the current Director is replaced and the new one is not so safety conscious. . . . There is no evidence that the current Director either made mistakes or abused the system. However, whenever a program or activity is controlled by a single person the possibility of error or omission of action is possible.

Tsoulfanidis, ff. Tr. 1939, Exh. GT-2, at 6. Dr. Tsoulfanidis stressed that separate budgets should be set up for the Director (for operational purposes) and for the RSO. *Id.* at 7.

*d. Other Parties' Positions*

The Applicant strongly favors the current organizational structure, where the MGRS reports to the Director. Dr. Karam, who was appointed Director on December 5, 1983 (prior to the reorganization), expressed his belief that inasmuch as his responsibilities as Director covered overall operation of the reactor (including radiation safety), and inasmuch as the radiation safety staff did not report administratively to him but operated independently, he was extremely uncomfortable about being held responsible for the work of a unit over which he had virtually no control. He also believed that he could better deal with the hostilities between HP and operations personnel if he had managerial control over both. Karam, ff. Tr. 2723, at 24-25; Tr. 2769 (Karam).

Thus, prior to the reorganization, the manager of the safety unit nominally reported to Vice President Stelson, to whom Dr. Karam also reported. But in actual practice, the manager of the HP unit (Mr. Boyd) was instructed "to run that thing and don't bother [Dr. Stelson]". He was "essentially unsupervised by anybody" (Tr. 2366-67 (Boyd)). Mr. Boyd added, however, that he felt the Chairman of the then Radiation Protection Committee and the Chairman of the Nuclear Safeguards Committee were "essentially [his] boss as far as safety concerns" (Tr. 2367-68 (Boyd)).

Prior to the reorganization, there had been extreme hostility between the health physics and operational staffs. This history of hostility, which among other things led to a shutdown of reactor operations by NRC, is reviewed in greater detail earlier in this opinion. One of the purposes of the reorganization where the RSO reports directly to the Director was to lessen the hostility. Initially following the reorganization the hostility actually increased. Thereafter, Dr. Karam replaced the entire health physics staff with persons with greater academic qualifications. The end result, according to Dr. Karam, was a better-qualified health physics staff and a diminution of the hostility between the two groups. As a result, Dr. Karam strongly supported the existing chains of command.

The Staff would have found either method of organization equally acceptable — both are sanctioned by the ANSI standards, and either would be acceptable under NRC regulations (Tr. 3175, 3182-83 (Adams, Mendonca)). "[E]ither can work." Tr. 1895 (Gibson); Tr. 1894-95 (Collins). But the Staff appeared to prefer the current form of organization on the basis of its success at the GTRR in terms of resulting in fewer and less severe violations than the previous unacceptable level that in part caused the Staff to have the reactor shut down.

### 4. *Licensing Board Conclusions*

Having carefully considered the various views of organizational format expressed by witnesses of all parties, we conclude that, in our opinion, the separation of functions inherent in having the MORS and other health physics personnel report to a person other than the operational director of the facility would be preferable to having him or her report to the Director, as is currently the practice at the GTRR. Because either form of organization is legally acceptable, however, we would need a strong record establishing the performance superiority of separate reporting chains (and safety deficiencies attributable to a single reporting chain) in order for us to mandate such a change for the GTRR.

Such a record is not here present. Even witnesses who favored the separate chains of command indicated that the present system at GTRR presents no threat to the public health and safety. Part of the rationale for this view stemmed from those witnesses' knowledge of the technical competence and dedication of the current Director, Dr. Ratib Karam. Dr. Karam is planning to retire within the next few months, however, effective June 30, 1997 (Tr. 2709-10, 3404 (Karam)). When that happens, Georgia Tech may wish to consider what organizational format it will utilize. But we will impose no license condition requiring any modification.

Apart from organizational format, GANE also seeks to deny license renewal on the basis of a continuing series of regulatory violations. The most serious occurred before (and in part caused) the reactor shutdown in 1988. Since restart, the numbers of violations/year has been decreasing over the years (Tr. 3149-50, 3151 (Mendonca, McAlpine, Bassett)), and none has been found by the Staff to be more serious than Severity Level IV. We decide herein whether the GTRR license renewal application should be denied or conditioned on the basis of events and violations of that severity cited by GANE from NRC inspection reports.

At the time of those citations, NRC's enforcement policy in 10 C.F.R. Part 2, Appendix C, defined Severity Level IV violations as of "more than minor concern, i.e., if left uncorrected they could lead to a more serious concern."[38] Table 2 of the enforcement policy indicates Commission policy to consider license suspension or revocation only for more serious violations at Severity Levels III, II, or I.[39] There is no indication in the enforcement policy (either that in effect in early 1995 or at present) that the Commission would suspend, revoke, or deny a license to operate on the basis of several Severity Level IV violations.

It is evident from the policy that the appropriate sanction for Severity Level IV violations is for the Applicant to be required to correct the cited deficiencies. The NRC Staff is now satisfied that Georgia Tech has recovered from management deficiencies of the past and that its performance now is generally satisfactory. Thus, although GANE calls for the Board to refuse to authorize license renewal on the basis of several Severity Level IV violations, we decline to do so. Under all but the most exceptional circumstances not relevant here, Severity Level IV violations do not rise to the level of significance that would place license renewal in jeopardy. GANE may well hold the view that reactor licensees should be held by the NRC to a standard of error-free performance. Although conceptually appealing, that is not the regulatory scheme. As evident from the enforcement policy, NRC takes account of the severity of violations and not just their occurrence when it decides what enforcement action to take.

One further matter warrants some brief comment. In its findings of fact, GANE claims that "Georgia Tech has denied GANE the respect due to ordinary citizens who are simply exercising their democratic right to due process. Up to and including their latest submission [i.e., Georgia Tech's proposed findings],

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[38] *See* note 10, *supra*, for a definition of each of the severity levels in effect at the time of the citations. Effective June 30, 1995, the Enforcement Policy was removed from 10 C.F.R. Part 2 and published as NUREG-1600. 60 Fed. Reg. 34,380 (June 30, 1995). At the time, Severity Level V violations were eliminated. *Id.* at 34,381.

[39] The NRC is authorized under the Atomic Energy Act to revoke licenses under the same conditions that would have warranted refusal of a license on an original application. 10 C.F.R. Part 2, Appendix C, § II (1995 ed.); NUREG-1600 § VI.C(e). The Board would only refuse to authorize a renewed license under the enforcement policy for reasons that were as serious as those that could lead to revocation.

we have been treated as a nuisance not worthy of their time and this attitude is not only rude, it does not speak well of the nuclear industry's willingness to act in good faith as a community citizen." GANE FOF at 3.

GANE provides no specific references to this alleged treatment, and our examination of Georgia Tech's findings of fact does not reveal any such disrespect. Suffice it to say, however, that this Board views GANE's efforts in this proceeding with great respect. Even though GANE did not succeed in its efforts to deny renewal of the Applicant's license, or to require a different management organization, it brought to light many aspects of Georgia Tech's operation that could lead to an operation in the future providing enhanced protection to the publ health and safety. GANE's efforts therefore deserve commendation.

## D. Conclusions of Law

The Licensing Board has considered all of the evidence presented by the parties on the admitted contention concerning the adequacy of the Applicant's management of the Georgia Tech Research Reactor. Based upon a review of the entire record in this proceeding and the proposed findings of fact and conclusions of law submitted by the parties, and based upon the findings of fact set forth herein, which are supported by reliable, probative, and substantial evidence in the record, the Board has decided all matters in controversy pertinent to management of the GTRR and reaches the following conclusions:

1. The Applicant's performance in the post-restart period, although not entirely satisfactory, has substantially improved since the shutdown of the reactor in 1988. Further, Georgia Tech's performance in the post-restart period does not support GANE's assertion that management of the GTRR is inadequate and that the license renewal application should therefore be denied. Nor has GANE met its burden of demonstrating that "substantial management deficiencies persist." LBP-95-6, 41 NRC 281, 299 (1995).

2. The Board has further examined the evidence in light of the guidance provided by the Commission at the start of this proceeding. We conclude that GANE has not demonstrated "management improprieties or poor 'integrity' . . . [that] relate directly to the proposed licensing action," or that "the GTRR as presently organized and staffed [fails to] provide reasonable assurance of candor and willingness to follow NRC regulations." Moreover, the evidence supports findings that "the facility's current management encourages a safety-conscious attitude, and provides an environment in which employees feel they can freely voice safety concerns," and there is "reasonable assurance that the GTRR facility can be safely operated" in that "the GTRR's current management [n]either is unfit [n]or structured unacceptably." CLI-95-12, 42 NRC 111, 120-21 (1995).

3. The Applicant's management of the Georgia Tech Research Reactor complies with all applicable regulatory requirements, and provides reasonable assurance that its management of the GTRR facility, upon the renewal of License No. R-97, will not be inimical to the common defense and security or to the health and safety of the public.

4. All issues, arguments, or proposed findings presented by the parties but not addressed herein have been found to be without merit or unnecessary for this Decision.

**E. Order**

1. Pursuant to 10 C.F.R. §§ 2.760 and 50.57, as applicable, the Director, Office of Nuclear Reactor Regulation, hereby is *authorized* to issue to the Georgia Institute of Technology, upon making requisite findings with respect to matters not embraced within this Initial Decision, a renewal of Operating License No. R-97, in accordance with Georgia Tech's application for such license renewal.

2. This Initial Decision *shall become effective* and constitute the final action of the Commission forty (40) days after the date of its issuance, subject to any review pursuant to the Commission's regulations.

3. In accordance with 10 C.F.R. § 2.786, any petition for review of this Initial Decision must be filed within fifteen (15) days after service of the Decision. Any other party may file, within ten (10) days after service of a petition for review, an answer in support of, or in opposition to, the petition for review. The petition for review may be granted or denied in the discretion of the Commission, giving weight to the considerations of 10 C.F.R. § 2.786(b)(4).

THE ATOMIC SAFETY AND
LICENSING BOARD

Charles Bechhoefer, Chairman
ADMINISTRATIVE JUDGE

Dr. Jerry R. Kline
ADMINISTRATIVE JUDGE

Dr. Peter S. Lam
ADMINISTRATIVE JUDGE

Rockville, Maryland
April 3, 1997

# Directors' Decisions Under 10 CFR 2.206

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

## OFFICE OF NUCLEAR REACTOR REGULATION

**Samuel J. Collins**, Director

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

**GENERAL PUBLIC UTILITIES NUCLEAR CORPORATION (Oyster Creek Nuclear Generating Station)** **April 2, 1997**

By a petition dated September 19, 1994, Reactor Watchdog Project Nuclear Information and Resource Service, and Oyster Creek Nuclear Watch (Petitioners) requested that the NRC take action with regard to Oyster Creek Nuclear Generating Station (OCNGS) operated by GPU Nuclear Corporation (GPU or Licensee). Petitioners requested that the NRC (1) immediately suspend the OCNGS operating license until the Licensee inspects and repairs or replaces all safety-class reactor internal component parts subject to embrittlement and cracking, (2) immediately suspend the OCNGS operating license until the Licensee submits an analysis regarding the synergistic effects of through-wall cracking of multiple safety-class components, (3) immediately suspend the OCNGS operating license until the Licensee has analyzed and mitigated any area of noncompliance with regard to irradiated fuel pool cooling as a single-unit boiling water reactor (BWR), and (4) issue a generic letter requiring other licensees of single-unit BWRs to submit information regarding fuel pool boiling in order to verify compliance with regulatory requirements and to promptly take appropriate mitigative action if the unit is not in compliance. By a letter dated December 13, 1994, Petitioners supplemented their petition and requested that the NRC: (1) suspend the OCNGS operating license until Petitioners' concerns regarding cracking are addressed including inspection of all reactor vessel internal components and other safety-related systems susceptible to intergranular stress-corrosion cracking and completion of any and all necessary repairs and modifications; (2) explain the discrepancies between the response of the NRC

Staff dated October 27, 1994, to the petition and time-to-boil calculations for the FitzPatrick Plant; (3) require GPU to produce documents for evaluation of the time-to-boil calculations for the OCNGS irradiated fuel pool; (4) identify redundant components that may be powered from onsite power supplies to be used for spent fuel pool cooling as qualified Class 1E systems; (5) hold a public meeting in Toms River, New Jersey, to permit presentation of additional information related to the petition; and (6) treat Petitioners' letter of December 13, 1994, as a formal appeal of the denial of their request of September 19, 1994, to immediately suspend the OCNGS operating license.

By letter dated October 27, 1994, the Director denied Petitioners' request for immediate suspension of the OCNGS operating license. By letter dated April 10, 1995, the Director denied requests (5) and (6) of the December 13, 1994 Supplemental Petition. On August 4, 1995, the Director issued a Partial Director's Decision (DD-95-18, 42 NRC 67) denying requests (1) and (2) of the September 19, 1994 Petition and request (1) of the December 13, 1994 Supplemental Petition.

By a Director's Decision issued on April 2, 1997, the Director granted in part requests (3) (exclusive of the request to suspend OCNGS operating license was previously denied) and (4) of the September 19, 1994 Petition, and granted requests (2), (3), and (4) of the December 13, 1994 Supplemental Petition.

## 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 September 19, 1994 (petition), Reactor Watchdog Project, Nuclear Information and Resource Service, and Oyster Creek Nuclear Watch (Petitioners) requested that the U.S. Nuclear Regulatory Commission (NRC) take immediate action with regard to Oyster Creek Nuclear Generating Station (OCNGS) operated by GPU Nuclear Corporation (GPU or Licensee). By letter dated December 13, 1994, Petitioners supplemented the petition.

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

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

On October 27, 1994, the Director of the Office of Nuclear Reactor Regulation informed the Petitioners that he was denying their request for immediate suspension of the OCNGS operating license, that their petition was being evaluated under section 2.206 of the Commission's regulations, and that action would be taken in a reasonable time. By letter dated April 10, 1995, the Director denied requests (5) and (6) of Petitioner's supplemental petition. On August 4, 1995, the Director issued a Partial Director's Decision (DD-95-18, 42 NRC 67), denying requests (1) and (2) of their Petition of September 19, 1994, and request (1) of the Supplemental Petition of December 13, 1994. A decision regarding requests (3) and (4) of the Petition of September 19, 1994, and requests (2), (3), and (4) of the Supplemental Petition of December 13, 1994, was deferred pending completion of our review.

The NRC Staff's review of the petition and supplemental petition is now complete. For the reasons set forth below, requests (3), with the exception of suspending OCNGS operating license which was previously denied, and (4) of the Petition of September 19, 1994, are granted in part and requests (2), (3), and (4) of the Supplemental Petition of December 13, 1994, are granted as described below.

## II. BACKGROUND

On November 27, 1992, a report was filed pursuant to 10 C.F.R. Part 21 by two contract engineers that notified the Commission of potential design deficiencies in spent fuel pool decay heat removal systems and containment systems at Susquehanna Steam Electric Station (SSES). 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 as a result of concurrent plant conditions it would not be possible for operators to place backup systems in service or that backup systems would otherwise be unable to perform their intended function. The report concluded 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 these conditions could be the failure (meltdown) 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 public health and safety.

Although the issues raised by this Part 21 report appeared to be of low safety significance, because of the low probability that the necessary sequence of events would take place,[1] the complex nature of the issues prompted the NRC Staff to undertake an extensive evaluation of the matter. The NRC Staff review process, which continued from November 1992 to June 1995, included information-gathering trips to the Licensee's engineering offices and to 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 (IN) 93-83, "Potential Loss of Spent Fuel Pool Cooling After a Loss-of-Coolant Accident or a Loss of Offsite Power," 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 issued a draft safety evaluation (SE) addressing the issues raised in the Part 21 report on 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 SE

[1] 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.

regarding the issues raised in the Part 21 report for the SSES on June 19, 1995 (SSES SE).[2]

The NRC Staff reviewed and evaluated the SSES plant design and inspected operation of SSES plant equipment with respect to the various event sequences described in the Part 21 report. The Staff also evaluated the response of SSES 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 a loss of offsite power events, from seismic events, and from flooding events. The Staff considered the safety significance of such events potentially leading 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. The Staff evaluated the safety significance of the issues as they pertained to the plant at the time the Part 21 report was submitted and as they pertained to the plant after the completion of certain voluntary modifications made at SSES during the course of the NRC Staff's review. Finally, the Staff examined licensing issues associated with the design of the spent fuel pool cooling system to determine the extent to which SSES's design and operation met the applicable regulatory requirements.

On the basis of the Staff's 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 its 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 Staff's evaluation concluded that even with consideration of the additional initiating events previously described, "loss of spent fuel pool cooling events" represented a challenge of low safety significance to the plant 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 removing 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

[2] Letter to R. Byram, Pennsylvania Power & Light Company, 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.

of spent fuel pool cooling events, the SSES SE concluded that the design of the SSES facility was adequate to protect public health and safety.

With regard to licensing-basis design issues, the Staff concluded that only a loss of spent fuel pool cooling initiated by a seismic event was considered in the original granting of the SSES license by the NRC.

The Staff issued IN 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, describing the conclusions of the June 19, 1995, SSES SE. The information notice described the Staff's plans to implement a generic 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, and (4) selection and execution of an appropriate course of action based on the safety significance of the findings.

As part of the Task Action Plan review, the Staff reviewed operating experience, as documented in licensee event reports and other information sources, as well as in previous studies of spent fuel pool issues. The Staff also gathered detailed design data relating to the design basis and functional capability of the fuel storage pool, the fuel pool cooling system, and other systems associated with fuel storage for every operating reactor and analyzed these data to identify potential safety issues regarding a loss of spent fuel pool cooling or a loss of coolant inventory.

The NRC Staff forwarded the results of its Task Action Plan review to the Commission on July 26, 1996.[4] The Staff concluded that existing spent fuel storage pool structures, systems, and components provide adequate protection of public health and safety at all operating reactors. Protection is provided by several layers of defenses that perform accident prevention functions (e.g., quality controls on design, construction, and operation), accident mitigation

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[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 as a result of 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 NRC 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 that 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.

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

functions (e.g., multiple cooling systems and multiple makeup water paths), radiation protection functions, and emergency preparedness functions. 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 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 these findings, the NRC Staff reviewed the design of every operating reactor's spent fuel pool to identify strengths and weaknesses and 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; OCNGS was not one of the 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 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. OCNGS was not one of the eleven sites. The Staff has insufficient information at this time to determine whether backfits pursuant to section 50.109(a)(3) are warranted at the eleven sites. 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 informed the Commission in the July 26, 1996 Task Action Plan report that it plans to address issues related 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, in a new proposed performance-based rule for shutdown operations (10 C.F.R. § 50.67) at *all* operating reactors. The new rule is scheduled to be issued for public comment in 1997.

The NRC Staff sent the Task Action Plan report of July 26, 1996, to all operating power reactor licensees. For those licensees whose plants have one or more of the design features that warrant a plant-specific safety enhancement backfit analysis, the Staff has provided an opportunity 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, and (4) the existing protection from the design concern provided by administrative controls or other means. In developing a schedule and plans for conducting all of the plant-specific regulatory analyses, the NRC Staff will consider comments received from licensees.

## III. DISCUSSION

### A. Issuance of Generic Letter, Compliance Verification, and Mitigative Action (September 19, 1994 Petition Items (3) and (4))

The Petitioners requested (Items (3) and (4) of the September 19, 1994 Petition) that the NRC immediately suspend the OCNGS operating license until GPU analyzes and mitigates any areas of noncompliance with regard to irradiated fuel pool cooling as a single-unit boiling water reactor, and that the NRC issue a generic letter requiring other licensees of single-unit BWRs to submit information regarding fuel pool boiling in order to verify compliance with NRC requirements and to take quick mitigative action if the unit is not in compliance.

As stated in the cover letter, the October 27, 1994 Director's letter informed you that he denied your request for immediate suspension of the OCNGS operating license.

While the NRC has not issued and does not plan to issue a generic letter, the Staff has communicated the importance of conducting relevant spent fuel pool decay heat removal activities in accordance with technical specifications and other plant-specific applicable regulatory requirements to licensees through the issuance of other generic communications, as described below. The Staff also surveyed all operating reactor licensees, including GPU Nuclear Corporation, Licensee for OCNGS, to collect information on, among other things, parameters affecting boiling of the spent fuel pool. Results of the survey relevant to this petition are discussed below.

The NRC Staff issued three information notices on matters related to adequate removal of decay heat from the spent fuel pool. IN 93-83, "Potential Loss of Spent Fuel Pool Cooling After a Loss-of-Coolant Accident or a Loss of Offsite Power," was issued on October 7, 1993, and described the concerns in the November 27, 1992 SSES Part 21 report discussed above. IN 93-83, Supplement 1, "Potential Loss of Spent Fuel Pool Cooling After a Loss-of-

Coolant Accident or a Loss of Offsite Power," issued on August 8, 1995, informed licensees of the results of the NRC's review of the concerns at SSES. IN 95-54, "Decay Heat Management Practices During Refueling Outages," was issued on December 1, 1995, and described recent NRC assessments of events at certain plants regarding the Licensee's control of refueling operations and the methods for removing decay heat produced by the irradiated fuel stored in the spent fuel pool during refueling outages. IN 95-54 communicated to licensees that the plant-specific events described therein and in the previous information notices illustrated the importance of ensuring that (1) planned core offload evolutions, including refueling practices and irradiated fuel decay heat removal, are consistent with the licensing basis, including the final safety analysis report, technical specifications, and license conditions; (2) changes to these evolutions are evaluated through the application of the provisions of 10 C.F.R. § 50.59, as appropriate; and (3) all relevant procedures associated with core offloads have been appropriately reviewed.

The Staff surveyed operating reactors, including Oyster Creek, as part of the (a) Spent Fuel Pool (SFP) Task Action Plan, and (b) followup actions related to issues identified at Millstone, and reviewed the degree to which fuel pool operations compared with each facility's design basis and the degree that the fuel pool design features conformed with accepted guidance and standards. In the case of Oyster Creek, the NRC Staff found no deviations in operation or design as a result of either review. The Staff issued its report on the results of spent fuel pool survey regarding Millstone followup issues on May 21, 1996. As described in Section II of this Decision, the NRC Staff forwarded its report on the resolution of the SFP Task Action Plan on July 25, 1996, to all operating power reactor licensees.

As part of the SFP 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 actions. The Staff found that SFP-related regulatory requirements have been evolving since the first nuclear power plants were licensed and that specific regulatory guidance on the design of spent fuel pool cooling systems was not formalized until 1975, when the Standard Review Plan was issued, which was after the issuance of construction permits for most currently operating reactors. Because the regulatory requirements were evolving during the era in which the Staff was conducting licensing reviews for the current generations of operating reactors, Staff-approved designs varied from plant to plant. However, based on the recent survey results, the Staff concluded that all operating reactors had design features for spent fuel storage (e.g., addressing accident prevention functions, accident mitigation functions, radiation protection functions, and emergency preparedness functions), which had been reviewed and approved in the past by the NRC. In addition, based on the review of the survey results, the Staff found that all licensees were in compliance with current NRC requirements.

Although the NRC Staff concluded that all plants, including OCNGS, are in compliance with the NRC spent fuel pool design requirements, the Staff reviewed certain operating practices at all operating reactor plants to verify that the plants were being operated consistent with the plant design as described in the licensing basis,[5] specifically with respect to refueling outage practices associated with 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, that core offload practices are consistent with the spent fuel pool decay heat removal licensing basis for all plants, or will be before the next refueling outage. It should be noted, however, that during the course of its review, the Staff determined that nine sites (involving fifteen units) needed to modify their licensing bases or plant practices, pursuant to 10 C.F.R. § 50.59 or 10 C.F.R. § 50.90, to ensure that their refueling 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 is reviewing potential enforcement action for these facilities. It should be noted that OCNGS is *not* one of the nine sites.

The Petitioners requested that the NRC immediately suspend the OCNGS operating license until GPU analyzes and mitigates any areas of noncompliance with regard to irradiated fuel pool cooling as a single-unit BWR, and that the NRC issue a generic letter requiring other licensees of single-unit BWRs to submit information regarding fuel pool boiling in order to verify compliance with NRC requirements and take quick mitigative action if the unit is not in compliance. These requests are granted in part as described above. Petitioners' request for immediate suspension of OCNGS operating license was previously denied.

### B. Time-to-Boil Calculations (December 13, 1994 Supplemental Petition Items (2) and (3))

Petitioners' supplementary request of December 13, 1994, asked the NRC to explain "discrepancies" between the response of the NRC Staff dated October 27, 1994, to the petition and the documented time-to-boil calculations for the FitzPatrick Plant as they bear on time-to-boil calculations for other single-unit General Electric BWRs, including OCNGS. Petitioners contend that documents available in the Public Document Room for FitzPatrick Plant, a single-unit

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

site, indicated a time-to-boil following a loss-of-coolant accident of 8 hours, considerably less than the 25 hours SSES, a dual-unit site, committed to in a letter dated June 1, 1994. Petitioners also requested that the Licensee, GPUN, produce time-to-boil calculations for OCNGS.

The NRC Staff letter of October 27, 1994, to Petitioners concluded that time-to-boil conditions at single-unit BWR sites, such as OCNGS, are of low safety significance because, unlike dual-unit sites, such as SSES, a large decay heat rate associated with a short time to reach boiling conditions is an unrealistic assumption during periods when the unit is operating and fuel in the reactor vessel is subject to a loss-of-coolant accident.

As explained in the Director's letter to Petitioners dated April 10, 1995, the time-to-boil calculation results for the FitzPatrick Plant single-unit BWR, which were presented in a New York Power Authority document dated May 31, 1990, were based on the maximum postulated decay heat rates during a refueling outage fuel discharge and full core offload that occurred about 7 and 10 days, respectively, after reactor shutdown. These calculations also assumed that spent fuel pool cooling was lost when the pool was at its maximum calculated temperature. In contrast, the Staff calculated the time-to-boil for FitzPatrick to be 25 hours for a one-third core discharge 30 days after reactor shutdown, assuming the spent fuel pool was at its maximum temperature limit for normal operation, which is 125°F. The details of this calculation were provided in our Director's letter to you dated April 10, 1995. Additionally, the Staff had surveyed the factors that would most significantly affect the time-to-boil (i.e., spent fuel pool volumes, rated reactor thermal power level, total number of fuel assemblies in the reactor vessel, and spent fuel pool temperature limits) for twelve General Electric Company BWR/3 and BWR/4 reactors. The Staff concluded that its time-to-boil calculations for FitzPatrick are representative for United States single-unit BWRs as a whole, and OCNGS in particular.

As part of the NRC Staff's Task Action Plan activities, the Staff collected information from Licensee documents to calculate the time-to-boil for all operating reactors on a consistent basis. While the Staff did not specifically require licensees (including GPU) to provide documentation to support time-to-boil calculations, the Staff did independently calculate the time-to-boil for each plant from Licensee-supplied information in Final Safety Analysis Reports and other design documents. On this basis, the Staff determined that the time-to-boil at Oyster Creek is average among single-unit BWRs, thus confirming the same conclusion reached earlier in the Director's letter of April 10, 1995.

Accordingly, the Petitioners' requests to explain the "discrepancies" between the response of the NRC Staff dated October 27, 1994, to the petition and the documented time-to-boil calculations for the FitzPatrick Plant as they bear on time-to-boil calculations for other single-unit General Electric BWRs, including

OCNGS, and that GPU produce documents for evaluation of time-to-boil calculations are granted as described above.

### C. Redundant Class 1E Components and Power Supplies (December 13, 1994 Supplemental Petition Item (4))

In the supplemental petition submittal of December 13, 1994, the Petitioners requested that the NRC identify redundant components that may be powered from onsite power supplies to be used for spent fuel pool cooling as qualified Class 1E systems at Oyster Creek.

The Petitioners noted that while Oyster Creek may have redundant components, in their view it is meaningless to have redundant components and power supplies if they have not been qualified to operate under emergency conditions.

At Oyster Creek, spent fuel decay heat removal consists of a two-train spent fuel pool cooling system. The first train ("Spent Fuel Pool Cooling System") has two pumps and two heat exchangers. The second or augmented train, installed in parallel with the first train, contains two full-capacity pumps and a single heat exchanger. The four pumps in both trains are powered from electrical buses supported by safety-related emergency diesels (MCCs 1A21, 1A23, 1B21, and 1B23). The augmented train is seismically qualified. Portions of the spent fuel pool cooling system initially designed to be a nonseismic system, has been upgraded to Seismic Category I requirements. Those portions of the system that do not meet seismic requirements can be isolated from the spent fuel pool cooling system if a seismic event renders them inoperable.

It should be made clear that the NRC Staff does not require Class 1E qualification for spent fuel pool cooling equipment and instrumentation. Class 1E is the safety classification of electric equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or are otherwise essential in preventing significant release of radioactive material to the environment.[6] The spent fuel pool cooling system and monitoring instrumentation are not required for such functions.

In his letter of April 10, 1995, the Director informed Petitioners that they have not presented, nor was the Staff aware of, any evidence that the spent fuel pool cooling system fails to comply with its design basis, or that the Licensee failed to qualify these components to the degree Petitioners describe such that it would alter his decision as it pertains to the safety significance of these issues. Therefore, further review of the qualification of spent fuel cooling system components at OCNGS is not warranted. Additionally, Petitioners were

[6] IEEE Std 308-1980.

informed that the Staff would continue its generic review of spent fuel storage pool safety and would take appropriate action based on the conclusions of that review. Based on the results of the generic review of spent fuel storage pool safety thus far, the Staff has concluded that no additional actions are warranted for the spent fuel pool cooling system components at OCNGS.

The Petitioners' request to identify redundant qualified Class 1E systems was granted as described above.

## IV. CONCLUSION

Although the Staff has not initiated formal enforcement proceedings in response to the petition, the Staff has taken a number of actions that address the concerns raised in the petition. For example, during the course of its review, the NRC Staff has issued generic communications responsive to Petitioners' request (4) of September 19, 1994. In addition, the NRC Staff reviewed the compliance of NRC-licensed facilities in the area of spent fuel pool design responsive to Petitioners' request (3) of September 19, 1994. To this extent, the petition is granted in part. Finally, Petitioners' supplemental petition requests (2), (3), and (4) are granted as explained above.

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

Samuel J. Collins, Director
Office of Nuclear Reactor
Regulation

Dated at Rockville, Maryland,
this 2d day of April 1997.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**OFFICE OF NUCLEAR REACTOR REGULATION**

**Samuel J. Collins,** Director

**In the Matter of**

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| **WISCONSIN ELECTRIC POWER COMPANY (Point Beach Nuclear Plant, Units 1 and 2)** | **Docket Nos. 50-266 50-301 72-5** |
| **CONSUMERS POWER COMPANY (Palisades Nuclear Plant)** | **Docket Nos. 50-255 72-7** |
| **ENTERGY OPERATIONS, INC. (Arkansas Nuclear One, Units 1 and 2)** | **Docket Nos. 50-313 50-368 72-13** |
| | **April 17, 1997** |

The Director of the Office of Nuclear Reactor Regulation denies a petition filed pursuant to 10 C.F.R. § 2.206 by Citizen's Utility Board on September 30, 1996, asking the NRC to (1) require the Licensee for Point Beach Nuclear Plant to reserve a fixed number of vacant spaces in the spent fuel pool to permit retrieval from a VSC-24 cask in the event the fuel in the cask must be removed, and (2) to order all users of the VSC-24 cask not to load any casks until the COC, SAR, and SER are amended to contain operating controls and limits to prevent hazardous conditions.

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

### I. INTRODUCTION

On September 30, 1996, Citizens' Utility Board 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 U.S. Nuclear Regulatory Commission (NRC) take the following actions:

1. Order Wisconsin Electric Power Company (WEPCO) to retain 24 empty and available spaces in the Point Beach Nuclear Plant spent fuel pool to provide the capability to permit retrieval of spent fuel from a VSC-24 cask in the event of an accident requiring removal of spent fuel from the cask or in the event that conditions of the certificate of compliance (COC) for the VSC-24 require removal of spent fuel from the cask, until such time that WEPCO has other options available to it to remove spent fuel from a cask in the event conditions warrant it; and
2. Order users of the VSC-24 cask not to load VSC-24 casks until the COC, safety analysis report (SAR), and safety evaluation report (SER) are amended to contain operating controls and limits that prevent hazardous conditions, including but not limited to the generation of explosive gases, due to VSC-24 material reactions with environments encountered during loading, storage, and unloading of the VSC-24 cask. The SAR and SER must be amended such that each operating control and limit is clearly documented and justified in the technical review sections of the SAR and associated SER as necessary and sufficient for safe cask operation.

The petition has been referred to me pursuant to 10 C.F.R. § 2.206. The NRC letters dated October 11 and December 10, 1996, to Mr. Dennis Dums, on behalf of the Petitioner, acknowledged receipt of the petition and provided the NRC Staff's determination that the petition did not require immediate action by the NRC. Notice of receipt was published in the *Federal Register* on December 16, 1996 (61 Fed. Reg. 66,063).

On the basis of the NRC Staff's evaluation of the issues and for the reasons given below, the Petitioner's requests are denied.

### II. BACKGROUND

The Petitioner's first request is for the NRC to order WEPCO to maintain sufficient empty space in the spent fuel pool at Point Beach to accommodate the unloading of a VSC-24 spent fuel storage cask. NRC regulations include a requirement that an independent spent fuel storage installation (ISFSI) be designed to provide for the ready retrieval of spent fuel or high-level radioactive waste for further processing or disposal. This requirement is applicable to ISFSIs so that the stored spent fuel can be retrieved for transport to either a monitored retrievable storage installation (MRS) or a high-level waste repository whenever it becomes available. This regulation, 10 C.F.R. § 72.122(*l*), provides as follows:

> (l) Retrievability. Storage systems must be designed to allow ready retrieval of spent fuel or high-level radioactive waste for further processing or disposal.

In addition to the regulatory requirements in section 72.122(*l*) pertaining to retrieval of the fuel assemblies for further processing or disposal, there are certain events or conditions that could warrant removing a VSC-24 cask from an ISFSI and returning the multiassembly sealed basket (MSB) to the spent fuel pool and unloading the stored fuel assemblies. The COC requires a VSC-24 cask to be returned to the spent fuel pool in response to those design-basis events or conditions that may challenge the integrity of the storage cask or the cladding of the spent fuel assemblies.[1]

Petitioner's second request is for an NRC order to WEPCO and other users of VSC-24 casks not to load additional casks until the COC, the SAR, and the SER are amended to contain operating controls and limits to prevent hazardous conditions. On May 28, 1996, a hydrogen gas ignition occurred during the welding of the shield lid after spent fuel had been loaded into a VSC-24 cask at the Point Beach Nuclear Plant. The hydrogen was formed by a chemical reaction between a zinc-based coating (Carbo Zinc 11) and the borated water in the spent fuel pool. Following the event, the NRC issued confirmatory action letters (CALs) to those Licensees using or planning to use VSC-24 casks for the storage of spent nuclear fuel (i.e., Licensees for Point Beach, Palisades, and Arkansas Nuclear One). The CALs documented the Licensees' commitments not to load or unload a VSC-24 cask without resolution of material compatibility issues identified in NRC Bulletin 96-04, "Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks," dated July 5, 1996, and subsequent confirmation of corrective actions by the NRC. The Staff has acknowledged that the event demonstrated that the SAR and related NRC review, as documented in the SER, did not adequately address the use of a zinc-based coating and its reaction with the acidic water in spent fuel pools.

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[1] The following sections of the COC include requirements for returning a VSC-24 cask to the spent fuel pool and/or unloading the cask:

Section 1.2.3, "Maximum Permissible Air Outlet Temperature";
Section 1.2.10, "Time Limit for Draining the MSB";
Section 1.2.15, "Handling Height"; and
Section 1.3.4, "Thermal Performance."

Each section is discussed later in this Decision.

The Licensees using VSC-24 casks submitted to the NRC information on operating controls and limits to prevent hazardous conditions implemented in response to NRC Bulletin 96-04 and subsequent Staff inquiries. The submittals from the Licensees included evaluations of possible material interactions and provided descriptions of how procedures were revised. The revisions include controls for the environments that the casks encounter during use, requirements for inspections and environmental sampling, and additional precautions for various cask operations. The NRC Staff has evaluated these responses for Arkansas Nuclear One (ANO) and Point Beach and, as documented in the safety evaluations dated December 3, 1996, and April 8, 1997, determined that the operating controls and limits proposed by these Licensees are acceptable and satisfy regulatory requirements. By a separate letter also dated December 3, 1996, the Staff informed the Licensee for ANO that its corrective actions had been verified by inspections performed by the NRC Staff. Shortly thereafter, the Licensee initiated cask loading activities.[2] The NRC will perform inspections in the near future in order to verify corrective actions implemented at Point Beach. The review of responses to the bulletin related to Palisades is ongoing. Cask operations at Point Beach and Palisades continue to be limited by the Licensees' commitments described in CALs.

## III. DISCUSSION

As noted, the petition requests two actions be taken by the NRC. They are addressed below.

### Item 1: Order WEPCO to Retain Twenty-Four Spaces in the Point Beach Spent Fuel Pool

The first requested action calls for the NRC to issue an order to WEPCO to retain twenty-four empty and available spaces in the Point Beach spent fuel pool to provide the capability to unload a VSC-24 dry storage cask. The two basic reasons to return a cask to the spent fuel pool would be either to (1) retrieve the fuel assemblies for further processing or disposal pursuant to section 72.122(*l*),

[2] The NRC Staff is looking into reports from Licensees on the need to perform weld repairs during the welding of the shield lid into the MSBs of several VSC-24 casks. This potential problem is not related to the requested actions or supporting information cited in the petition. The NRC Staff determined that the issuance of this Director's Decision should not be delayed pending resolution of potential problems associated with the weld repairs because the weld repairs are not related to concerns presented in the petition and the welding issue is being addressed by ongoing NRC activities. The Petitioner was informed of the welding issue and the NRC Staff's decision to not include the issue in the Staff's evaluation of the petition.

or (2) respond to an event or condition that has potentially degraded the cask or spent fuel in regard to the requirements established in the COC.

As previously discussed, section 72.122(*l*) sets forth requirements pertaining to retrieval of the fuel for further processing or disposal; however, it provides no basis for the NRC to require a licensee to maintain a specified reserve capacity in the spent fuel pool. Licensees will have considerable opportunity to plan and schedule the activities associated with retrieving fuel assemblies from existing storage casks for transfer to other casks for further processing or disposal. This ability to control the activity includes either ensuring that existing spent fuel pool facilities will support the transfer or developing alternate approaches. Alternate approaches could involve, for example, making room in spent fuel pools by use of other storage or transportation casks, expanding the wet storage capacity by making changes to the spent fuel pool or other parts of the reactor facility, or development of a system for direct cask-to-cask transfer under dry conditions. Therefore, the design requirement for ready retrieval in section 72.122(*l*) does not provide a basis for issuing an order as requested by the Petitioner.

Similarly, requiring the Licensee to maintain space in the spent fuel pool is not necessary as a contingency for certain events or conditions for which a cask must be returned to the spent fuel pool to facilitate inspections or ensure adequate cooling of the fuel assemblies. During its reviews performed during certification of the VSC-24 design, the NRC Staff confirmed that the design features of the cask provide reasonable assurance that the cask and fuel assemblies will confine the radioactive materials following the design-basis events established for dry storage casks. These design features include the confinement function provided by the welded MSB, the cooling and shielding functions provided by the ventilated concrete cask (VCC), the limitations on the fuel to be stored, and other cask characteristics and limitations placed on its use that were relied upon during the NRC's certification of the cask. Although the NRC Staff considered it prudent to require a cask to be returned to the spent fuel pool to ensure cooling of the spent fuel and support inspections to confirm that the cask could remain in service following certain design-basis events, the ability of the VSC-24 casks to withstand such events made it unnecessary for the NRC to include specific time constraints in which the operation needed to be completed.[3]

In the event that a condition would arise requiring a cask to be returned to the spent fuel pool, the continued confinement of the radioactive materials within the MSB would afford the Licensee ample time to develop corrective actions that would maintain safe storage conditions and minimize occupational exposures.

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[3] The position that a time-urgent unloading of a cask need not be considered is also supported by the analysis of a hypothetical event involving the failure of the stored fuel pins with subsequent ground-level breach of an MSB that was presented in the SAR for the VSC-24 design. Although no identified accident results in such failures, the event was analyzed to demonstrate the limited radiological consequences from accidents involving VSC-24 casks.

The design features of the cask, the unlikely nature of events that may require unloading a cask, and the NRC Staff's judgment that Licensees could develop an alternate approach if a spent fuel pool could not support an immediate unloading of a cask have previously been cited as reasonable justification for not requiring Licensees to maintain a fixed reserve capacity in spent fuel pools.[4]

Requirements defining conditions for returning a cask to the spent fuel pool were included in the COC for the VSC-24 cask in order to maintain the cask components and stored spent fuel assemblies within the boundaries evaluated and accepted by the NRC Staff during the certification process. The COC addresses those events or conditions that might lead to degradation of the cask or fuel assemblies. The required actions normally include restoring operations to within the acceptable limits or otherwise ensuring the spent fuel is placed in a safe storage condition. The COC requirements for some events or conditions include returning the MSB to the spent fuel pool to provide a safe storage condition and unloading of the spent fuel assemblies in order to support inspections of the cask.

The COC-required action in section 1.2.10, "Time Limit for Draining the MSB," states that a cask should be returned to the spent fuel pool for cooling if the water cannot be drained within the specified time after the MSB is removed from the spent fuel pool with twenty-four spent fuel assemblies. The referenced draining operation is part of the cask-loading sequence and it is reasonable to assume, therefore, that the cask-loading area within or adjacent to the spent fuel pool would be available for the cask should this contingency need to be implemented. Further, the COC-required action is meant to restore cooling to maintain safety margins pertaining to fuel assembly subcriticality and can be accomplished without unloading the fuel assemblies from the MSB. It is likely, however, that the locations in the spent fuel pool that had contained the fuel assemblies loaded into the storage cask would remain available during the loading and draining of the cask.

Section 1.2.15, "Handling Height," requires fuel assemblies to be returned to the spent fuel pool, and inspections and evaluations performed for cask components in the event a loaded cask is dropped from a height greater than 18 inches. The COC prohibits handling of a loaded VCC at a height greater than 80 inches. The NRC evaluation of the MSB drop analysis concurred that drops up to 80 inches of the MSB inside the VCC can be sustained without breaching the confinement boundary, preventing removal of the spent fuel assemblies, or causing a criticality accident. However, it is deemed prudent to return the cask to the spent fuel pool to perform inspections and evaluations in the event a cask experiences a significant drop, which is considered to be a drop from a

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[4] *See* resolution of public comments published with rulemakings to add the VSC-24 cask (58 Fed. Reg. 17,948) and TN-24 cask (58 Fed. Reg. 51,762) to the list of NRC-certified casks.

height greater than 18 inches. The requirement to perform such inspections and evaluations was, therefore, included in the COC in the event that a cask were to be dropped during movement. However, since the most likely time for a cask drop event to occur would be during movement of a newly loaded cask to the ISFSI, it is reasonable to assume that the spaces in the spent fuel pool that had contained the fuel assemblies loaded into the cask would remain available. Moreover, even assuming for the sake of this analysis that the drop occurs when spaces might not be available in the spent fuel pool, reviews of the cask have shown that the cask and fuel will remain intact following a drop from the maximum allowable height. Because a drop from the maximum allowable height would not pose an immediate threat to the safety of the public or plant personnel, adequate time would be available for the Licensee to develop and implement approaches to perform the required inspections and evaluations if spaces were not available in the spent fuel pool to support an immediate unloading of the cask. Temporary shielding, loading the affected MSB into a spare VCC, placing the affected MSB into the cask loading area within or adjacent to the spent fuel pool, or other contingency actions could ensure safe storage conditions while the Licensee developed and implemented an approach to allow for the actual unloading of the cask that had been dropped.

The requirements contained in sections 1.2.3, "Maximum Permissible Air Outlet Temperature," and 1.3.4, "Thermal Performance," were included in the COC to provide reasonable assurance that the temperatures of the fuel cladding and the VSC-24 concrete do not exceed design limits. Concrete temperature limits are intended to prevent gradual degradation of the VCC and the shielding it provides for the MSB, which is the containment vessel for the spent fuel. Other temperature limits pertain to the fuel cladding and are intended to maintain the stored fuel assemblies below the temperatures at which damage might occur. However, in the event that excessive temperatures are detected, cooling of the cask and subsequent placement of the MSB into the spent fuel pool, if necessary, are sufficient to avoid immediate safety concerns. Because safe storage of the fuel assemblies is achieved by placing the affected MSB into the cask loading area adjacent to or within the spent fuel pool, the actual unloading of the assemblies from the MSB to the storage racks within the spent fuel pool can await the Licensee's development of alternative approaches if that were necessary due to a lack of storage space in the spent fuel pool. Such approaches may require the Licensee to make modifications to the spent fuel pool or other parts of the reactor facility.

In addition to the specific COC requirements previously discussed, a cask might need to be returned to the spent fuel pool if the cask fails to meet some criteria provided in NRC regulations or the COC and should, therefore, be removed from service. Tests and surveillances performed before and after loading spent fuel into a storage cask are designed to detect failures to conform to design or regulatory requirements before a problem presents an imminent threat to the cask or stored fuel. Therefore, while discovery of a nonconformance or previously unidentified vulnerability may require removing a cask from service as part of a Licensee's corrective actions, it is highly improbable that the discovery of such a condition would pose an immediate safety concern. As in the previous examples, safe storage of the spent fuel could be accomplished by returning the affected MSB to the cask loading area within or adjacent to the spent fuel pool and the MSB and spent fuel could remain there while the Licensee determined an appropriate course of action, including provisions for unloading the cask, if necessary.

In sum, no credible accident has been identified that would require the immediate unloading of a storage cask as a necessary protective measure to avoid significant radiological consequences to members of the public. In addition, there is no event or condition that was identified during the certification of the VSC-24 cask that would require a time-urgent unloading of a cask. Therefore, there is no need for NRC to require continuous availability of space in the spent fuel pool to accommodate the potential need to unload a cask. Further, the NRC Staff has reasonable assurance that Licensees could, if necessary, develop and implement an approach to unload a cask if required to do so by unplanned events or conditions, such as those identified in the COC. If space is not immediately available in the spent fuel pool, there would be time to make it available by relocating other spent fuel assemblies or removing them for temporary storage in a cask or by making modifications to the spent fuel pool or other parts of the reactor facility. Therefore, the NRC does not see a need to require the Licensee to reserve a fixed number of vacant spaces in the spent fuel pool or to maintain the capability to retrieve the spent fuel from a cask within a specified period of time, particularly when there is no such prescriptive requirement stated in NRC rules.

**Item 2: Order VSC-24 Users Not to Load Casks Pending Amendment of Documents**

The Petitioner's second request was for the NRC to order all users of the VSC-24 cask not to load VSC-24 casks until the COC, the SAR, and the SER are amended to contain operating controls and limits that prevent hazardous conditions. As noted previously, following the event at Point Beach, the NRC Staff recognized that additional evaluation of potential material interactions was warranted for all transportation and storage casks. In regard to the VSC-24 cask, the event and subsequent NRC inspections made it apparent that actual changes in the operating procedures or the design of the cask would be necessary. CALs were issued to confirm Licensees' commitments to refrain from loading VSC-24 casks pending completion of the Staff's review of the responses to NRC Bulletin

96-04 and verification of the associated corrective actions. As discussed, the CALs established a process by which the NRC Staff could obtain confidence that operating controls and limits to address potential hazardous conditions are developed and implemented by each Licensee using VSC-24 casks.

In particular, the CAL process ensures that Licensees will incorporate the necessary operating controls and limits into revised plant procedures. Moreover, under existing NRC requirements, the Licensee must adequately implement those revised procedures. For this reason, no changes to the COC or the SAR are needed to ensure that enforceable operating controls and limits are in place to address potential hazardous conditions during the loading or unloading of a cask. Further, as previously indicated, the Staff has documented the process, information, and results of its review of the Licensee's response to Bulletin 96-04 for use of the VSC-24 at ANO and Point Beach in safety evaluations available for public review. The NRC Staff is currently reviewing the responses to the bulletin submitted by the Licensee for Palisades.

Although the actions taken as part of the CAL process provide adequate assurance that technical and regulatory compliance issues raised by the event at Point Beach will be resolved before a Licensee loads or unloads a VSC-24 cask, the NRC Staff agrees with the Petitioner that it would be beneficial if the SAR and other licensing-basis documents accurately described the identified chemical reaction and the associated operating controls and limits. The NRC Staff is currently reviewing a proposed amendment to the SAR and the COC for the VSC-24 cask design and will ensure that the information related to the identified chemical reaction and associated operating controls is adequately addressed in the appropriate licensing-basis document. In addition, the NRC Staff is processing a petition for rulemaking, PRM-72-3, that may lead to additional updating of ISFSI SARs and the inclusion of information on operating controls and limits implemented as a result of the event at Point Beach. However, the previously discussed controls to be implemented by the Licensees and verified by the NRC Staff as part of the CAL process, and the enforceability of those controls under existing NRC requirements, make it unnecessary to require revision of the specific licensing documents cited by the Petitioner as a precondition for resuming cask operations at the facilities using VSC-24 casks.

## IV. CONCLUSION

The Petitioner requested that the NRC (1) require WEPCO to retain twenty-four empty and available spaces in the Point Beach Nuclear Plant spent fuel pool to accommodate retrieval of spent fuel from a VSC-24 cask, and (2) prohibit loading of VSC-24 casks until the COC, the SAR, and the SER are amended to contain operating controls and limits to prevent hazardous conditions. Each of the claims by the Petitioner has been reviewed. I conclude that for the reasons discussed above, no adequate basis exists for granting the Petitioner's request for either (1) requiring the Licensee for Point Beach to reserve a fixed number of vacant spaces in the spent fuel pool or (2) suspension of the Licensees' use of the general license for dry cask storage of spent nuclear fuel at Palisades, Point Beach, or Arkansas Nuclear One pending revision of the SAR, the SER, and the COC for the VSC-24 cask.

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

Samuel J. Collins, Director
Office of Nuclear Reactor
Regulation

Dated at Rockville, Maryland,
this 17th day of April 1997.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS**

**Carl J. Paperiello,** Director

**In the Matter of** **Docket No. 040-07102**

**SHIELDALLOY METALLURGICAL CORPORATION (Newfield, New Jersey)** **April 15, 1997**

By an undated letter received October 11, 1996, and supplemented by a letter dated February 7, 1997, Mr. Sherwood Bauman, Chairperson of Save Wills Creek (Petitioner), requested modification of Shieldalloy Metallurgical Corporation's (SMC) license to allow only possession of radioactive material for the express purpose of decommissioning and decontaminating its Newfield, New Jersey facility, and further requested that current operations at the facility that result in additional radioactive material being stored at the site be halted. The request was considered as a petition submitted pursuant to 10 C.F.R. § 2.206.

In a Director's Decision dated April 15, 1997, the Director of Nuclear Material Safety and Safeguards granted in part and denied in part the relief sought by Petitioner. The Director concluded that concerns regarding SMC's proposed decommissioning funding plan warranted conditioning SMC's license as part of any future renewal to require SMC to provide additional proof of a proposed slag disposition method, in the form of an NRC-approved export application, within 1 year of the license's renewal. Additionally, any renewed SMC license will require financial assurance commensurate in value with the costs of offsite disposal for future source-material possession increases. The Director also concluded that Petitioner had otherwise failed to provide a basis to warrant modification of SMC's license in the manner requested or to halt current operations.

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

## I. INTRODUCTION

In an undated letter addressed to U.S. Nuclear Regulatory Commission ("NRC") Chairman Shirley Jackson and received on October 11, 1996, Sherwood Bauman, Chairperson of Save Wills Creek ("Petitioner"), requested that the NRC take action with respect to NRC Licensee Shieldalloy Metallurgical Corporation ("SMC"), of Newfield, New Jersey. The Petitioner requested, pursuant to 10 C.F.R. § 2.206, that the NRC modify SMC's license to allow only possession of radioactive material for the express purpose of decommissioning and decontaminating its Newfield facility, and that current operations resulting in additional radioactive material being stored at the site be immediately halted. The Petitioner cites the lack of adequate financial assurance, as required by 10 C.F.R. § 40.36, as the basis for his request.

The Petitioner submitted a followup letter, addressed to the NRC Executive Director for Operations and dated February 7, 1997, reiterating the above request. In this letter, the Petitioner stated that SMC is attempting to reclassify wastes as potential resources for which the Petitioner believes there is no viable market. Furthermore, the Petitioner concludes that without a viable market and the resultant inadequate financial assurance for the company, SMC is jeopardizing the health and safety of the local Newfield community.

By letter dated November 14, 1996, I formally acknowledged receipt of the Petitioner's original correspondence and informed the Petitioner that his request was being treated pursuant to section 2.206 of the Commission's regulations. A notice of receipt of the petition was published in the *Federal Register* on Thursday, November 21, 1996 (61 Fed. Reg. 59,251). By letter dated March 7, 1997, I formally acknowledged receipt of the Petitioner's supplementary letter.

I have evaluated the Petitioner's request and have determined that, for the reasons stated below, the petition is granted in part and denied in part.

## II. BACKGROUND

At its Newfield, New Jersey facility, SMC processes pyrochlore, a concentrated ore containing columbium (niobium), to produce ferro-columbium, an additive/conditioner used in the production of specialty steel and superalloys. The pyrochlore contains, by weight, more than 0.05% natural uranium and thorium, which are source materials and therefore require an NRC license pursuant to 10 C.F.R. Part 40. SMC operates this process under the authority of NRC Source Material License No. SMB-743.

During the manufacturing process, the radioactive materials are concentrated in both high-temperature slag and baghouse[1] dust, which are then stored in the source-material storage yard at the site. The slag contains most of the licensed material. In a letter to the NRC, dated June 24, 1996, the Licensee indicated that the concentration of source material in the baghouse dust is, on average, less than the "unimportant quantity" source material threshold of 0.05% by weight, as described in 10 C.F.R. § 40.13(a),[2] and need not be treated as licensed material after it is removed from the site. The Licensee has stored source material in this manner at the Newfield site since the 1950s and has accumulated approximately 295,000 kilograms (kg) of thorium and 40,000 kg of uranium at the site. SMC's current license limits SMC to 303,050 kg of thorium and 45,000 kg of uranium. That license expired on July 31, 1985, and SMC has continued operations in accordance with its existing license under the timely renewal provisions of 10 C.F.R. § 40.42(a). The SMC site has been included in the NRC's Site Decommissioning Management Plan because it contains a large volume of contaminated material for which disposal may prove difficult.

The primary issue significantly delaying SMC's license renewal is SMC's ability to meet the financial assurance requirements of section 40.36.[3] To meet its obligation under section 40.36, SMC originally provided the NRC with a Letter of Credit, dated July 23, 1990, in the amount of $750,000 to serve as financial assurance pending completion of the NRC's review of SMC's decommissioning funding plan.

In September 1993, SMC notified the NRC that it had filed for bankruptcy under Chapter 11 of the U.S. Bankruptcy Code. At that time, SMC also informed the NRC that it could not provide an acceptable decommissioning funding plan for reaching unrestricted release limits[4] by disposing of all stored material in a

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[1] The baghouses contain filters comprised of cloth (or similar material) arranged in a tubular fashion in an enclosed housing. The effluent stream from the production area is blown through the filter bags, which trap the particulates on the collected material that builds up on the bags. As the buildup of material on the bags increases, so too does resistance to flow. For that reason, the baghouse filters are equipped with shaking/vibrating devices to remove the collected dust and recondition the bags. The rated efficiency of the filters used in the D-111 baghouses is over 99%.

[2] Under section 40.13(a), any person is exempt from the requirements of 10 C.F.R. Part 40 and from the requirements for a license under section 62 of the Atomic Energy Act to the extent that such person receives, possesses, uses, transfers, or delivers source material in any chemical mixture, compound, solution, or alloy in which the source material is by weight less than 0.05% of the mixture, compound, solution, or alloy.

[3] The NRC's financial assurance requirements in section 40.36, as pertain to SMC's Newfield license, state that:

> (a) Each applicant for a specific license authorizing the possession and use of more than 100 mCi of source material in a readily dispersible form shall submit a decommissioning funding plan [DFP] as described in paragraph (d) of this section.
>
> * * * * *
>
> (d) Each [DFP] must contain a cost estimate for decommissioning and a description of the method [such as a prepayment, a surety, or an external sinking fund as described in § 40.36(e)] of assuring funds for decommissioning.

[4] The NRC's guidance for unrestricted release limits can be found in "Disposal or Onsite Storage of Thorium or Uranium Wastes from Past Operations" (46 Fed. Reg. 52,061 (Oct. 23, 1981)).

licensed disposal facility. Despite SMC's filing for bankruptcy and continued efforts to satisfy the NRC's financial assurance requirements, SMC has and continues to maintain public health and safety at its Newfield facility during continued operations under its existing license. Therefore, the status of current public health and safety protection is not at issue in this case.

By letter dated December 12, 1995, SMC submitted a new decommissioning funding plan to the NRC, proposing that the licensed slag be exported for use in steel production. The decommissioning funding plan also proposes that SMC sell the baghouse dust domestically (for cement manufacturing) without restriction because it is, on average, less than the 10 C.F.R. § 40.13(a) "unimportant quantity" threshold described above. Finally, under the new decommissioning funding plan, SMC would decontaminate and decommission the remainder of the Newfield site, after offsite shipment of the aforementioned products and in accordance with the NRC's unrestricted release criteria, by disposing of remaining contaminated structures and soils in a licensed disposal facility.

In December 1994, SMC submitted an application to the NRC for a license to export a test shipment of slag to a steel mill in Trinidad. The NRC's review of the export license application became moot in early 1996 when public concern in Trinidad led SMC's potential customer to reconsider purchasing the material. SMC has unofficially indicated to the NRC that it is currently negotiating with other steel mills and will likely revise its export application for export to steel mills in one or more countries during 1997.

By letter dated June 24, 1996, SMC requested permission for the proposed domestic sale and transfer of the baghouse dust to unlicensed persons; the Staff is currently reviewing the request.

## III. DISCUSSION

The Petitioner cites the lack of adequate financial assurance, as required by section 40.36, as the basis for his request. The Petitioner states that SMC is attempting to reclassify wastes as potential resources for which the Petitioner believes there is no viable market. Furthermore, the Petitioner concludes that lacking both a viable market and adequate decommissioning funding, SMC is jeopardizing the health and safety of the local Newfield community. To support his request, the Petitioner presents three factors he believes are relevant to his petition:

1. The Petitioner stated that the NRC's draft environmental impact statement, dated July 1996, for SMC's Cambridge facility (Docket 040-8948), discussed an identical proposal to sell slag from the Cambridge site. As part of that discussion, the Petitioner noted that the NRC Staff stated

that SMC could not actually demonstrate that SMC's proposal for sale of ferro-columbium slag at the Cambridge site is a workable and viable option.

2. The Petitioner also stated that to prove the lack of marketability for sale of ferro-columbium, the NRC could determine whether or not potential customers in the United States would require a license to possess the material in question. The Petitioner believes that few, if any, domestic companies will be willing to obtain any NRC licenses that may be required for the use of this material.
3. Finally, the Petitioner stated that the only customer SMC has been able to locate, to date, was not in the United States, but in an underdeveloped third-world country with little protection. After adverse publicity in the affected country, the facility purchasing the material canceled its order, and SMC has been unable to develop a new market during the succeeding 3 years.

## A. Regulatory Framework

### *1. Summary of 10 C.F.R. § 40.36*

Under section 40.36, a licensee is required to submit a detailed decommissioning funding plan, describing both the plan for decommissioning the site upon termination of operations and the method of assuring funds to complete the actions described in the decommissioning plan. The purpose of this requirement is to ensure that a licensee possesses sufficient funds to eventually decontaminate and decommission the site to a level at which public health and safety is assured. This rule was originally implemented in 1990. The NRC generally requires its licensees to provide financial assurance sufficient to decommission a site for unrestricted release consistent with the definition of decommissioning in 10 C.F.R. § 40.4. To meet these unrestricted release criteria, licensees generally transfer any radioactive waste generated during decommissioning to a licensed disposal facility. However, in some cases the Staff has used its discretion to accept lesser amounts of financial assurance, based on a finding of the acceptability of alternative approaches (e.g., *in-situ* disposal) or a binding commitment (such as a license condition or NRC order) from the licensee to pursue alternative approaches. In cases that involve a major federal action and where the potential environmental impacts of the alternative approaches may be significant, the NRC prepares an Environmental Impact Statement (EIS) and Record of Decision in accordance with the requirements of 10 C.F.R. Part 51.

## 2. *Application of 10 C.F.R. § 40.36 to License No. SMB-743*

Prior to 1990, the NRC did not require financial assurance for decommissioning from its licensees. During the period prior to the rule's implementation, SMC amassed large quantities of slag at the site contaminated with source material. Because SMC was in timely renewal at the time, SMC was only required to provide certification of financial assurance for $750,000 to meet the financial assurance requirements pursuant to 10 C.F.R. § 40.36(c)(2).

In 1993, after SMC notified the NRC that it could not provide adequate financial assurance to meet unrestricted release limits, the NRC began to develop an EIS for the decommissioning of the SMC Newfield site in response to the Licensee's request to dispose of the contaminated slag and baghouse dust *in situ*. The NRC suspended EIS development in 1995 when the Licensee informed the NRC of its intent to transfer the slag for use in steel smelting and the baghouse dust for other, nonlicensed purposes.

In December 1995, SMC submitted a modified decommissioning funding plan. That plan proposes that the licensed slag be exported for use in steel production as a fluxing agent that also removes impurities from the steel mixture, the result being a derived slag containing the impurities including the source material. This derived slag would be sold as an aggregate with no restrictions, because the concentrations of uranium and thorium would be, on average, well below the NRC's 10 C.F.R. § 40.13(a) "unimportant quantity" limit. The concentration of source material in the derived slag is less than in SMC's slag because it is diluted with other inert materials (such as lime and alumina) during the smelting process. The latest decommissioning funding plan also proposes that SMC sell the baghouse dust domestically for other purposes (e.g., cement manufacturing) without restriction because the contaminated baghouse dust would also be, on average, less than 0.05% of source material by weight. By letter dated June 24, 1996, SMC requested permission for the proposed domestic sale of the baghouse dust; the Staff is currently reviewing the request. Finally, under the new decommissioning funding plan, SMC would decontaminate and decommission the remainder of the Newfield site to conform to the NRC's unrestricted release limits; contaminated structures, soils, and radioactive wastes generated during decontamination and decommissioning would be sent to a licensed disposal facility. SMC calculated the cost for executing the decommissioning activities described in the 1995 modified decommissioning plan to be slightly less than $750,000.

The NRC has held a Letter of Credit for $750,000 from SMC, pursuant to 10 C.F.R. § 40.36(c)(2), since 1990. On February 26, 1997, at SMC's request, the NRC drew upon the Letter of Credit and is currently holding the funds in trust.[5] Because SMC has in place the required decommissioning funding plan and a financial assurance mechanism that encompasses the cost estimates to perform the actions proposed in the decommissioning funding plan, SMC is considered to be in compliance with section 40.36 until such time as the NRC determines whether the submitted decommissioning funding plan is acceptable (as discussed below). Therefore, the issue being decided herein is whether the Licensee's current decommissioning funding plan is acceptable.

**B. Acceptability of Decommissioning Funding Plan**

In SECY-96-210, dated October 1, 1996, the NRC Staff informed the Commission of its concerns regarding the acceptability of SMC's decommissioning funding plan and described its plan to resolve the associated issues. As part of its plan, the Staff informed the Commission of its intent to permit interim acceptance of the decommissioning funding plan to allow renewal of the license; however, the Staff's plan also requires that SMC present adequate evidence (e.g., obtaining NRC approval of an export license application) regarding the marketability of the slag within one year after renewal of License SMB-743. If SMC cannot provide such evidence, the NRC will reconsider the acceptability of the Licensee's decommissioning funding plan. This could include requiring the plan's revision to include a different approach for decommissioning and disposal of the radioactive slag (e.g., *in-situ* disposal). The NRC transmitted a copy of SECY-96-210 to the Petitioner as an enclosure to the November 14, 1996 acknowledgment letter.

In the Petitioner's February 7, 1997 supplementary letter, the Petitioner elaborates upon his belief that the current decommissioning funding plan should be considered unacceptable and the Licensee is not in compliance with the regulations in section 40.36 by stating that SMC's proposed plans to disposition the slags are neither technologically nor financially viable.

The Petitioner argues that the NRC has already stated that the sale of ferro-columbium slag is not viable, as referenced in the "Draft Environmental Impact Statement on Decommissioning of the Shieldalloy Metallurgical Corporation, Cambridge, Ohio," NUREG-1543, July 1996 (Draft EIS). This is not correct.

The respective viabilities of the Newfield and Cambridge ferro-columbium slags for use in steel production are considered by the NRC to be different in each case. As stated below, the Newfield ferro-columbium slag was produced using the same process that produced a previously marketed Newfield ferro-vanadium slag, demonstrating that the process using the Newfield ferro-columbium slag appears to be viable. In contrast, the Cambridge ferro-columbium slag was pro-

[5] To facilitate its planned exit from bankruptcy proceedings and with the Bankruptcy Court's approval, SMC requested by letter dated October 25, 1996, that the NRC draw upon the existing Letter of Credit.

duced using a different process and different feedstock materials. Consequently, the metallurgical properties of the Cambridge slags have not yet been demonstrated to be technologically viable. For this reason, the export sale alternative was not included for consideration in the Draft EIS for decommissioning of the Cambridge site.

With regard to the previously marketed ferro-vanadium slag, SMC delivered, on average, 7000 tons of ferro-vanadium slag per year to the domestic steel industry from 1991 to 1995, with the highest annual amount reaching 9000 tons. By comparison, SMC currently stores approximately 70,000 tons of ferro-columbium slag at its Newfield site. The licensed ferro-columbium slag at the Newfield site was produced in a manner similar to the ferro-vanadium slag. SMC's extensive metallurgical evaluations indicate that the ferro-columbium slag has metallurgical properties relating to the proposed steel process that are similar, if not superior, to relevant properties of the ferro-vanadium slag.

The NRC Staff acknowledges the Petitioner's statement that the domestic use of ferro-columbium slag would likely require an NRC or Agreement State license for possession and use, thus possibly constraining domestic commercial interest in the product and thereby impacting the financial viability of the slag product. However, SMC is marketing the material to international locations where regulatory conditions may be less of a factor in determining the product's financial viability. As part of any international export application and prior to issuance of an export license, the NRC will inform the importing government of the proposed importation and use of the product containing the source material, in accordance with the International Atomic Energy Agency's Code of Practice on the International Transboundary Movement of Radioactive Waste.

Finally, the Petitioner argues that the only potential customer SMC has been able to locate, to date, has been in Trinidad. Because of internal country concerns, the customer purchasing the material canceled its order, and SMC has been unable to develop a new market during the succeeding years, thus significantly decreasing viability of the product. The NRC agrees with the Petitioner that this raises a concern as to the viability of the proposed decommissioning funding plan and therefore grants the Petitioner's request in part. The NRC intends to require, in the form of a license condition as part of any future license renewal, that SMC provide additional proof (in the form of an NRC-approved export application) of the viability of the proposed disposition method within 1 year of the license's renewal. If such proof is not forthcoming within the time limit, the NRC Staff plans to issue an order requiring the submission of a new decommissioning funding plan along with appropriate mechanisms for financial assurance. Furthermore, the NRC will include a condition in any renewed SMC license requiring SMC to provide financial assurance commensurate in value for the costs of offsite disposal for future source material possession increases. These two conditions are intended to prevent SMC from continuing to accumulate licensed material at the site in perpetuity without adequate financial assurance.

## IV. CONCLUSION

The Staff has carefully considered the request of the Petitioner. For the reasons discussed above, I conclude that no substantial public health and safety concerns warrant NRC action concerning the request. However, because the Staff is proposing to impose certain restrictions on the Licensee for reasons similar to those presented by the Petitioner, I grant the Petitioner's request to that extent and deny it in other respects.

A copy of this Decision will be placed in the Commission's Public Document Room, Gelman Building, 2120 L Street, NW, Washington, DC, and at the Local Public Document Room for the named facility. A copy of this Decision will also be filed with the Secretary for the Commission's review as provided in 10 C.F.R. § 2.206(c) of the Commission's regulations.

As provided by this regulation, the 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

Carl J. Paperiello, Director
Office of Nuclear Material Safety
and Safeguards

Dated at Rockville, Maryland,
this 15th day of April 1997.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

## OFFICE OF NUCLEAR REACTOR REGULATION

**Samuel J. Collins**, Director

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| **In the Matter of** | **Docket Nos. 50-245** |
| | **50-336** |
| | **50-423** |
| | **(License Nos. DPR-21** |
| | **DPR-65** |
| | **NPF-49)** |
| **NORTHEAST UTILITIES** | |
| **(Millstone Nuclear Power Station,** | |
| **Units 1, 2, and 3)** | **April 29, 1997** |

The Director, Office of Nuclear Reactor Regulation, has granted in part and denied in part a petition filed by Anthony J. Ross requesting that the Commission take action with regard to Millstone Nuclear Power Station. Specifically, the Petitioner requested that accelerated enforcement action be taken for violations at Millstone involving procedure compliance, work control, and tagging control. As a basis for his request, the Petitioner alleged that violations in these areas have increased significantly, that many of these violations had never been assigned a severity level, and that when the violations are considered collectively, escalated enforcement action is warranted due to the repetitive nature of the violations. For reasons fully explained in the Director's Decision, to the extent that the Petitioner requested that the NRC take action against the Licensee for violations in these areas, the petition has been granted; in other respects, the petition has been denied.

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

## I. INTRODUCTION

On October 28, 1994, Mr. Anthony J. Ross (Petitioner) filed a petition with the Executive Director for Operations pursuant to section 2.206 of Title 10 of the *Code of Federal Regulations* (10 C.F.R. § 2.206). By letter dated December 15, 1994, the NRC informed the Petitioner that he had not provided a sufficient factual basis to warrant action under section 2.206. The NRC stated that if the Petitioner wished the Staff to take action under section 2.206, he needed to provide more information describing the specific technical violations that he alleged the NRC had not adequately addressed. By letters dated January 15, February 8, and February 20, 1995, the Petitioner supplemented his petition by submitting lists of alleged violations. In the petition, the Petitioner requested that "accelerated enforcement action" be taken against Northeast Utilities (NU) for violations at Millstone[1] involving procedure compliance, work control, and tagging control. As a basis for his request, the Petitioner asserted that since August 1993, violations in these areas had increased significantly, that many of these violations had never been assigned a severity level by the NRC, and that when all of the violations are considered collectively, escalated enforcement action is warranted because of the repetitive nature of the violations.

On February 23, 1995, the NRC informed the Petitioner that the petition had been referred to the Office of Nuclear Reactor Regulation, and that action would be taken within a reasonable time regarding the specific concerns raised in the petition.

NU responded to the NRC on May 12, 1995, regarding the issues raised in the petition; the Petitioner submitted a response on July 11, 1995, regarding issues raised in the NU submittal.

On October 14, 1995, the Petitioner submitted a petition requesting that the NRC take immediate enforcement action consisting of immediate suspension of the licenses to operate the three units at the Millstone Station, and immediate imposition of the maximum daily civil penalty allowed because of the numerous continuing and repetitive violations committed by the Licensee since early 1989. The NRC informed the Petitioner by letter dated November 24, 1995, that because his October 14, 1995 Petition did not contain any new information but merely raised again the same issues as in his previous petition, his October 14,

[1] Northeast Nuclear Energy Company (NNECO/Licensee), an electric-power operating subsidiary of NU, holds licenses for the operation of Millstone Nuclear Power Station, Units 1, 2, and 3.

1995 Petition would be considered as an additional supplement to his January 15, 1995 Petition.[2]

## II. DISCUSSION

The Petitioner requested that "accelerated enforcement action" be taken against NU for violations at Millstone involving procedure compliance, work control, and tagging control. As a basis for his request, the Petitioner alleged that since August 1993, violations in these areas had increased significantly, that many of these violations had never been assigned a severity level, and that when these violations are considered collectively with violations that had been assigned a severity level, escalated enforcement action is warranted because of the repetitive nature of the violations. In his October 14, 1995 supplement to the petition, the Petitioner requested that the NRC suspend the Licensee's licenses to operate all three Millstone units, and impose a daily civil penalty until the Licensee can assure the public and NRC that there will be no more violations in certain areas.

In the petition and its supplements, the Petitioner provided numerous examples of what he believed were violations in the areas of procedure compliance, work control, and tagging control. The NRC had been aware of the examples described by the Petitioner. These examples were taken from NRC inspection reports dating back to 1989 and from other NRC documents. The NRC considered whether enforcement action should be taken for these violations in accordance with the guidance provided in the "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy) in effect at the time that the violations occurred.[3] As provided in the Enforcement Policy, the basic enforcement sanctions available to the NRC include Notices of Violation (NOVs), civil penalties, and orders of various types, including Suspension Orders. As further provided in the Enforcement Policy, for those cases in which a strong message is warranted for a significant violation that continues for more than one day, the NRC may exercise discretion and assess a separate violation and attendant civil penalty for each day that the violation continues.

In accordance with that guidance, some of the examples cited by the Petitioner were violations for which the NRC issued a NOV, but for the majority of the

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[2] The Petitioner also asserted in his October 14, 1995 Petition that, since many of the violations had been substantiated by the NRC inspectors and/or the Licensee, but have not been identified as violations by the NRC, the Office of the Inspector General (OIG) should conduct a full investigation of the NRC's neglect. In its November 24, 1995 letter, the NRC informed the Petitioner that this assertion would be referred to the OIG. In addition, in this letter, the Petitioner's request for immediate action was denied. The Petitioner's assertion of neglect by the NRC was referred to the OIG.

[3] The Enforcement Policy in effect at the time that the violations occurred was set forth at 10 C.F.R. Part 2, Appendix C. The Commission's present Enforcement Policy is described in NUREG-1600.

examples, no NOV was issued. In some instances in which no NOV was issued, the example was considered to be of only minor safety significance because it was not a violation that could reasonably be expected to have been prevented by the Licensee's corrective actions for a previous violation, it was or will be, corrected within a reasonable time, and it was not willful, and therefore, was not cited in accordance with the above-mentioned Enforcement Policy. With regard to other instances, the examples cited by the Petitioner did not constitute violations of NRC regulatory requirements, but instead were deviations from established procedures in non-safety-related areas, or simply constituted certain equipment problems or weaknesses in certain areas, which required further clarification or the attention of Licensee management.

Nonetheless, the NRC shares the Petitioner's concern about the number and duration of these examples of failures in the areas of procedural compliance, work control, and tagging control. If the NRC were to reassess the examples provided by the Petitioner, it is possible that many could be classified as repetitive violations under the Enforcement Policy.[4] However, the NRC has determined that these examples are indicative of a more significant problem; specifically, a programmatic breakdown in management at the Millstone facility.

The NRC has been aware of weaknesses in the Licensee's operations at Millstone, and has taken significant regulatory action as a result. Specifically, programmatic concerns in the areas of procedural compliance, work control, and tagging control, were among the programmatic weaknesses common to all three Millstone units, which were identified in the most recent systematic assessment of licensee performance (SALP) report of August 26, 1994. These weaknesses included continuing problems with procedure quality and implementation, the informality in several maintenance and engineering programs that contributed to instances of poor performance, and the failure to take proper corrective action at the site. Based on these identified weaknesses, the NRC continued its increased inspection and oversight activities at the facility.

On November 4, 1995, the Licensee shut down Millstone Unit 1 for a scheduled refueling outage. During an NRC inspection of licensed activities at Millstone Unit 1 in the fall of 1995, the NRC identified refueling practices and operations regarding the spent fuel pool cooling systems that were inconsistent with the updated Final Safety Analysis Report (UFSAR). The NRC sent a letter to the Licensee on December 13, 1995, requiring that, before the restart of Millstone Unit 1, it inform the NRC, pursuant to section 182a of the Atomic Energy Act of 1954, as amended, and 10 C.F.R. § 50.54(f), of the actions taken to ensure that in the future it would operate that facility according to the terms

[4] Section IV.B of the Enforcement Policy defines a repetitive violation as a violation that reasonably could have been prevented by a licensee's corrective action for a previous violation normally occurring (1) within the past 2 years of the inspection at issue, or (2) during the period within the last two inspections, whichever is longer.

and conditions of the plant's operating license, the Commission's regulations, and the plant's UFSAR.

In January 1996, the NRC designated the units at Millstone as Category 2 plants. Plants in this category have weaknesses that warrant increased NRC attention until the Licensee demonstrates a period of improved performance. In February and March 1996, the Licensee shut down Millstone Units 2 and 3, respectively, due to design issues. In response to (1) a Licensee root-cause analysis of inaccuracies in the Millstone Unit 1 UFSAR that identified the potential for similar configuration-management conditions at Millstone Units 2 and 3 and (2) design configuration issues identified at these units, the NRC issued letters to the Licensee, pursuant to section 50.54(f), on March 7 and April 4, 1996. These letters required that the Licensee inform the NRC of the corrective actions taken regarding design configuration issues at Millstone Units 2 and 3 before the restart of each unit.[5]

In June 1996, the NRC designated the units at Millstone as Category 3 plants due to additional inspection findings regarding design bases and design control, some of which were similar to the examples the Petitioner raised. Plants in this category have significant weaknesses that warrant maintaining them in a shutdown condition until the Licensee can demonstrate to the NRC that it has both established and implemented adequate programs to ensure substantial improvement. Plants in this category require Commission authorization to resume operations.

On August 14, 1996, the NRC issued a Confirmatory Order directing the Licensee to contract with a third party to implement an Independent Corrective Action Verification Program (ICAVP) to verify the adequacy of its efforts to establish adequate design bases and design controls. The ICAVP is intended to provide additional assurance, before each of the three Millstone units restart, that the Licensee has identified and corrected existing problems in the design and configuration control processes.

The guidelines for approving the restart of a nuclear power plant after a shutdown resulting from a significant event, a complex hardware issue, or a serious management deficiency are found in NRC Inspection Manual Chapter (MC) 0350, "Staff Guidelines for Restart Approval." MC 0350 states that the Staff should develop a plant-specific restart action plan for NRC oversight of each plant startup. The restart action plan is to include those issues listed in MC 0350 that the NRC restart panel has deemed applicable to the reasons for the shutdown. In the case of Millstone, the restart action plan will include those issues that the Petitioner has raised; specifically, procedure compliance, work

[5] By letter dated April 16, 1997, the NRC clarified the information it needed pursuant to section 50.54(f).

control, and tagging control. Therefore, the NRC Staff will thoroughly review these areas prior to the restart of each unit.

Following a determination that the relevant issues have been identified and corrected by the Licensee, the NRC Staff will make its recommendation for restart approval to the Commission regarding restart for each Millstone unit. Upon receipt of the Staff's recommendation, the Commission will meet to assess the recommendation and vote on whether to approve the restart of the unit.

In addition, during eight NRC inspections conducted between October 1995 and August 1996, more than sixty apparent violations of NRC requirements were identified at Millstone, some of which were similar to the examples the Petitioner raised. These apparent violations were discussed with the Licensee at a public predecisional enforcement conference held at the Millstone site on December 5, 1996. During the meeting, the Licensee stated that management failed to provide clear direction and oversight, performance standards were low, management expectations were weak, and station priorities were inappropriate. Following its evaluation of the information presented at the enforcement conference, the NRC will determine whether further enforcement action is warranted for these apparent violations.

In sum, the issues raised by the Petitioner are indicative of a more fundamental problem of inadequate management oversight at the Millstone facility. The NRC has been aware of this programmatic problem and weaknesses in numerous areas of the Licensee's program, including the areas of procedural compliance, work control, and tagging control, and has taken extensive regulatory action. In particular, as a result of action taken by the NRC, all three units at Millstone will remain shut down until the Commission approves restart of operations. Prior to such approval, the Licensee is required to submit a response to the NRC's section 50.54(f) letter dated April 16, 1997, identifying what actions the Licensee has taken to ensure that in the future it would operate that facility according to the terms and conditions of the plant's operating license, the Commission's regulations, and the plant's UFSAR. This response will encompass the areas identified by the Petitioner and will be thoroughly reviewed by the NRC. In addition, the NRC is currently reviewing the apparent violations that have been identified as a result of inspections conducted at the facility between October 1995 and August 1996, and, following its review, will take such enforcement action as it deems is warranted.

These actions go beyond those requested by the Petitioner. Therefore, to the extent that the Petitioner has requested that the NRC take action against the Licensee for violations at Millstone involving procedural compliance, work control, and tagging control, the petition has been granted. Given the action already taken by the NRC, the NRC has determined that the additional enforcement action requested by the Petitioner is not warranted at this time.

## III. CONCLUSION

The Staff has completed its review of the information submitted by the Petitioner in his petition and its supplements. The Staff has concluded that the actions taken by the NRC against NU are appropriate and encompass the Petitioner's examples of violations in the areas of procedural compliance, work control, and tagging control. To this extent, the Petitioner's requests for enforcement action against NU is granted, in part. In other respects, the petition is denied. As provided for in 10 C.F.R. § 2.206(c), a copy of this 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 review of the Decision in that time.

FOR THE NUCLEAR
REGULATORY COMMISSION

Samuel J. Collins, Director
Office of Nuclear Reactor
Regulation

Dated at Rockville, Maryland,
this 29th day of April 1997.